FROM RESEARCH TO INDUSTRY

cea

Nuclear Energy Division

REPORT ON SUSTAINABLE RADIOACTIVE WASTE MANAGEMENT

DECEMBER 2012

SUSTAINABLE RADIOACTIVE WASTE MANAGEMENT ACT OF JUNE 28, 2006: RESULTS OF RESEARCH CARRIED OUT ON THE SEPARATION AND TRANSMUTATION OF LONG-LIVED RADIOACTIVE ELEMENTS, AND ON THE DEVELOPMENT OF A NEW GENERATION OF NUCLEAR REACTORS

EXECUTIVE SUMMARY

Summary

The Sustainable Radioactive Waste Management Act of June 28, 2006, specified clear guidelines for spent nuclear fuel management. It states two complementary principles:

- The policy of treating and recycling spent nuclear fuel is valid for reducing the quantity and toxicity of suitably packaged ultimate radioactive wasteforms.
- The reference process for high-activity and long-lived ultimate waste is deep geological disposal.

Act 2006-739 dated June 28, 2006, supplemented by decrees on April 16, 2008, and April 23, 2012, implementing the French National Plan for the Management of Radioactive Materials and Waste (PNGMDR), called on the CEA to "coordinate research on partitioning and transmutation of long-lived radioactive elements":

- "in relation with research conducted on the new generations of nuclear reactors mentioned in article 5 of Act 2005-781 dated July 13, 2005" (which highlighted the importance of the nuclear industry for France, and called for research on future nuclear reactors), "as well as with accelerator-driven systems dedicated to waste transmutation";
- "in order to obtain an assessment of the industrial prospects of these approaches by 2012";
- and to "commissioning a prototype facility before December 31, 2020", for which the documents submitted in 2012 must also enable a "choice" of options.

The report prepared by the CEA in response to these requirements was completed after several years of work in cooperation with the other French actors in this field (EDF, AREVA) and with contribution of the CNRS and Andra. It addresses the following topics in several volumes:

- guidelines for research on 4th-generation systems, and a description of the various systems examined;
- the results of research coordinated by the CEA on partitioning and transmutation of long-lived radioactive elements;
- choices proposed for the Astrid integrated technology demonstrator – a sodium-cooled fast reactor (SFR) – and a reasonable timetable for its construction;
- a review of research conducted around the world on 4th-generation systems based on fast neutron reactors (FNRs).

The principal results and findings compiled by the CEA from these studies are summarized below.

1.

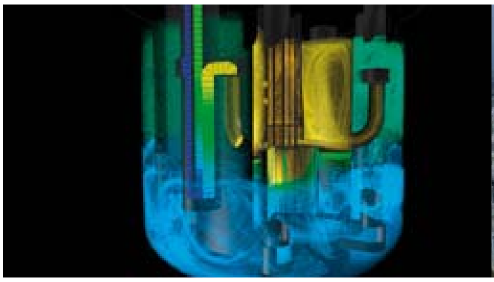
The development of 4th-generation systems, based on the use of fast neutron reactors to recycle uranium and plutonium contained in spent fuel, offers the prospect of producing electricity without greenhouse gas emissions, that is safe and economically competitive, and meets the objectives of sustainable management of nuclear material.

Systematic recycling of uranium and plutonium along the lines of current practice in France for its pressurized water reactors will offer a number of advantages:

- It will use and recycle all the existing stockpiled plutonium (which is difficult or even impossible today because of the degraded plutonium isotopic composition when recycling in thermal neutron reactors, making it unsuitable for repeated recycling in a fleet of light water reactors).
- It can indefinitely prolong the closed fuel cycle strategy that produces ultimate waste containing practically no plutonium, which is the radioelement that imposes the most severe constraints on a deep geological repository, especially after the first 500 years.
- It enables a dramatic improvement – by more than a factor of 100 – in the utilization of uranium resources. The known conventional uranium resources would then correspond to an energy potential 10 times greater than that of coal, oil, and gas combined.
- It will limit "at the source" the quantity of some long-lived radioactive products (minor actinides) contained in the final waste. For the same electricity production, recycling plutonium in a fast neutron reactor produces only one-fourth the amount of minor actinides as recycling in a light water reactor.

Multi-recycling of uranium and plutonium in fast neutron reactors is thus one of the keys to sustainable nuclear energy, by preventing the accumulation of sensitive materials, preserving natural resources, and limiting the ultimate waste volume. This is the primary objective of Astrid: to demonstrate the multi-recycle capability of reusable materials.

There is an international consensus on the possibilities of these systems. For nearly a decade the Generation IV International Forum (GIF) has analyzed the various possible systems to identify the main criteria for their development. Reactor safety is a key issue, and the objective of improving safety standards is one of the primary development objectives.



Simulation for the development of 4th-generation nuclear systems.

Overall view of La Hague reprocessing plant.

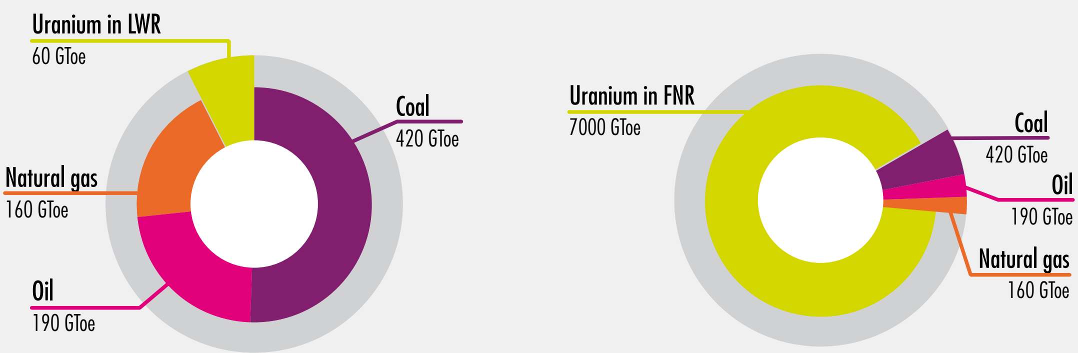
ENERGY POTENTIAL OF FOSSIL RESOURCES

Proven reserves of conventional fossil fuels were estimated in 2011 at 189 billion metric tons of oil, 187 trillion cubic meters of natural gas, and 860 billion metric tons of coal [source: BP Statistical Review of World Energy, 2011].

The proven conventional resources of uranium were estimated at 4 million metric tons [source: IAEA Red Book, 2009]. The figure below represents the energy potential of these resources expressed in billions of metric tons of oil equivalent (GToe):

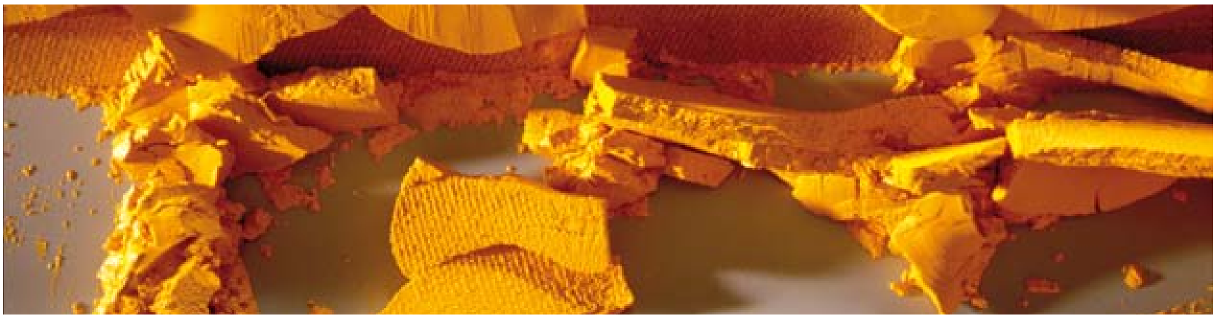
- In the chart on the left, for uranium as utilized today in light water reactors, this amounts to about 7% of the total fossil energy resources.
- The chart on the right corresponds to uranium utilization in fast neutron reactors; in this case, uranium becomes the first energy resource with a potential 10 times greater than the other fossil resources.

ENERGY VALUE OF URANIUM IN LIGHT WATER REACTORS (LEFT) AND FAST NEUTRON REACTORS (RIGHT)



Proven reserves:
Oil: 189 GT
Natural gas: 187 Tm^3
Coal: 860 GT
Source: BP Statistical Review of World Energy (June 2011)

Uranium: 4.0 MT
Source: IAEA Red Book, 2009 edition ("Reasonably assured resources")

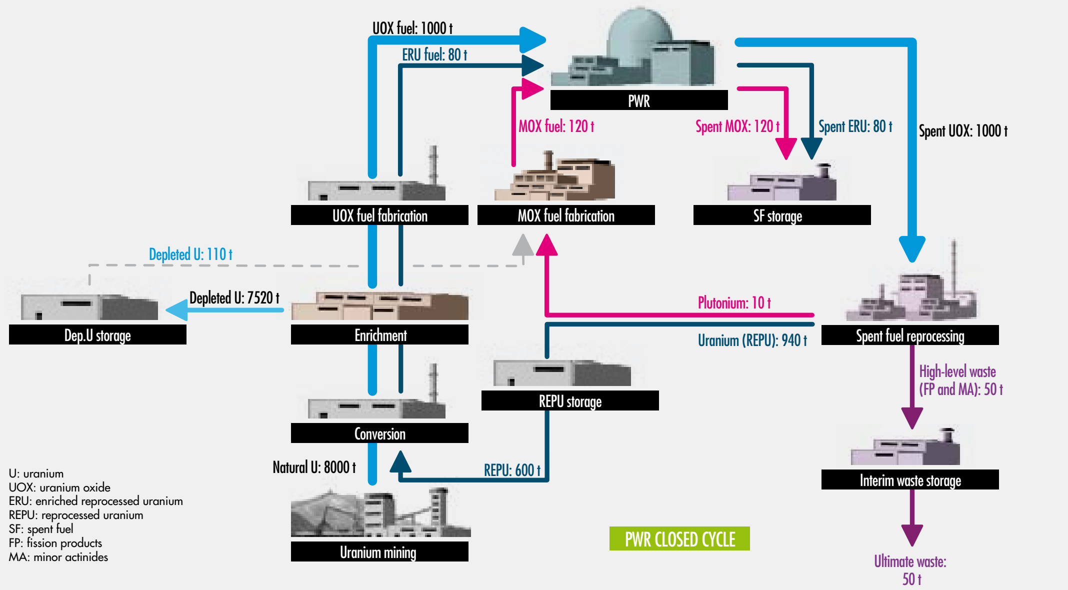


Uranium in the form of yellowcake.

