Closing the cycle with fourth-generation reactors

Aside from achieving optimum utilization of fissile materials, fourth-generation nuclear systems will have to ensure the reprocessing of their own waste, and afford effective defenses against proliferation. Researchers are working on a whole range of processes, to cater for the options that must yet be taken up, with respect to reactors, fuels, and reprocessing–recycling goals. They are exploring, at the same time, hydrometallurgical paths, allowing as of now fuel cycle closure for second- and third-generation reactors, and the pyrochemical path, which would allow in situ reprocessing of materials.

P. Stroppa/CEA.

Remote handling of an irradiated fuel pin in the reception cell of the Shielded Process Line (CBP: Chaîne blindée procédé) in the ATALANTE complex, at CEA's Marcoule Center, where, since 2004, a whole range of experiments have been carried out, to complete the technical feasibility demonstration of advanced partitioning. The facility processed 15 kg of irradiated fuel, to demonstrate, successively, uranium and plutonium separation, then the separation of lanthanides and minor actinides, and finally that of americium and curium, with recovery rates in excess of 99.9%.

If they are to provide a sustainable source for energy supply, **nuclear systems** of the future needs must meet a number of criteria: remain economically competitive, ensure the availability of resources over the long term, exhibit further enhanced safety, guarantee minimal environmental impact, and control **proliferation** risks. The challenge of sustainability is intimately bound up with the options selected for the **fuel cycle**, and materials management. One leading idea has gained acceptance, in order to ensure better value-added utilization of **uranium** resources, bring down long-term waste **radiotoxicity**, denature and destroy materials that might be misappropriated for purposes other than energy production: the recycling of **actinides** in reactors having the capability to turn them to best advantage, to wit **fast reactors** (**FRs**).

Such a strategy, initially implemented with the **reprocessing** and reuse of **plutonium** in **MOX** fuels, and further pursued with the investigation of more advanced options, in France, under the aegis of the **Act of 30 December 1991** (advanced **partitioning**, and **transmutation** of **minor actinides** [**MAs**]), was further reaffirmed with the French **Act of 28 June 2006**: **recycling** is seen to be an indispensable component of nuclear energy development.

Granting that principle, many options may yet be contemplated: Which actinides should be recycled (uranium, plutonium, all actinides)? How should they be recycled (through group management of all actinides, according to a **homogeneous** recycling concept, or separately, using dedicated substrates, or even dedicated strata, or machines)? In which type of fast reactor? And in what kind of fuel?

Such is the whole purpose of the investigations that are to be conducted over the coming years, with the aim of evaluating, against the various criteria mentio-

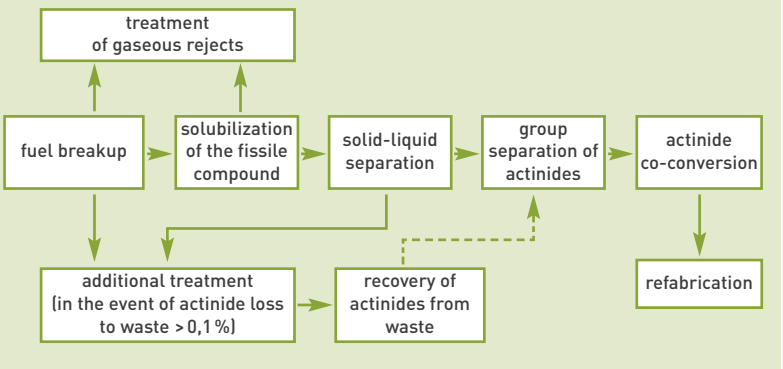


Figure 1.
General scheme for the reprocessing of fourth-generation waste.

ned above, the main paths that may be considered. The Act of 2006 has set new waypoints for research: 2012, for the selection of the best options, 2020 for the commissioning of a **fourth-generation** prototype, entailing that fuel cycle installations are available upstream, to keep it supplied.

About 30 treatment processes

The aim of recycling, at the lowest cost feasible, all of the actinides contained in spent fuel calls for the development of effective treatment processes, exhibiting high recovery efficiencies (> 99.9%), and compatible with ultimate waste conditioning processes, and fuel refabrication processes.