THE NUCLEAR MATERIAL CYCLE

The operation of French nuclear power plants (63 GWe) represents an annual consumption of nearly 8000 metric tons of natural uranium, which must be enriched for fuel fabrication. The spent fuel discharged on completion of its irradiation cycle in the reactor (about 1000 metric tons each year) is reprocessed and the plutonium and uranium recovered are recycled respectively as MOX and ERU (enriched reprocessed uranium) fuel. The remaining materials (fission products and minor actinides) are the principal ultimate waste, and are immobilized in a glass matrix. Spent MOX and ERU fuel are not currently recycled, but are stored pending subsequent treatment.

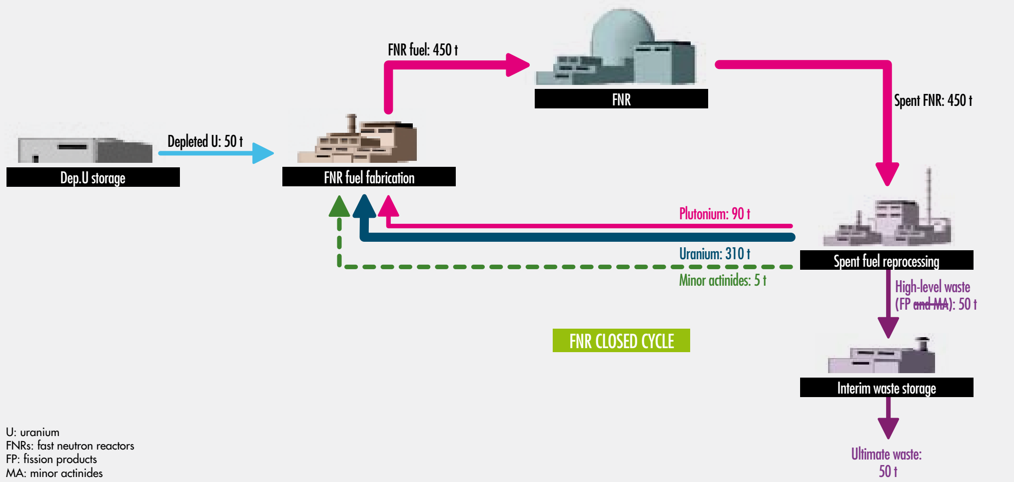
THE CURRENT NUCLEAR MATERIAL CYCLE IN A FLEET OF PWRs



A fleet of nuclear power plants with the same capacity but consisting of isogenerating[1] fast neutron reactors could multi-recycle uranium and plutonium indefinitely, and thus make use of the full energy potential of these materials. Only 50 metric tons of depleted uranium per year would be necessary to supply these reactors, for which the "front end" processes (uranium mining, conversion and enrichment) would no longer be necessary. Recycling can also be envisaged for some long-lived elements (americium in particular) to reduce the long-term radiotoxicity and thermal power of the ultimate waste.

1 – Reactors that produce as much fissile material as they consume.

THE NUCLEAR MATERIAL CYCLE IN A FLEET OF ISOGENERATING FNRs



1.

MULTI-RECYCLING OF URANIUM AND PLUTONIUM

Plutonium is the key to exploiting the full energy potential of nuclear fuel. Uranium-238 (99.3% of natural uranium) is not directly fissionable, but can be transformed by neutron irradiation in a nuclear reactor into fissile plutonium-239 (^{239}Pu): the overall energy efficiency of the fuel depends on the efficiency of this conversion, but also on the capability of the implemented systems to effectively fission the resulting plutonium.

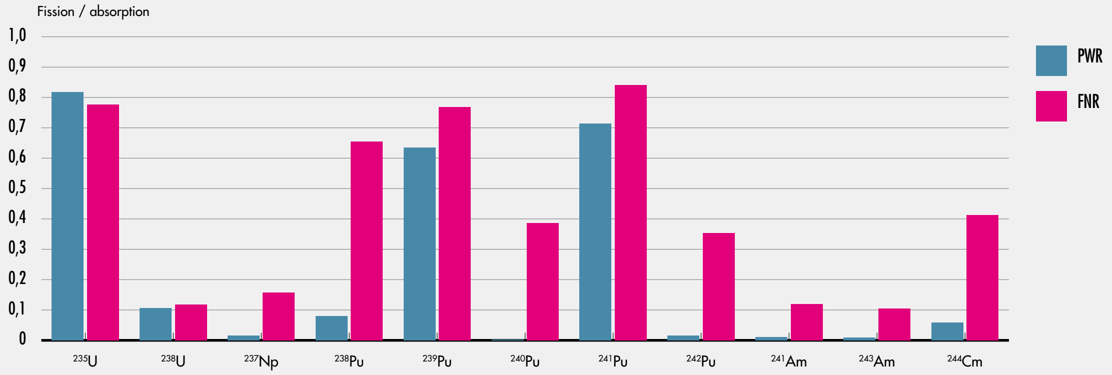
Two conditions must be met for this purpose: recycle uranium and plutonium, but also perform recycling in suitable reactors. Only fast neutron reactors permit repeated recycling of nuclear materials, and can thus make use of virtually all their energy potential.

Successive recycling modifies the isotopic composition of the residual plutonium, which becomes enriched in heavier isotopes (^{240}Pu to ^{242}Pu) through neutron capture reactions: this is particularly true in light water reactors, which are also very inefficient in fissioning the even-numbered plutonium isotopes. The accumulation of heavy isotopes currently limits the French PWR fleet to a single recycle with MOX fuel, and less than 1% of the total energy potential of the initial uranium is actually used.

Fast neutron reactors, on the contrary, ensure interactions between neutrons and the various plutonium isotopes, favoring fission over capture: the accumulation of heavier isotopes is limited, and successive recycling results in an equilibrium isotopic composition that allows sustainable long-term operation, and thus repeated recycling. Multi-recycling, which is possible only in FNRs, is the key to fully exploiting the potential of natural uranium.

Moreover, for same reasons – fewer capture reactions – recycling plutonium in FNRs results in significantly lower production of minor actinides, which are the main contributors to the potential radiotoxicity of the ultimate waste.

PROBABILITY OF FISSION OF ACTINIDES IN PWRs AND FNRs



2.

The ability to develop 4th-generation fast neutron reactor systems is an asset for France, not only for its long-term energy supply security but also for its industrial competitiveness and employment.

First of all, it ensures that French energy needs can be met by nuclear electricity production, founded on abundant and accessible resources (several thousand years of electricity production at current levels with the quantities of depleted uranium – the primary fuel of a FNR – and plutonium available in France today), and thereby helps to increase French energy independence.
At the same time, France can take advantage of its acknowledged expertise in the field of nuclear technologies. The development of a new generation of reactors and the international market for them – led by countries such as China, India, and Russia, which are determined to deploy tens of fast neutron reactors in the coming decades – is a remarkable opportunity for the French nuclear industry, particularly in terms of employment prospects. The FNR market is a major sector, but it is inseparable from the FNR fuel cycle: spent fuel reprocessing and plutonium fuel fabrication are areas in which French excellence is internationally recognized. This is a major asset for the deployment of fast neutron reactor technology for its own requirements, but also for playing a leading international role and ensuring an essential part of future developments, thereby confirming or even extending the export activities of the French nuclear industry in this area.

THE INTERNATIONAL FNR MARKET

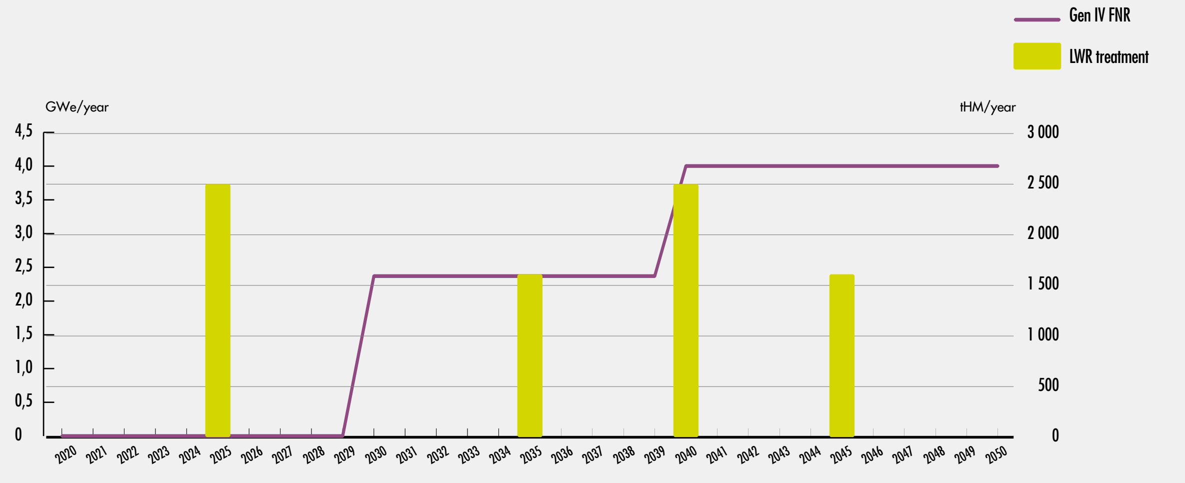
The world market for fast neutron reactors between 2030 and 2050 is estimated at one or two 1500 MWe reactors a year, assuming they do not become economically competitive before 2050. When they become competitive, 10 to 15 fast neutron reactors could be built around the world each year, according to the number of thermal neutron reactors planned for deployment during this period.

Initially the FNR market should follow national policies related to the issues of energy security and/or waste management. During this period economic competitiveness will probably not be the decisive selection criterion.

The date at which this technology will become competitive is unknown: it will depend on the volume of natural uranium resources and on their rate of consumption. Depending on the hypotheses involved, it will correspond to a natural uranium price of €250 to €600 per kg (compared with slightly less than €100 today). It could occur during the second half of this century.

It is important to note that the FNR market can develop only if a major effort is applied to spent fuel reprocessing for both FNR fuel and light water reactor (LWR) fuel. The market for spent fuel treatment and recycling is closely related to the market for fast neutron reactors, and must also be taken into consideration.

WORLD MARKET FOR GEN IV FNRs AND FOR SPENT LWR FUEL TREATMENT



The largest contributors to these markets are expected to be China, Russia, and India, all of which have a strong nuclear industry, experience with fast neutron reactors, and especially a firm political will to deploy FNR technology. Assuming this determination continues between 2030 and 2050, requiring a rigorous but realistic deployment program, the construction of FNRs can be expected to represent about 3 GWe per year on average, or roughly two 1500 MWe reactors (refer to the figure above). Over the same period, spent LWR fuel reprocessing plants will have to be built for an overall tonnage of about 8000 tons of "heavy metal" (uranium and plutonium) per year [tHM/year], or about five times the capacity of La Hague complex (UP3 + UP2 800).

The total sales revenue generated for French industry, for example assuming an agreement has been signed with another industrial partner or country involved in reactor development, would probably be higher for the fuel cycle facilities (where France holds a greater technological lead) than for the reactors themselves.

A sales figure of about €200 billion between 2025 and 2050 could be anticipated, with the creation of a large number of jobs.

3.

Two main 4th-generation FNR systems are being investigated in France: gas-cooled fast reactors (GFRs) and sodium-cooled fast reactors (SFRs). At the present time, SFRs appear to be the best solution for deployment during the first half of this century. They combine several essential advantages: the highest degree of industrial maturity based on extensive feedback, and clearly identified avenues for progress (prospects of major technological breakthroughs have appeared in recent years following research conducted by the CEA, especially in terms of safety). GFRs are a promising alternative in many respects, but still require significant research efforts before the construction of a technology demonstrator, let alone industrial maturity.

Two main pathways were identified and confirmed by the French government in 2005 for the design of future nuclear systems: SFRs and, over the longer term, GFRs.

SFRs have many advantages: sodium is an excellent coolant, suitable for power reactors, relatively noncorrosive and with excellent safety guarantees (especially with regard to thermal inertia: SFRs demonstrate excellent behavior in the event of incidents involving the loss of an external heat sink).

Sodium also has a number of drawbacks. They are well known, which allows the development of effective countermeasures. The disadvantages of sodium can be overcome and are not an obstacle to meeting the criteria of 4th-generation systems. Several lines of defense can be set up:

- The strong chemical reactivity between sodium and water can be mitigated "at the source" by eliminating water as the fluid for the final energy conversion (by designing gas turbines for use downstream from the sodium-cooled primary and secondary circuits).
- Special design and management features, including the heterogeneous fuel concept (in which the composition of the fuel pins varies with the height in the core) can mitigate the risk of a reactivity excursion in the event of a loss of coolant (frequently cited as "a design basis problem" of SFRs). This is a major breakthrough in the R&D conducted in recent years by the CEA on the "low void effect" core concept that limits the reactivity in the core in the event of a loss of sodium coolant.
- Additional provisions can be developed, such as the reactor shutdown system designed by the CEA using purely passive actuators to prevent the progression of incident situations.

SFRs benefit from extensive feedback – with nearly 400 reactor-years, including 100 in industrial operation – that have revealed several interesting characteristics, including the appreciably lower collective committed dose incurred by workers operating this type of reactor. They have been adopted today by all the countries engaged in the development of SFRs (Russia, China, India, Japan, South Korea, etc.).

Other fast neutron reactor concepts are possible: France is particularly interested in helium gas-cooled fast systems, which have the potential advantage of more favorable thermodynamic characteristics and open the perspective of high-temperature industrial heating applications. The CEA has conducted a major research program on suitable fuels, which must be compatible with particularly severe operating conditions due to the high operating temperatures in these systems. Given the difficulty of conducting research on several fronts,

the CEA has focused on sodium-cooled fast reactors. Gas-cooled fast reactor development work is carried out at a European level via the Allegro project, an experimental 75 MWth reactor for which a consortium has been formed with the long-term objective of building a reactor in central Europe.
Lead cooled fast reactors (LFRs) or lead alloys have been built in Russia for specific applications such as naval propulsion, and work is now in progress within the GIF. The high boiling point of lead and its low reactivity (with water in particular) are theoretical advantages, but the higher melting point of lead and especially the risk of accelerated corrosion of structural materials are significant drawbacks that have led the CEA to prefer sodium as the reference technology. Finally, from a much longer term perspective, molten salt reactors (MSRs) are also of interest. They are investigated in France by the CNRS, which is studying a Molten Salt Fast Reactor (MSFR) project using thorium fuel (instead of uranium) in the form of a fluoride salt as both fuel and coolant, which has a number of theoretical advantages. Major technological (or even conceptual) breakthroughs remain necessary, however, before this system reaches industrial maturity, which does not appear possible before the second half of this century.

THE GENERATION IV INTERNATIONAL FORUM

The Generation IV International Forum (GIF) is an intergovernmental association created in 2000 at the initiative of the United States. It currently includes thirteen members: Argentina, Brazil, Canada, China, EURATOM, France, Japan, Russia, South Africa, South Korea, Switzerland, United Kingdom, United States. The signatories of the Forum charter acknowledge the importance of developing future systems for nuclear energy production, as well as the need to better preserve the environment and prevent the risks of proliferation.

The GIF defines and manages coordinated programs of research and development in support of nuclear systems, which must meet the criteria of 4th-generation nuclear reactors: sustainable nuclear energy, savings of uranium resources, enhanced competitiveness and safety compared with the levels reached by 3rd-generation light water reactors, minimization of radioactive waste production, greater resistance to nuclear proliferation, use of nuclear energy for applications other than electricity production.

The GIF has selected six reactor concepts: sodium-cooled fast reactor (SFR), gas-cooled fast reactor (GFR), supercritical water-cooled reactor (SCWR), very high temperature reactor (VHTR), lead-cooled fast reactor (LFR) and molten salt reactor (MSR). Four of these concepts (SFR, GFR, LFR, and MSR) are based on fast neutron reactors and constitute sustainable nuclear options.

France has decided to contribute to the development of three of these systems:

- SFR (which also interests Russia, Japan, the United States, China, South Korea, and the European Union);
- GFR (the main alternative for France, also of interest to Japan, and benefiting from progress in high temperature reactors);
- MSR (only of interest to France and the European Union, over the very long term).

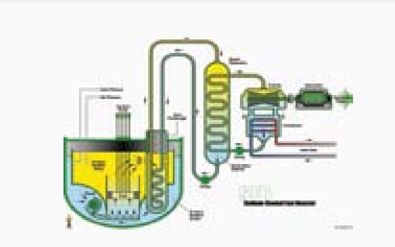
The LFR concept was not selected in France because of corrosion problems and the difficulties of operating cooling circuits with molten lead.

PARTICIPATION OF GIF MEMBERS IN VARIOUS PROJECTS (2012)

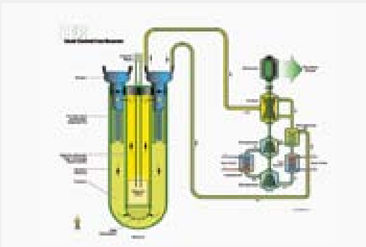
	Canada	European Union	France	Japan	South Korea	Switzerland	United States	China	South Africa	Russia
VHTR	●	●		●	●	●	●	●	●	
GFR		●	●	●		●				
SFR		●	●	●	●		●	●		●
SCWR	●	●		●						
LFR		●		●						
MSR		●	●							

SCHEMATIC DIAGRAMS OF THE SIX NUCLEAR SYSTEMS INVESTIGATED BY THE GIF

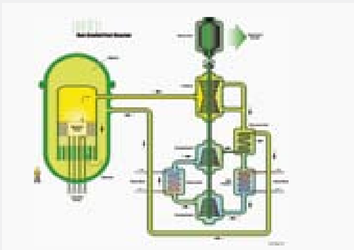
(Source: http://www.gen-4.org/)



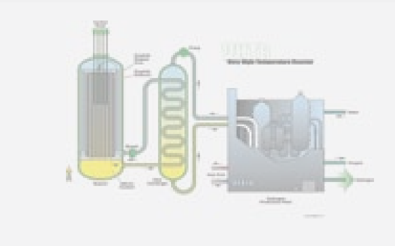
Schematic diagram of a SFR



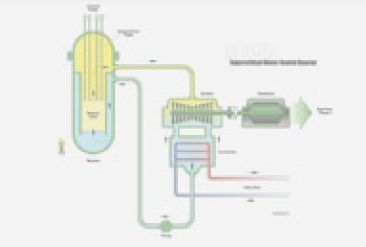
Schematic diagram of a LFR



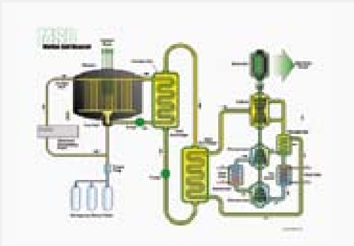
Schematic diagram of a GFR



Schematic diagram of a VHTR



Schematic diagram of a SCWR



Schematic diagram of a MSR

3.

INNOVATIVE SFR CORE

Sodium moderates, reflects, and captures neutrons. A local absence of sodium (sodium void) in the core results in a reactivity variation that is the sum of two opposing effects:

- a positive reactivity effect due to the decrease in neutron moderation and capture;
- a negative reactivity effect related to the increased neutron leakage from the core.

As the neutron leakage rate diminishes with the size of the core, the reactivity effect caused by a sodium void is positive in conventional large, high-power core designs. This raises safety concerns that are difficult to manage in SFRs.

To overcome this problem, a "low void effect core" is currently being designed as the reference core to ensure a negative reactivity effect in the case of sodium boiling, for example. This is made possible by:

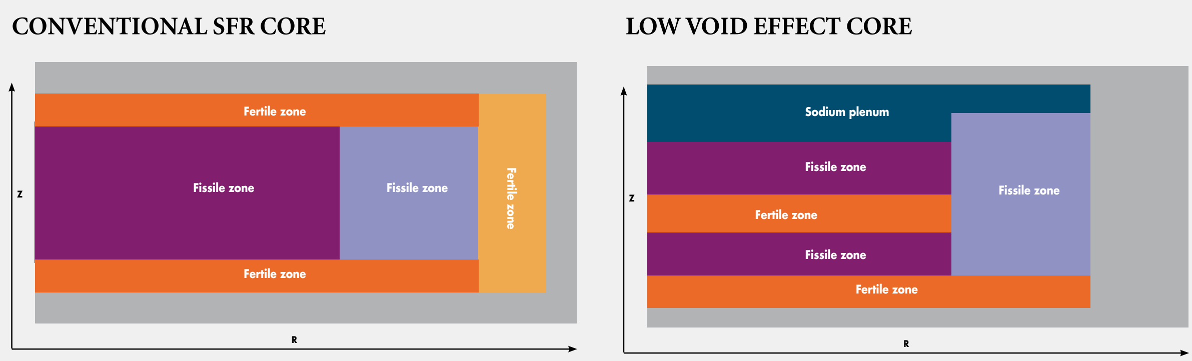
- a reduction in the sodium volume fraction in the core (by reducing the diameter of the spacer wire between fuel pins);
- the adoption of the "sodium plenum" concept, in the form of a cavity filled with sodium above the fuel pin bundle inside each fuel assembly. In a voiding situation, the plenum favors neutron leakage outside the core.