The choice of treatment process is heavily dependent on the choice of fuel, which remains open. However, on the basis of the experience gained with current processes, and of the fuel concepts being considered, a general scheme has been drawn up (see Figure 1). About 30 processes have been identified, for the various operations that must be carried out.

Pending choice of a reference fuel, investigations are mainly addressing the crucial steps involved in reprocessing, in particular pulsed-current breakup of fuels, group separation of actinides by liquid–liquid extraction, along with co-conversion of actinides by oxalic **precipitation**, or by the **sol–gel** path, while equally looking into the potentials afforded by innovative actinide recycling processes, such as **pyrochemical** processes. The various processes surveyed in the following paragraphs illustrate some of the directions pursued by the investigations being carried out in the ATALANTE facility,(1) at CEA's Marcoule Center.

Pulsed-current breakup of fuel elements

The pulsed-current process for the breakup of fuel elements is based on the fact that solid bodies, subjected to energies of several **kilojoules**, yielded by high-voltage pulses – around 200–500 kV – and discharge currents of 10–20 kA, locally undergo, over a few microseconds, energy density transfers of some 10–100 J/cm. Such an energy input causes local increases in temperature that may reach 10,000 **K**, and pressures of around 10^{10} **Pa**, instantly fragmenting the solid, which is reduced to the state of more or less fine debris.

Initial tests, involving the breakup by such means of **GFR** fuel constituent materials (a silicon carbide [SiC] plate, featuring honeycomb cells filled with SiC, simulating the **fissile** compound), have yielded encouraging results (see Figure 2). These allow the application of this process to be considered, to access the fissile compound in plate-type fuels, without going through dissolution of the fissile compound–inert matrix complex. It is planned to pursue this investigation, with more representative objects: closed plate fuel, and coated particles dispersed in an inert matrix.

Group extraction of all actinides

The dissolution of actinide oxides and carbides in nitric media being well understood (operational feedback from the 1960s and 1970s), and liquid–liquid extraction technology proving well suited to the nuclear energy domain, the group actinide extraction, or GANEX (for "Group An EXtraction"), concept was devised by CEA on this principle. The process (see Figure 3) involves a step of preliminary uranium separation, followed by the group extraction of **transuranic elements**, along with the residual uranium, from the raffinate. For this separation, two paths are being investigated: a modification of the existing process, developed for the advanced partitioning of minor actinides; and the search for new extractant(2) **molecules**.

The DIAMEX–SANEX process,(3) developed for the separation of americium (Am) and curium (Cm) from a solution of raffinate, yielded by an initial extraction cycle, may be adapted for the group extraction of actinides, uranium (U), plutonium (Pu), and neptunium (Np) included. Laboratory investigations are currently ongoing, to acquire data as to the way the U, Pu, and Np actinides are shared between the nitric acid, and the diamide extractant, used alone, or mixed with dialkylphosphoric acid. Partial preseparation of uranium at the head end of the process, feasible by a scheme involving monoamide extractants, makes it possible to adjust, subsequently, uranium flows to the desired values, during the following co-conversion, or fabrication steps. The scientific demonstration of this two-cycle process is scheduled for 2007–8 in ATALANTE, on an actual solution.

The search for higher-performance extractant molecules (in terms of selectivity, extraction capability, robustness), and the synthesis of molecules that may carry, or act as substrates for, a variety of coordination sites(4) (polytopic extractants), for the coextraction of actinides brought to varying degrees of oxidation [e.g. An(IV)–An(III), and An(VI)–An(III) mixtures] have been embarked upon, on the basis of the development

Figure 2.
Principle schematic of breakup by pulsed currents yielded by a Marx generator, with a voltage in the 200–500 kV range, and intensity of 10–20 kA (a), and view of an SiC honeycomb structure with cells filled with SiC (dispersed fuel), before pulsed-current treatment (b), and after 30 pulses (c).

(1) ATALANTE: a complex of laboratories, at CEA's Marcoule Center, featuring shielded cell lines, dedicated to investigations on spent fuel reprocessing, and high-level waste conditioning.

(2) Extractant: a molecule exhibiting characteristics endowing it with the ability to extract a given substance from a solution. A polytopic extractant features a number of sites able to bond with that substance.

(3) DIAMEX (DIAMide EXtraction): a chemical process for the partitioning of fission products from the ensemble formed by lanthanides and minor actinides. SANEX (Selective Actinide [An] EXtraction, or Séparation des actinides par extraction): a process for the separation of lanthanides from actinides.