The innovation of the low void effect core is that it combines the concept of a sodium plenum with a heterogeneous core geometry (with the presence of a fertile plate at about mid-height in the core) and with the asymmetric "crucible-shaped" core concept (internal and external fissile zones of different heights). This combination accentuates the effect of neutron leakage from the plenum (tripling the plenum effect) and thereby offsets the increased reactivity in the event of sodium voiding.

With these features the natural behavior of the low void effect core can be expected to be favorable in the event of a loss-of-coolant accident: sodium boiling, which could potentially begin in the plenum (the hottest zone) would result in an overall negative reactivity effect and a diminishing power level.

An experimental physics program will be carried out in a zero power reactor in Russia (BFS) and France (Masurca, CEA) to develop expertise with the specific features of the low void effect core based on a combination of options and multiple effects, and to certify the uncertainties on the calculations of the main neutronic parameters.

This major innovation was presented to the French Academy of Sciences at the hearings following the Fukushima accident.



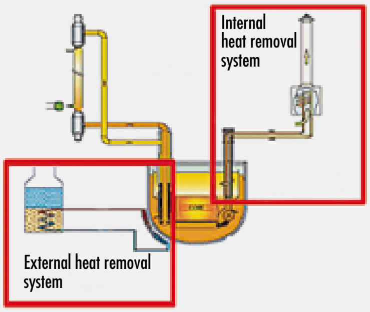
DECAY HEAT REMOVAL IN A SFR

Removal of the decay heat from the fuel after shutdown of the chain reaction is another essential aspect of nuclear reactor safety, as shown by the Fukushima accident.
Sodium offers several advantages in this respect:

- a large boiling margin (more than 300°C between the operating temperature and the sodium boiling point);
- very high thermal inertia, and a much lower heating rate in a loss-of-coolant situation.

The Astrid project is designed with multiple and diversified decay heat removal systems taking into account a synergistic approach combining active systems and passive systems. The objective is to make them more effective by emphasizing natural convection in the primary system and installing the heat exchangers near the heat sources to enable core cooling even in case of failure of the electric power supply and pumps.
The issue of decay heat removal was presented to the French Academy of Sciences as part of an analysis of the lessons learned from the Fukushima accident.

DECAY HEAT REMOVAL PROVISIONS

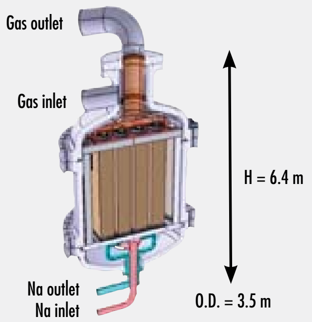


SIMULATION OF NATURAL CONVECTION IN THE REACTOR VESSEL IN A LOSS-OF-COOLANT ACCIDENT

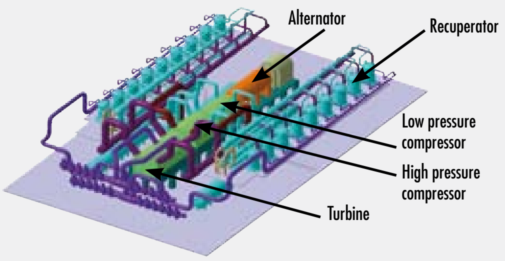
NEW SFR ENERGY CONVERSION SYSTEMS

The objective of eradicating the risk of a sodium-water reaction (one of the major issues in terms of availability and safety for previous generations of sodium-cooled fast reactors) led to the development of an alternative to the use of water (Rankine water-steam cycle) for the final conversion to electric power.
One of the options examined for Astrid includes a tertiary heat transfer circuit (downstream from the first two sodium cooling circuits) using nitrogen (Brayton cycle at a pressure of about 180 bars and an operating temperature of 300–500°C) based on design studies for high temperature reactors and gas turbines.
This is a major innovation compared with the industrial systems currently deployed throughout the world, and will require further R&D to optimize and qualify the technology. If its feasibility is confirmed, this conversion system gives priority to safety, at the cost of a reduction by a few percent in the overall thermodynamic efficiency, which still remains higher than for existing light water reactors.

MODULAR SODIUM-GAS HEAT EXCHANGER



INNOVATIVE CONVERSION ENERGY SYSTEM WITH NITROGEN CIRCUIT



4.

The Astrid integrated technology demonstrator ("Advanced Sodium Technological Reactor for Industrial Demonstration"), a 600 MWe prototype reactor, is the indispensable step before a possible industrial deployment. It is representative of the main necessary industrial characteristics and its demonstration capabilities are designed for the qualification of innovative concepts. The ongoing R&D will lead to a selection of particularly advanced options, especially in terms of safety and operability.

The industrial deployment of 4th-generation reactors requires prior qualification at a representative scale of the technological advances corresponding to the performance objectives assigned to the new generation of nuclear reactors, which could begin operating around 2040 (and perhaps earlier in some areas of the globe). The CEA proposes the Astrid integrated technology demonstrator, with a power rating of 1500 MWth (or about 600 MWe) making it representative of commercial reactors (particularly for the demonstration of safety and operating modes) while ensuring sufficient flexibility for its objectives (with the possibility of subsequent changes or deferred installation of certain highly innovative options).

Astrid is designed to allow experiments with innovative fuels and transmutation – at least at the scale of individual pins containing minor actinides, or possibly even complete assemblies.

The design studies have led to the adoption of an oxide core ($UPuO_2$, with a mean Pu content of about 25%) with pellets of unusual design having very interesting properties, especially in terms of safety (see the sidebar on the low void effect core).

The Astrid nuclear steam supply system is a cylindrical vessel with a core catcher, primary and secondary sodium circuits (the primary system is enclosed within the vessel), improved decay heat removal systems, and the possibility of energy conversion systems using nitrogen instead of steam (to eliminate any risk of sodium/water reaction). The latter option is a major technological breakthrough, for which studies are still in progress to confirm its feasibility.

The Astrid project also includes fuel management facilities for fabricating fuel, recycling uranium and plutonium, or preparing demonstrations of transmutation. No existing industrial facility is currently able to address the needs in these three areas, and the CEA is now examining several options:

- the design of a fabrication unit for $UPuO_2$ fuel assemblies: the core fabrication facility (AFC);
- the design of a special facility for Astrid fuel reprocessing (core treatment facility: ATC) or possibly the examination of conditions under which Astrid fuel could be treated in the facilities at La Hague;
- the design of facilities that may be necessary to perform transmutation experiments, depending on the scale envisaged (individual pins to full sub-assembly).

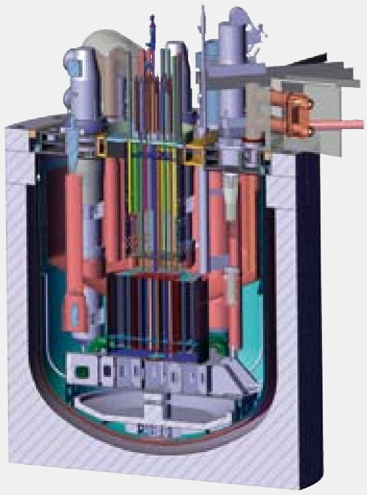
ASTRID INTEGRATED TECHNOLOGY DEMONSTRATOR

Astrid is designed to demonstrate at industrial scale the validity of the major innovations proposed for 4th-generation sodium-cooled fast reactors. Its characteristics must be suitable for extrapolation to the future high-power (typically 1500 MWe) commercial SFRs, especially with regard to safety and operability.

Its main features include the following:

- A thermal power rating of 1500 MW (corresponding to an electrical rating of about 600 MW) was adopted for Astrid. This is sufficient power to ensure the representativeness not only of the reactor itself in terms of operability and energy availability in normal operation, and with respect to safety studies including severe accident situations, but also for the main components. It also helps offset the operating cost by producing a significant amount of electricity.
- Its level of safety must be at least equivalent to that of the 3rd-generation power plants commissioned at the same time. Every effort will be made to ensure a more robust safety demonstration than for previous SFR designs. From the design stage, Astrid will take into account the lessons learned from the Fukushima accident.
- Astrid will provide fast neutron spectrum irradiation services, as Phenix did in the past. These irradiations will gradually improve the performance of the core, and allow testing of new fuels and structural materials such as carbide fuel and oxide dispersion strengthened (ODS) steel cladding tubes.
- In terms of availability, the continuous operation of Astrid may be affected by these experiments. Otherwise, the reactor availability should exceed 80%. The selected options should demonstrate that 90% availability is possible when extrapolated to commercial power plants.
- Although future fast neutron power plants may operate as breeders, Astrid will be an isogenerator, with breeding as a later option if necessary. Astrid will demonstrate the industrial feasibility of multiple recycling of plutonium and will conduct minor actinide transmutation experiments.

THE ASTRID REACTOR



ASTRID DESIGN OPTIONS

Main features

- 600 MWe reactor
- Pool-type architecture for nuclear island
- Intermediate sodium circuit
- Strategy for severe accidents (core catcher)
- Innovative oxide fuel core ($UPuO_2$)
- Transmutation capability

Open options

- Energy conversion system (gas system or conventional water/steam system)
- Provisions to eliminate severe accidents (3rd shutdown system)
- Provisions to mitigate core meltdown accidents
- Core catcher technology and location
- Diversified and autonomous decay heat removal systems
- New instrumentation and inspection technologies

5.

The Astrid project is conducted within a broad cooperative framework in France (in cooperation with EDF, AREVA, Alstom, Bouygues, Comex Nucléaire, Jacobs France, etc.) and internationally (cooperation with Toshiba, Rolls-Royce, and Astrium, as well as actions within the GIF or under bilateral programs). According to the current project timetable, the detailed conceptual design studies will be completed in 2017 at the earliest, before the decision to begin construction for a planned startup during the first half of the 2020s.

The Astrid project currently includes the following main phases:
- design work until 2017 leading to the Basic Design (APD);
- construction from 2018 to 2023–2025 (assuming the decision is made on completion of the Basic Design).

Funding of the studies prior to the decision to build (expected only at the end of 2017) is ensured under the program of investments for the future. France is determined to succeed and – with its international partners – to be one of the first countries with a comprehensive 4th-generation reactor design package. Commissioning of the Astrid integrated technology demonstrator in 2023–2025 will not only restore the experimental fast neutron flux irradiation capability, but also ensure the continued skills necessary for the project and prepare the local industrial fabric to be ready for industrial deployment after 2040.
More than 500 people are already involved in Astrid project, half of them employed by industrial partners in the project. The CEA is responsible for contracting and project management, with many French and foreign industrial firms participating in the project: AREVA for the nuclear steam supply system, instrumentation & control, and nuclear auxiliaries; EDF at several levels including R&D and contracting assistance, especially as related to reactor operability characteristics; Alstom for the energy conversion system; Bouygues for civil engineering and ventilation; Comex Nucléaire mainly for in-service inspection; Jacobs France for common resources and infrastructures; Toshiba for large electromagnetic pumps; Rolls-Royce for sodium-gas heat exchangers and fuel handling; Astrium for dependability.
The core fuel assembly fabrication facility (AFC) – which must be available before the startup of Astrid (it is estimated that about three years of production could necessary for the startup core) – is being designed jointly with AREVA. Powder metallurgy, already used for the MOX fuel supplying French light water reactors, will be implemented with the modifications necessary for the specific characteristics of FNR fuel. New fuel fabrication precursors may be provided with powder produced by the COEX™ process (a decisive step in fuel cycle research in recent years). The Basic Design should be ready in 2016 or 2017 and, if the decision to build is made at that point, the production of the first sub-assemblies could begin in the early 2020s.

CURRENT ASTRID PROJECT TIMETABLE

Between 2010 and 2012, the CEA and its industrial partners carried out the pre-conceptual design phase (AVP1) to provide the French government with the technical information concerning the Astrid project within the deadline stipulated by the act of June 28, 2006. This phase led to the preparation of a Safety Guidelines document specifying the project objectives and procedure in this fundamental area.

The phase of conceptual design (AVP2) will consolidate and finalize the design options selected for the Astrid reactor. This phase will be marked by a Safety Options Report, a regulatory milestone in the Nuclear Regulatory Authority (ASN) review of the validity of the choice of options in this area.

The Basic Design (APD) will then compile all the technical, organizational and cost-related items necessary before the decision to build Astrid. The first version of the interim safety analysis report will be issued at the end of this phase.

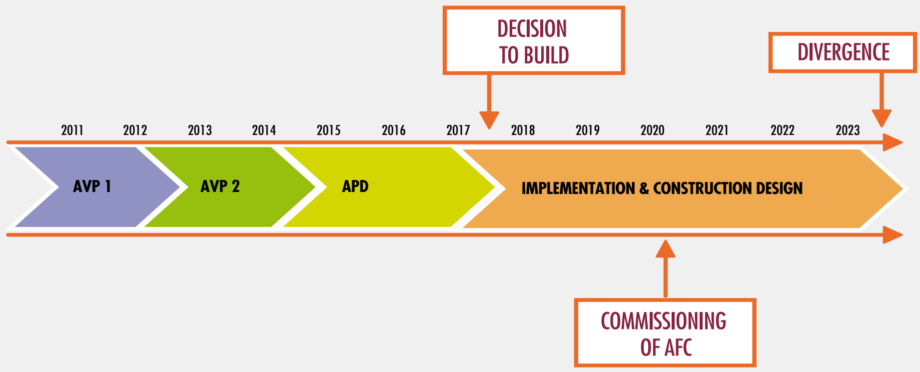
Finally, the actual implementation and construction design phase will be conducted with a view to reactor startup about six years after the end of the APD.

The Astrid timetable as of June 2012 is shown below.

The main milestones and the dates currently planned for the material cycle demonstrations are:

- Commissioning of the core assembly fabrication facility (AFC) around 2020.
- First demonstration of americium transmutation (in a test pin) a few years after Astrid diverges.
- First recycling of Astrid fuel in 2030–2035;
- If applicable, demonstration of americium transmutation (at the sub-assembly scale) after 2035.

ASTRID TIMETABLE (JUNE 2012)

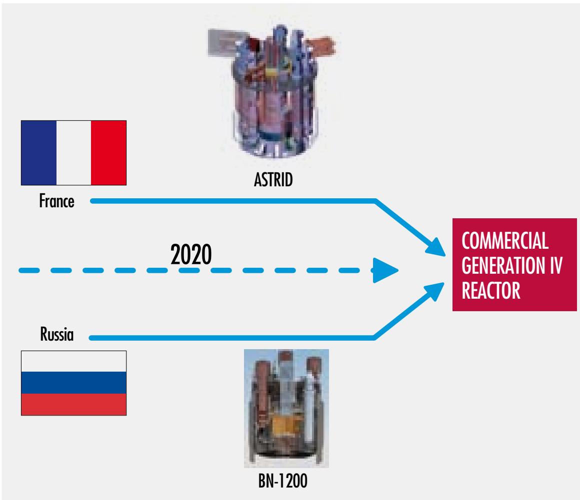


CEA-ROSATOM (RUSSIA) COOPERATION

The CEA and ROSATOM are currently examining an ambitious program to define a joint approach leading to the definition of a French-Russian 4th-generation sodium-cooled fast reactor. This project will be carried out within a consortium that could be set up by 2015.

Meanwhile, work is in progress to define the target market, the reactor technical specifications, and the R&D procedure leading to the desired innovations, particular with regard to safety and operation.

Both countries are also reviewing national projects: Astrid for France and BN-1200 for Russia. The possible convergence of these two projects is one of the issues examined in this review process.



6.

FNR deployment could begin sooner in some countries, where the security of energy supplies is of primary importance. Various options have been examined for the deployment of fast neutron reactors to supply nuclear electric power in France. The CEA considers that preference should be given to a phased approach with scenarios in which initially a limited number of FNRs would be deployed in synergy with the existing light water reactors (massive deployment would be considered only in a subsequent phase). Studies have also shown that it would be advantageous not to delay the initial deployment, which could begin around 2040. Industrial scenario studies will be conducted with EDF and AREVA to refine this approach.

The CEA has evaluated various possible options for adding fast neutron reactors to the existing nuclear generating capacity in a country such as France. If it is assumed that the technology will have reached industrial maturity around 2040 (based on over a decade of experience with the Astrid integrated technology demonstrator), then by that date the available quantities of plutonium would allow the deployment of a few fast neutron reactors.

The very gradual implementation of a few fast neutron reactors in a fleet consisting mainly of light water reactors is an interesting scenario in several respects:

- If at that point (following the operation of an advanced technology demonstrator such as Astrid) fast neutron reactor technology remains more complex to implement and at a higher unit cost than light water reactors, a very limited deployment of a few fast neutron reactors would provide increased operating experience feedback, ensure continued expertise, sustain the necessary industrial potential, and continue to enhance safety. This will be invaluable when the increasing rarity of uranium natural resources makes it necessary to deploy fast neutron reactors on a larger scale, based on the best industrial technology.
- Most importantly, the first fast neutron reactors will demonstrate feasibility of the closed fuel cycle, capable of using the plutonium contained in spent MOX fuel that is currently stored pending recycling. The commissioning of a 1500 MWe FNR every four years would absorb the plutonium produced by the light water reactors during the same period, stabilizing the stored MOX fuel inventory. This would be an important step toward complete control of the plutonium inventory.

Moreover, plutonium recycling is more effective when started early. Plutonium consists of a series of isotopes, and one of them – plutonium-241 – decays into americium with a half-life of 14 years. The longer plutonium recycling is delayed, the larger the amount of americium present in the waste. It is estimated that a 40-year delay in MOX recycling would generate in the waste the amount of americium produced in nearly ten years by the PWRs in operation today.

FNR DEPLOYMENT SCENARIOS IN THE FRENCH NUCLEAR POWER PROGRAM

Recycling the plutonium contained in spent fuel from light water reactors is limited by the degradation of its isotopic quality, which rapidly makes it unsuitable for reuse in PWRs. This is why spent MOX fuel is stored in the perspective of deferred reprocessing to recover the plutonium when new reactors capable of using it to the full become available.

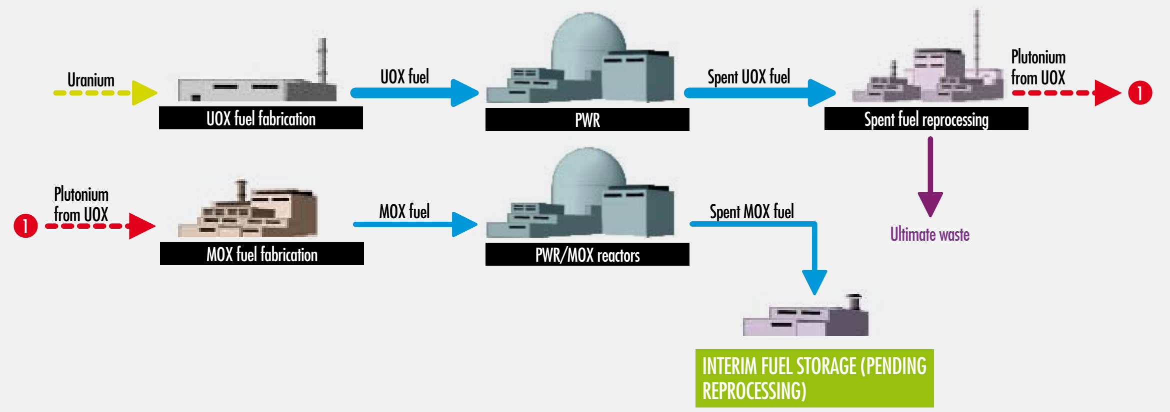
This strategy fully guarantees its safety, but is not the optimum solution. This is why the option preferred by the CEA is to deploy a limited number of fast neutron reactors as soon as it is technically and industrially feasible and economically acceptable. By recycling the plutonium contained in spent MOX fuel, this approach avoids the storage of spent fuel with a continually increasing mass of plutonium awaiting reuse.

A large-scale deployment of fast neutron reactors would be considered only at a later stage, in response to diminishing uranium resources and based on the experience gained in operation with the first reactors.