(4) Coordination site: a site for the bonds set up by an ion with electron-donor atoms. In hydrometallurgy, a ligand is an ionized atom, or molecule present in an aqueous solution, liable to bind with a metal ion through coordination bonds.

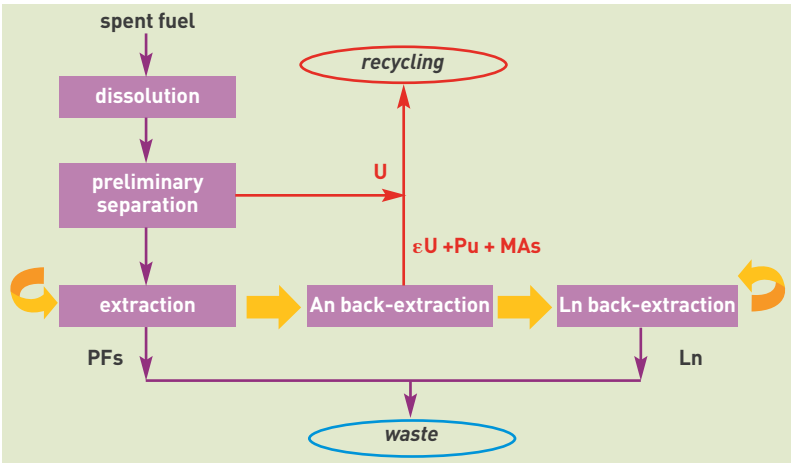


Figure 3.
Principle schematic of the GANEX process, investigated by CEA for the group extraction of actinides.

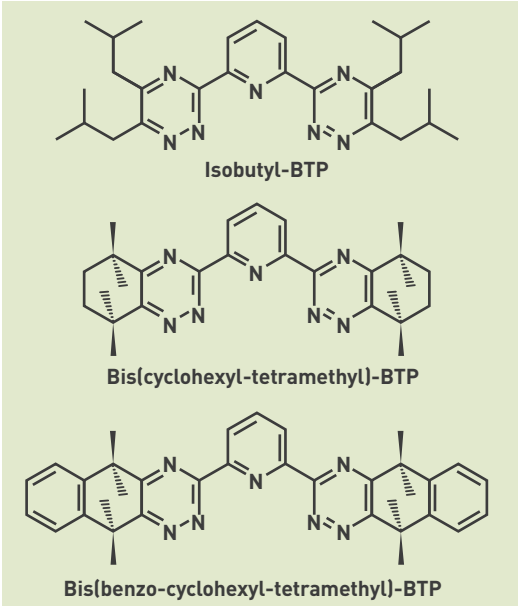


Figure 4.
Architecture of molecules from the bis-triazinylpyridine family, tested for single-cycle actinide separation.

of molecules from the bis-triazinylpyridine (BTP) family. These molecules were tested in the context of the technical demonstration of advanced partitioning processes, for the separation of americium and curium from high-activity effluents yielded by the **Purex** cycle, showing the ability to extract actinides from a high-acidity effluent, with an outstanding separation capability with respect to **lanthanides**. As tests with the initial molecules had shown high sensitivity to radiolysis (i.e. decomposition under the effects of radiation), and hydrolysis (chemical decomposition by water), new, more resistant molecules were synthesized: bis-cyclotetramethyl-BTP, and bis-(benzo-cyclotetramethyl)-BTP, by inserting triazine patterns in alpha position (see Figure 4). With these molecules, actinide separation and extraction performance remains excellent, while radiolysis resistance is markedly enhanced. On the other hand, a degradation has been found to occur with regard to extraction and back-extraction kinetics; investigations are ongoing to remedy this.

Actinide co-conversion

Actinide co-conversion is a key step, between fuel reprocessing, and refabrication. Two processes are being developed, to produce a solid actinide compound, at the head of the fabrication line, having practically the right, desired preform: these are the sol–gel process, using internal gelation, to form spherical actinide kernels (coated particle concept); and oxalic coprecipitation, to form a solid solution of actinides (pellet, or cylindrical pellet concept).