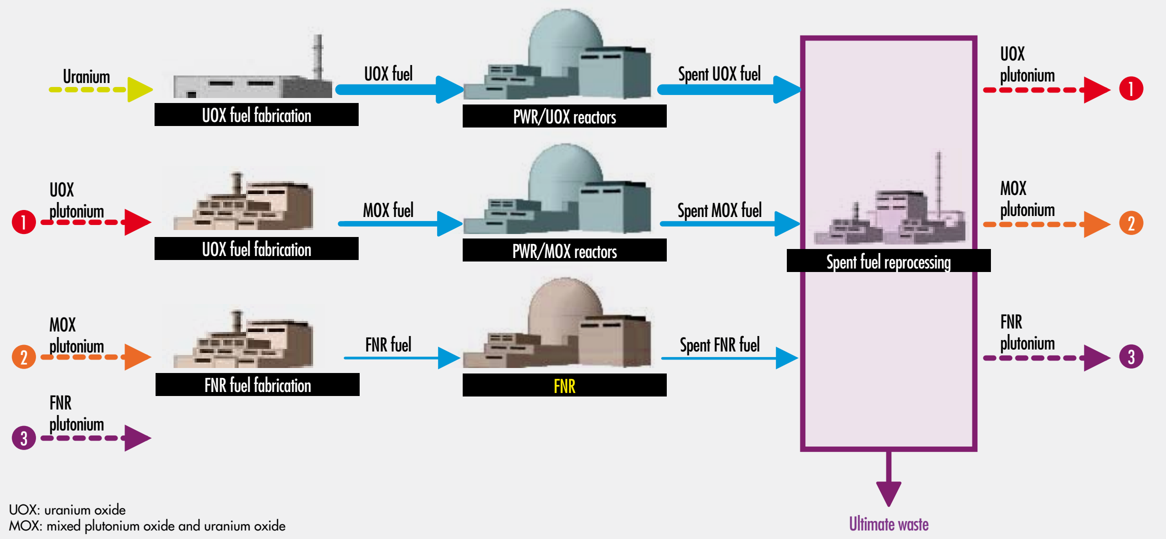
The scenarios analyzed by the CEA, EDF, and AREVA indicate that the deployment of a 1500 MWe FNR every four years in the context of the current French reactor population would stabilize the stored MOX fuel inventory. The idea of supplementing the existing PWRs by a limited number of fast neutron reactors is applicable irrespective of the installed nuclear power capacity.

Although no safety-related problems have been identified for the interim storage of MOX fuel, delaying the commissioning of the first FNRs beyond 2040 would result in a degradation of the isotopic quality of the plutonium contained in the spent fuel, which would be unfavorable in several respects (especially with regard to the quantities of americium generated by radioactive decay).

PLUTONIUM MANAGEMENT IN CURRENT FRENCH PWRs



PLUTONIUM MANAGEMENT IN A MIXED PWR/FNR POPULATION



UOX: uranium oxide
MOX: mixed plutonium oxide and uranium oxide

6.

MATERIALS AVAILABLE FOR FNRs

(Source: French national inventory, Andra, June 2012)

- The French stockpile of depleted uranium in 2040 can be estimated at 450 000 metric tons (increasing by more than 7000 metric tons a year). This inventory can be compared with the estimated needs of a fast neutron reactor capacity of 60 GWe (after the startup period) corresponding to about 50 metric tons a year, ensuring security of supplies over the long term.
- Fast neutron reactors require a startup quantity of fissile material, such as the plutonium contained in spent fuel. Enriched uranium oxide (UOX) fuel, when unloaded at the end of its reactor residence time, comprises about 1% plutonium. After treatment, plutonium is recycled in MOX fuel containing up to 10% plutonium. Unrecycled spent MOX fuel still contains nearly 6% residual plutonium (about 7 metric tons of plutonium in MOX fuel is placed in interim storage each year). An estimated 3800 metric tons of spent MOX fuel will be stockpiled by 2030. The amount of plutonium that could be recovered at that time by treating spent MOX fuel combined with a fraction of UOX fuel would be near 300 metric tons.

A 1500 MWe fast neutron reactor core contains about 12 metric tons of plutonium (of isotopic quality equivalent to the plutonium in spent MOX fuel). In addition to this "core inventory", a "cycle inventory" of at least an equivalent quantity of plutonium is necessary for refueling and for in-process fuel cycle operations. The total order of magnitude is about 25 metric tons of plutonium per reactor. If recycled, the quantities contained in spent MOX fuel produced by the existing French reactors would be sufficient for the startup of up to twelve 1500 MWe fast neutron reactors.

7.

Fast neutron reactors also allow considerable flexibility in the management of nuclear materials: depending on the need, they could be used to deploy additional generating capacity without resorting to external natural uranium resources, or they could subsequently reduce the nuclear material inventory in "burner" mode if fast neutron reactors were one day phased out.

SFR technology provides essential flexibility with the possibility of adapting the management of nuclear material inventories to future strategy changes. The Astrid integrated technology demonstrator is therefore designed to be an isogenerator (it produces as much fissile material as it consumes). This means that following the reactor startup with external plutonium (mainly from reprocessed spent MOX fuel) it can operate with only a supply of natural or depleted uranium, without any further need for external fissile material. If a need arose for additional installed capacity, the reactor could be switched to "breeder" mode to produce the additional fissile material necessary for the startup of new reactors.
Conversely, in the event of a decision to shut down the operating fast neutron reactors, they could be placed in "burner" mode; this would ensure a substantial reduction in the residual plutonium inventory (by about two-thirds in 60 years, while maintaining electricity production until the final shutdown). Studies are still in progress to optimize this operating mode.

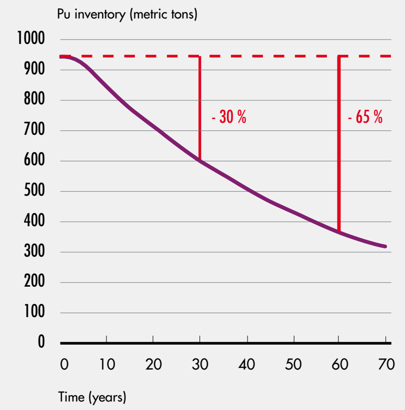
SHUTDOWN SCENARIOS

Fast neutron reactors offer flexibility for the management of nuclear materials, and in particular for plutonium. Based on the same concept, the reactor can be operated in three modes:

- In isogenerator mode it produces as much fissile material (plutonium) as it consumes, with only a supply of depleted uranium. This is the basic operating mode selected for the Astrid reactor project.
- In breeder mode it produces more fissile material than it consumes, periodically allowing the startup of new reactors. This is the mode envisaged in countries (such as India) that wish to rapidly increase their nuclear electric power generating capacity. Breeding is generally obtained by placing "fertile blankets" of depleted uranium around the core periphery.
- In burner mode it consumes more fissile material than it produces. This mode could be useful for diminishing the quantities plutonium at the end of the lifetime of a series of reactors, for example, to avoid interim or definitive storage of disused sensitive nuclear materials.

This property of fast neutron reactors makes them adaptable to different policy decisions that could be in effect at different points in time; in this respect the deployment of fast neutron reactors ensures flexibility for future policy orientations.

PLUTONIUM INVENTORY REDUCTION BY A 60 GWe REACTOR FLEET



The CEA has carried out exploratory studies on the capacity of fast neutron reactors to reduce the actinide inventory during the FNR phase-out period by modifying the fuel elements to convert the isogenerator reactors to burners. It appears that the in-process plutonium could be reduced by two-thirds in 60 years operation (while maintaining electricity production). At this point this is only a theoretical result and does not take into account the physical and industrial constraints that remain to be evaluated.

8.

Transmutation of the minor actinides will not eliminate the need for a deep geological repository, but could open the way to longer-term progress. The dimensions of a long-lived high-level waste repository could be reduced by a factor of 10 and, after the first few centuries, the radiotoxic inventory of the waste could be diminished by up to factor of 100.

Studies by Andra show that the actinides have very low mobility in a clay repository site and will not contribute to the radiological impact of the repository. Nevertheless, the presence of minor actinides in the waste is the main source of long-term radiotoxicity (after three centuries, nearly 99% of the residual radiotoxicity of the vitrified wasteforms currently produced will be due to the presence of americium, curium, and their decay products). Removing them from the waste and transmuting them into shorter-lived elements is a challenge that is essential to further progress, not only in France but also in several European countries as well as in the United States and Japan, and has been a focus of research for more than 20 years.

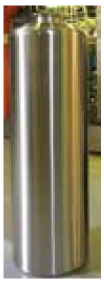
The minor actinides are also the main contributors to the heat released from vitrified waste packages, which to a large extent determines the design of repository disposal cells: the lower the heat load in the waste, the higher the disposal density of the repository. Removing the minor actinides from the final waste and providing a prior interim storage period of about a hundred years (to allow radioactive decay of the short-lived fission products) could thus significantly reduce the size of the repository. A study conducted jointly with Andra (with financial support from EDF) estimated a gain by an order of magnitude on the footprint of the disposal cells for long-lived high-level waste in the clay disposal concept currently envisaged by Andra. This would represent an overall footprint reduction for the entire repository, including intermediate-level waste storage and access infrastructures, by about a factor of 3.

The minor actinides do not all contribute equally to the disadvantages mentioned above. The first target for a transmutation strategy could be americium, the element whose transmutation would be of the greatest benefit to waste management, and which has the most limited impact on recycling operations.

Transmutation can only be considered in the context of a future installed fast neutron reactor capacity and suitable fuel cycle facilities. This strategy is therefore not applicable to the waste already produced or committed in existing nuclear power stations.

This report provides information for decision-makers to evaluate and choose among the advantages and drawbacks of different transmutation options in view of technical, radiological, economic or other criteria.

CIGEO: Industrial geological repository for high-level waste and long-lived intermediate-level waste.



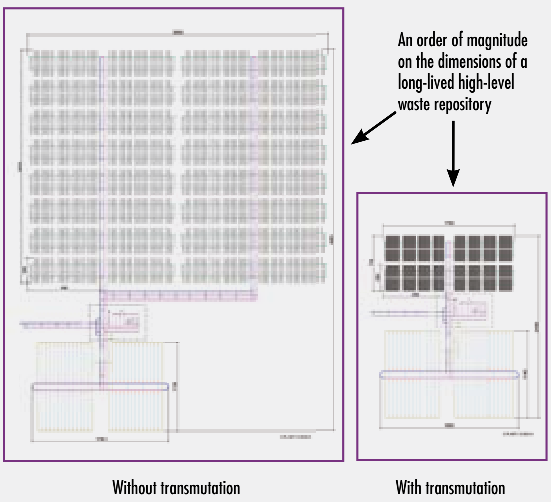
Vitrified waste canister (type CSD-V).

CONTRIBUTION OF TRANSMUTATION TO WASTE MANAGEMENT

Studies by the CEA and Andra have clarified the main advantages of a transmutation strategy for final nuclear waste management.

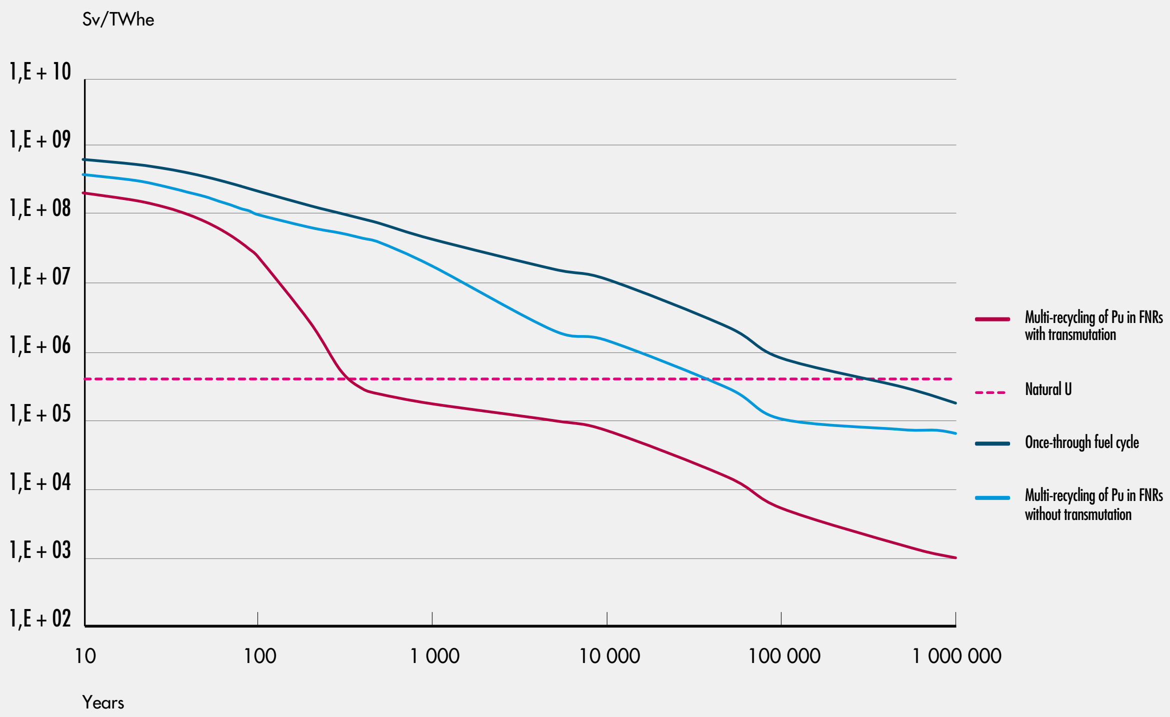
1. First is a reduction in the secular heat load of long-lived high-level waste, for which the main contributor is an isotope of americium (^{241}Am). For a repository concept similar to the one currently investigated by Andra in a clay formation, the transmutation of americium could diminish the high-level waste repository footprint by up to a factor of 10, assuming prior interim storage of the vitrified waste packages for 120 years (to allow decay of the fission products, the main contributors to heat release in the early decades).
2. The transmutation of both americium and curium would diminish the long-term radiotoxicity (the potential harmfulness arising from ingestion of the radioactivity present in long-lived high-level waste) by a factor of 100 after a few centuries. In less than 500 years, the waste radiotoxic inventory would fall to a level equivalent to that of all the uranium mined for fuel fabrication today.

REPOSITORY DIMENSIONS WITH OR WITHOUT TRANSMUTATION OF THE MINOR ACTINIDES



Andra-CEA study: 2012

RADIOTOXICITY OF ULTIMATE WASTE WITH OR WITHOUT TRANSMUTATION OF THE MINOR ACTINIDES



9.

The feasibility of minor actinide separation has been demonstrated in the laboratory for all the options under consideration today. There are no theoretical obstacles to extrapolating these processes to commercial scale: R&D could be pursued to optimize and consolidate these concepts.

At laboratory scale on actual spent fuel samples (several kg), the CEA has tested the processes it has developed to recover minor actinides using new extractants that are both selective and resistant. Several options were considered, corresponding to various possible recycling routes (group separation of all the actinides for homogeneous recycling, sequential separation for minor actinide recycling in core blankets, separation of americium alone, etc.). The observed performance was excellent (recovery yields above 99%) and additional tests were carried out to approach the operating conditions at industrial scale: the results are very encouraging (especially with regard to the endurance of the molecules involved) and are very promising for possible commercial application of the concepts investigated.
Studies are still in progress today to better define the conditions of commercial-scale implementation (testing of individual technologies at pilot scale, integration tests, detailed specification of process control modes, etc.), as well as to further optimize the concepts.

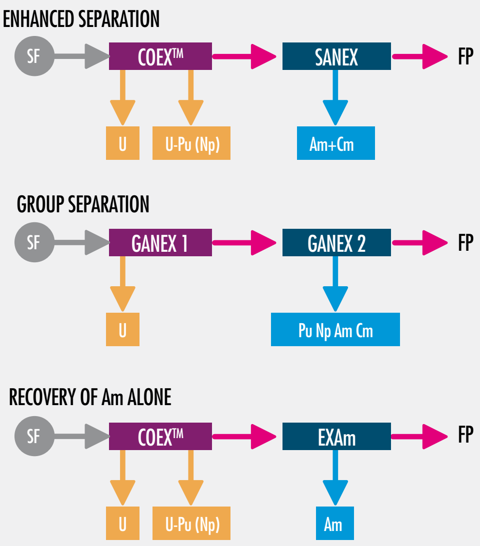
PROCESSES STUDIED BY THE CEA (SANEX, GANEX, EXAm)

CEA research has substantiated processes of partitioning americium and curium for each of the transmutation options under consideration:

- The SANEX (Selective minor ActiNide EXtraction) process seeks to recover americium and curium after the "conventional" reprocessing steps (i.e. after the recovery of uranium and plutonium).
- The EXAm (EXtraction of Americium) process seeks to recover americium alone after conventional reprocessing.
- The GANEX (Group ActiNide EXtraction) process seeks to recover plutonium together with all the minor actinides.

These processes have been tested experimentally on actual spent fuel samples at the CEA's Atalante hot laboratory, in a laboratory-scale model of a possible commercial facility designed to use these processes. The tests showed that these concepts effectively permit the recovery of the elements of interest with high separation factors. At this stage no potentially insurmountable problem has been identified, although many issues must still be investigated on the road to a possible commercial-scale implementation.

SEPARATION PROCESSES



SF: spent fuel
Am: americium
Cm: curium
U: uranium
Pu: plutonium
Np: neptunium
FP: fission products

Minor actinide separation test in the hot laboratory Atalante.

10.

The feasibility of transmutation of americium has been demonstrated at the scale of a few pellets in homogeneous mode, in the core of fast neutron reactors. The first analytical irradiation experiments are now in progress for the heterogeneous transmutation option in the core periphery.

The results submitted in 2005 by the CEA (reporting on the research carried out under the 1991 Radioactive Waste Management Act) established that the minor actinides could be efficiently transmuted only in a fast neutron spectrum (whether in power reactors or in dedicated devices). This was confirmed by analyzing the results of irradiation experiments in the Phenix reactor in homogeneous mode (Superfact experiment) and heterogeneous mode (irradiation of pin segments containing americium-bearing pellets on various media).

The CEA studies focus on transmutation concepts in critical fast neutron reactors, involving multi-recycling of americium (to eliminate about 99% of the residual americium) using uranium oxide fuel (which can be reprocessed in the same facilities as the driver fuel). Two concepts are still under consideration: recycling of material diluted in the reactor fuel (the “homogeneous” concept, resulting in an equilibrium americium content of about 1% in the fuel), and the “americium-bearing UO_2 blanket” concept of recycling in the core periphery (with one ring of blankets containing 10% americium at equilibrium). The second concept has the advantage of limiting the number of actinide-bearing objects, and does not otherwise affect reactor core operation.

The recent experiments have mainly concerned recycling in blankets, which had not previously been tested. Since the shutdown of the Phenix reactor, only small-scale experiments (on a few grams of americium) have been carried out in irradiation reactors – the Petten high-flux reactor (HFR) under a European project, and the Osiris reactor at Saclay – while attempting to reproduce the conditions in the periphery of a fast neutron reactor core (the temperature in particular). The purpose of these experiments is primarily to assess the behavior of the gases generated within oxide pellets by the transmutation of americium in order to define detailed specifications for their fabrication parameters, rather than to confirm the reality of americium transmutation, which has already been demonstrated. The first results will be available during 2013.

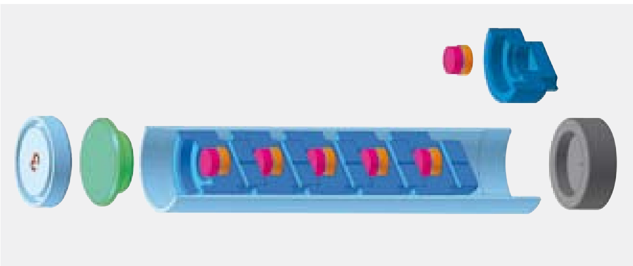


Diagram of the experimental irradiation device for transmutation in americium-bearing blankets.

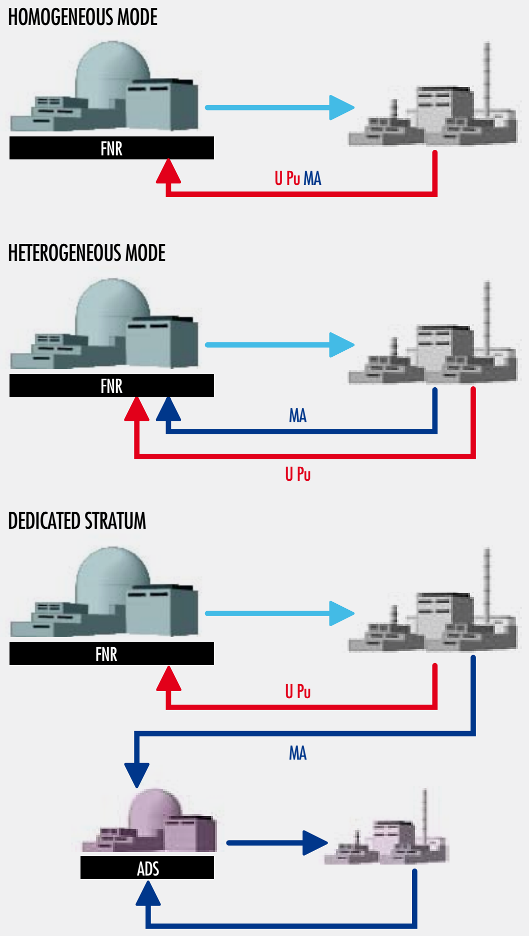
Phenix reactor at Marcoule.