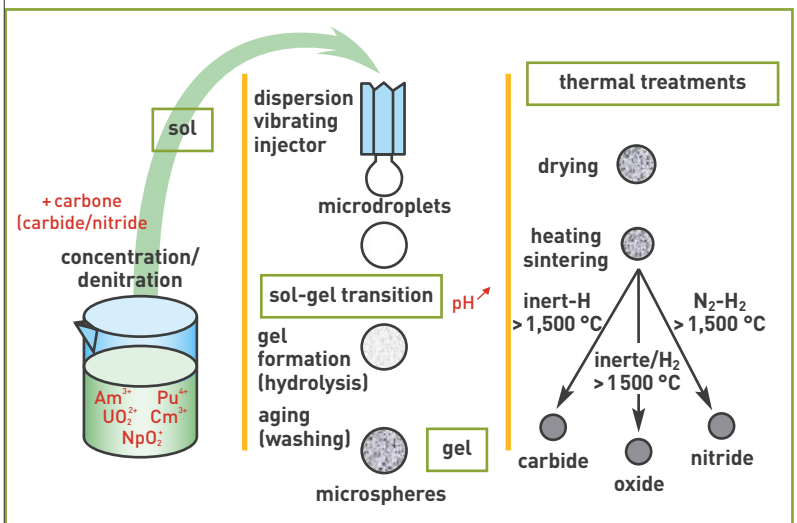


Figure 5.
Principle schematic of the internal gelation process for actinide co-conversion.

The sol–gel path

The internal gelation process (see Figure 5) developed involves hydrolysis of the actinides, this being initiated within the "sol" droplet, by thermal decomposition of an ammonia precursor: hexamethylenetetramine, which is dissolved into the sol. Thus, on the one hand, co-condensation of the actinides, even when exhibiting varying degrees of oxidation, is promoted, and, on the other hand, the aging of gel spheres may be achieved in more dilute ammonium hydroxide baths.

This process, equally applicable to the stable oxidation states[5] of actinides, and to their reduced states, was developed on the basis of trials in the presence of Ce(III) and U(IV), then tested in the context of U–Pu co-conversion, to an 85/15 ratio, for two modes: gelation of a U(VI)–Pu(IV) mixture, and gelation of a U(IV)–Pu(III) mixture.

In the fuel refabrication context, the first mode would correspond to a U(VI)–Pu(IV)–Np(V,VI) –Am(III)–Cm(III) co-conversion, having the advantage of concerning initial oxidation states that are stable in a nitric medium, while hinting however at a foreseeable limitation, with respect to full minor actinide hydrolysis, as regards Np in particular.

The second mode would correspond to a U(IV)–Pu(III) –Np(IV)–Am(III)–Cm(III) co-conversion, in a reducing[6] medium, with optimizations required as regards implementation conditions, in order to achieve simultaneous, homogeneous gelation of (IV) and (III) actinides, an issue that has been very little investigated to date.

Tests have been carried out, on achieving formation of uranium and plutonium hydroxide[7] droplets, as soon as these are injected into a heated substrate, and of U(VI)–Pu(IV) and U(IV)–Pu(III) hydroxide gel microspheres (see Figure 6). These tests are currently being followed through, with the investigation of heat treatment conditions for such microspheres, to obtain actinide oxide kernels exhibiting good mechanical strength. The investigations carried out have made it possible to identify a domain of optimum conditions, as regards the two major actinides. Tests to investigate the behavior of minor actinides [Np(V,VI) and Am(III)] are currently ongoing.

The oxalic path

Metal cations exhibiting degrees of oxidation III, and IV interact with oxalate anions, to form com-

(5) Oxidation state (or oxidation number, degree of oxidation): a number indicating the number of electrons that must be supplied to, or stripped from, an atom, whether free or in a compound, to make it neutral. An increased number corresponds to a reduction, a decrease to an oxidation.

(6) Reduction: the opposite process to an oxidation reaction, i.e. a reaction whereby an atom or an ion gains electrons (yielded by a reducing agent), thus reaching the reduced state.

(7) Hydroxide: featuring the OH– anion.

plexes[8] that are poorly soluble in a nitric solution. Application of this property, already used as it is on an industrial basis for the conversion of plutonium nitrate into oxide, was initially considered for U–Pu co-conversion operations, then, more recently, for the group co-conversion of U–Pu–Np–Am–Cm actinides, according to the following overall reaction:

$$An_1^{n_1+}, n_1 NO_3^- + An_2^{n_2+}, n_2 NO_3^- + \frac{n_1+n_2}{2} H_2C_2O_4 + nH_2O \rightarrow An_1An_2\,(C_2O_4)_{\frac{n_1+n_2}{2}}, nH_2O + (n_1+n_2)HNO_3$$

Precipitation conditions (in terms of concentration, acidity, hydrodynamics…), as a batch[9] or continuous process, in a controlled atmosphere, have been specified for various actinides, separately at first (U, Pu…), then for mixtures (U–Pu, Pu–Np, Pu–Am, and U–Pu–Np) in a variety of proportions, to achieve precipitation efficiencies higher than 99%.

The outcomes of these tests (as illustrated in Figure 7) are encouraging: precipitation efficiencies are excellent, and structural analysis of the solids formed indeed shows the formation of a solid solution of actinides, with no segregation between actinides. For instance, batches of several tens of grams of $(Pu,Am)O_2$, with americium contents ranging from 20% to 80%, were obtained through this technique for the FUTURIX experiments (see *Phénix, a unique instrument in the area of fuel,* p. 98). With regard to process optimization, involving all actinides entails acquisition of targeted, relevant experimental data. Process modeling, on the basis of a coupling between chemical engineering (equipment, hydrodynamics) and precipitation chemistry is an indispensable step. As of now, on the basis of the experiments carried out in ATALANTE, operational feedback from the functioning of industrial equipment, and advances in numerical modeling, an initial, complete model of oxalic precipitation has been finalized. The prospects, as regards investigations for the development of the oxalic precipitation process, will presently be concerned with the chemistry of group actinide coprecipitation, process engineering, the adjustment of heat treatments to obtain oxide, carbide, or nitride compounds, and gaining sufficient data for the safety demonstrations, with a view to industrial deployment.

Pyrometallurgical processes

The qualities generally attached to molten halide salts: the ability to attack, and dissolve compounds as a rule resistant to nitric acid, high radiolysis resistance, and low moderation capability with respect to **neutrons**, allowing both highly irradiating, and highly **enriched** fuels to be worked with, potentially confer on pyrochemical treatment processes a character of compactness in use, these thus standing as an advantageous alternative to hydrometallurgical processes. As with the latter, the challenge lies in the full separation, and full recovery, in terms of quantity, of actinides from the salt bath, upstream of refabrication. The process affording the highest potential, as regards carrying out this operation, is reductive chemical extraction by means of a molten metal. One binary system of interest, for such separation, is the aluminum–fluoride system: LiF–AlF_3/Al–Cu (high selectivity with regard to extraction of actinides, compared with **fission products**, high actinide solubility in aluminum…).

Tests have been carried out on the separation of actinides at 830 °C, using an LiF–AlF_3 (85–15) salt bath charged with lanthanides (lanthanum, cerium, samarium, europium), for a Pu/Ln ratio of 3, by means of the Al–Cu (78–22) metal alloy. The distribution coefficients (D: namely mass D) measured for the various metallic elements, as distributed across the salt and metal alloy, are set out in the Table, together with the actinide–lanthanide separation factors ($S_{Pu/M}$) obtained subsequent to phase contact, and equilibrium (see Figure 8).

These findings confirm the excellent extraction of actinides from the salt by aluminum, and the potentials of this binary system as regards actinide and lanthanide separation. In the test conditions, two or three contacts between the salt and aluminum would, in effect, be adequate to ensure actinide extraction efficiencies higher than 99.9%.

In order to finalize the feasibility demonstration for this process, two avenues of investigation are being pursued concurrently. The first one involves tests on the group extraction of actinides from a fission-product-rich salt. For the purposes of this demonstration, a batch molten salt–liquid metal contactor, suitable for a nuclear environment, was constructed, for tests to be carried out in ATALANTE. The second avenue is concerned with hydrodynamic phase investigations, and the development of multistage contactors, to ensure a technology is available, allowing, on the one hand, the testing, in actual conditions, of schemes for the separation of actinides from a salt bath yielded by the dissolution of spent fuel, and such as may, on the other hand, be transferred to industrial-scale equipment.

metal	mass distribution coefficient	separation coefficient S (Pu/M)
Pu	197 ± 30	1
Am	144 ± 20	1.4 ± 0.4
Ce	0,.14 ± 0,01	1,307 ± 308
Sm	0.06 ± 0.1	3,177± 760
Eu	< 0.013	> 15,000
La	< 0.06	> 3,000

Table.
Initial findings from tests on the selective separation of minor actinides from lanthanides, in fluoride salts, by molten aluminum.
The mass distribution coefficient is the ratio of the amount of a given chemical element per solid phase unit mass, over the concentration of the dissolved element in the liquid phase.