DIFFERENT TRANSMUTATION ROUTES FOR THE MINOR ACTINIDES

Transmutation of the minor actinides consists in recycling them in a fast neutron flux, causing them to fission (while taking advantage of the energy resulting from their fission – even if very modest – to produce electricity). Several approaches can be considered:

- Homogeneous transmutation: the minor actinides are recycled by "diluting" them in the fuel of nuclear power fast reactors (up to a concentration of a few percent).
- Heterogeneous transmutation: the minor actinides are recycled in power reactors at higher concentrations in a limited number of dedicated fuel elements. A particularly interesting option appears to be recycling in minor actinide-bearing uranium "blankets" (containing about 10% minor actinides) around the periphery of the core.
- Transmutation by "dedicated systems" in a "dedicated stratum" of the cycle. This is notably transmutation in accelerator-driven systems (ADS) in which the minor actinides are managed separately from the uranium and plutonium management cycle.

MINOR ACTINIDE TRANSMUTATION ROUTES



U: uranium
Pu: plutonium
MA: minor actinides

Transmutation can also be performed in a separate stratum specifically dedicated to this function. It would include accelerator-driven systems (ADS) designed to accept high concentrations of minor actinides. The R&D effort necessary to bring these systems to industrial maturity appears much greater – despite the progress to date – than for critical systems. Moreover, transmutation by ADS (in the case of the devices being studied today) would raise the cost per kWh by an estimated 20% compared with the use of critical reactors.

Accelerator-driven systems (ADS) are being investigated in France by the CNRS in conjunction with the Myrrha project (Multipurpose hYbrid Research Reactor for High-tech Applications) proposed by SCK•CEN (Belgium). An ADS is a complex system requiring the development of highly technical components for which no significant large-scale experience is available (accelerator, spallation target, fuel, etc.) and which have never been previously integrated. The theoretical transmutation performance of these systems is rather promising (some 20 times higher than that of critical reactors per unit of energy produced), but they cannot be efficiently implemented because of the low power level currently envisaged for these systems.

The technical and economic assessment carried out by the CEA is based on the industrial models of the EUROTRANS European research program, which has a maximum thermal power rating of about 400 MWth. The study showed that 18 ADS would be necessary to obtain a transmutation capability equivalent a fast neutron reactor generating capacity of 60 GWe (a fleet of forty 1500 MWe reactors). Fast neutron reactors are necessary to ensure multiple recycling of plutonium, which cannot be efficiently managed by ADS designs at this time. The estimated supplemental cost (on the levelized average cost per kWh) of this option is about 20% compared with transmutation in critical reactors.

THE MYRRHA PROJECT

Accelerator-driven systems consist of a subcritical core where a chain reaction is possible only when additional neutrons are supplied by an external source. External neutrons are produced by high-energy protons interacting with some nuclides, through what is known as a spallation reaction. In addition to the subcritical reactor itself, an ADS includes a proton accelerator coupled with a spallation target.

The subcritical core can be loaded with a large percentage of minor actinides, which is favorable for transmutation, while controlling the chain reaction. Nevertheless, the development of these systems requires the design of innovative and highly technical equipment:

- an accelerator, supplying an intense beam of high-energy protons with very high reliability;
- a target, which reacts to the beam by generating high-energy neutrons (corrosion problems and material behavior are now being investigated);
- a subcritical reactor core;
- fuel without uranium but with high concentrations of plutonium and minor actinides.

Guinevere is a very low-power ADS demonstrator model built in 2007 at Mol, Belgium, by SCK•CEN and the CNRS. Commissioned in 2012, this facility is a replica of an ADS coupled to a lead-cooled fast reactor for research purposes and feasibility studies for a larger ADS. Belgium proposes the Myrrha (Multipurpose hYbrid Research Reactor for High-tech Applications) project with a 100 MWth ADS for 2025.

The Myrrha project (SCK•CEN, Mol, Belgium).

12.

The consequences of a commercial-scale implementation of transmutation in fast neutron reactors (in the different possible modes) were examined according to a variety of criteria. The findings indicate significant gains in the properties of the ultimate wasteforms, but also disadvantages, especially related to the operation of the material cycle. The impact of transmutation on the cost of electricity produced by fast neutron reactors would be about 5% to 10% (the cost per kWh is determined to a very large extent by the reactor cost, but only slightly affected by the implementation of transmutation options). A sustained R&D effort will be necessary to assess the various aspects of commercial implementation, and in particular the fabrication and handling of objects containing minor actinides.

A comprehensive study was carried out by the CEA with EDF and AREVA, reviewing all the criteria relevant to commercial implementation, in order to assess various nuclear fuel cycle options, without transmutation or with transmutation of minor actinides in the different modes proposed. The objective at this stage is to assess the options and not to describe a future industrial reality. Interesting results were highlighted by using a more mature methodology than in previous evaluations. The following main points were identified:

- Transmutation of the minor actinides avoids the presence of these elements in nuclear waste. By transmuting all the minor actinides, the long-term radiotoxicity of the ultimate waste (after a few centuries) can be reduced by a factor of between 20 and 100, depending on the time frame considered. Transmuting americium alone would limit this reduction to a factor of 2. In addition, the overall dimensions of a high-level waste repository footprint can be reduced by a factor of up to 10.
- Transmutation of the minor actinides increases their inventory in reactors and fuel cycle facilities. For a generating capacity of 60 GWe, this inventory increases from about twenty metric tons (without transmutation) to between 60 and 160 metric tons depending on the transmutation scenario (compared with a plutonium inventory of nearly 1000 metric tons).
- Homogeneous recycling may affect some core safety parameters; recycling minor actinides in blankets leads to difficulties in fabrication and handling of assemblies, before and after irradiation in the reactor.
- The presence of curium in transmutation assemblies generates constraints (irradiation, heat flow) an order of magnitude greater than for recycling americium, resulting in major operating difficulties.
- The dosimetry impact of transmutation (collective dose to workers) is difficult to assess accurately (protective measures will be adapted to the increased source term, but with a definite impact on costs).
- The supplemental cost per kWh (levelized cost) of transmutation in critical reactors is about 5% to 10% for the various options (homogeneous or blankets, minor actinides or americium alone).
- The incidence of transmutation on the level of industrial risks (for example the consequences of failure) is difficult to assess accurately at this time, although homogeneous recycling appears to have a higher level of risk because it involves the fuel of all the operating reactors.

Joint work with EDF and AREVA will continue, notably in order to assess a possible gradual implementation of more advanced options (fast neutron reactors used for multi-recycling of plutonium initially, then americium and possibly all the actinides at a later time).

ECONOMICS OF THE NUCLEAR FUEL CYCLE

The CEA has evaluated the economic aspects of the fuel cycle options, seeking to assess the impact of various strategies (notably transmutation) on the "levelized cost of electricity" (LCOE).

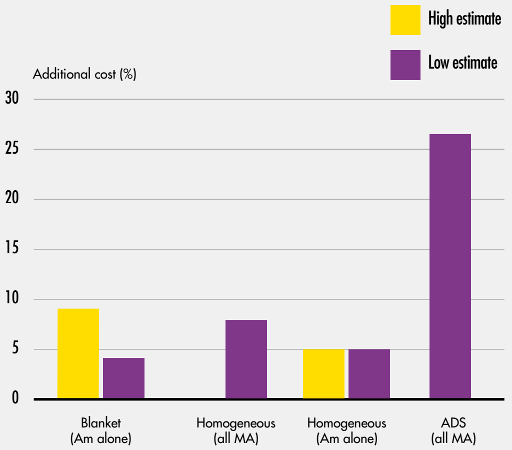
Detailed assessments were carried out in cooperation with EDF and AREVA to assess as accurately as possible the costs of implementing these options, from mining to ultimate waste management.

The cost benchmark used for this analysis was the cost per kWh of electricity for an installed fast neutron reactor capacity of 60 GWe with multi-recycling of uranium and plutonium.
The fuel cycle options involving transmutation of the minor actinides represent a levelized cost increase ranging from less than 5% (low estimate for transmutation of americium-bearing blankets) to 25% (transmutation by ADS in a dedicated stratum). In addition:

- the estimated additional fuel cycle and reactor cost is not offset by the estimated reduction in waste storage costs (due to the effects of economic discounting, which diminishes the weight of long-term expenditures);
- the predominant weight of reactor costs (capital investment and operation) in the total cost makes the analysis result very sensitive to any impact on this aspect (especially the capital cost and load factor).

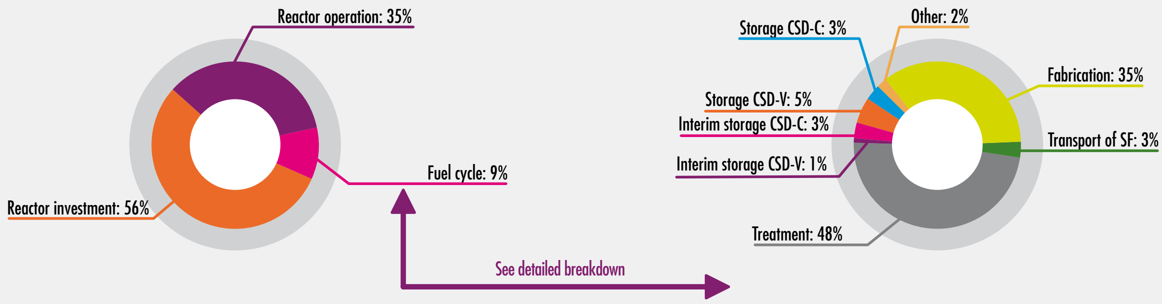
Although some cost items remain uncertain (for the fuel cycle in particular), their impact is limited and this should not modify the orders of magnitude indicated above.

ESTIMATED SUPPLEMENTAL COSTS PER kWh (%)



LEVELIZED COST BREAKDOWN PER kWh FOR AN INSTALLED FAST NEUTRON REACTOR CAPACITY OF 60 GWE (WITHOUT TRANSMUTATION)

Discount rate: 4% for the 30 first years, 2% afterward.



SF: spent fuel
CSD-V: standard vitrified waste package
CSD-C: standard compacted waste package

Design and production: **avant**garde

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