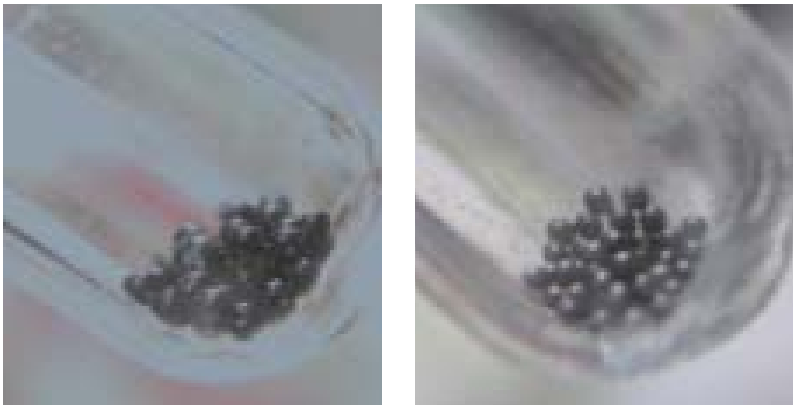


Figure 6.
Co-conversion of uranium and plutonium, in an 85/15 ratio, by internal gelation. The two views show mixed oxide microspheres, as obtained before, and after calcination at 900 °C in an argon and hydrogen atmosphere.

P. Stroppa/CEA.

(8) Complexation: the formation, from extractant systems and the species being extracted, of a structure (or complex compound) comprising ions, and molecules. A complexing molecule is a chemical species having the ability to bind with a metal ion, to form a complex compound.

(9) Batch: in chemical synthesis, this refers to a process involving batches, i.e. discrete amounts of materials, as opposed to a continuous process.

Gradualness and flexibility

The recycling of actinides is a key factor for a sustainable development of nuclear energy. Fuel cycle strategies for fourth-generation systems are still, at this stage, very much open (**homogeneous**, or **heterogeneous recycling** of all, or part of the actinides), as indeed the design options for reactors, or fuel elements. The whole, obviously, must be considered consistently, in design studies. With regard to the cycle, two further components must be taken on board, that do appear important – and, first of all, a conception of gradualness. Should what some are hailing as the "resurgence," or "rebirth" of the nuclear option be confirmed, with a swift deployment of nuclear capacities in various parts of the world, it would appear advisable to put to advantage, at the earliest opportunity, the recycling potentials that the "best available technologies" may afford, by deploying with no undue delay processes of the COEX™ type, which ensure the full, joint recycle of major actinides, thus meeting, as of now, the criteria of savings in terms of resources, and enhanced proliferation resistance. More advanced options may then be deployed at a later date, in a gradual approach, further going on to – quite conceivably – the full **multi-recycling** of actinides. The latter option, the most complete and comprehensive in some respects, is at the same time the most complex one: it entails a thoroughgoing rethink of recycling operations, to preserve fuel cycle economic competitiveness.

The second conception that then becomes imperative is one of flexibility: in future years, the reactor fleet, across the world, will probably be highly diversified, as indeed fuel requirements. The issue, therefore, may not be for R&D to seek to specify a single, optimum fuel cycle, rather it may be, conceivably, one of developing processes that are sufficiently robust, and flexible to adjust to diverse fuel objects, and a variety of recycling strategies.

CEA stands, currently, as a crucial player in the investigations involved with research and development work for these novel cycle options, for fourth-generation reactors. In the ATALANTE laboratories, in particular, drawing on the major results achieved over the past 15 years, in investigations on advanced recycling processes, and taking advantage of the facility's remarkable potentialities, researchers will be striving to investigate a variety of concepts, for various types of fuel, to support diverse management strategies, their contribution ranging from concept design studies to major demonstration experiments.

> **Bernard Boullis*** and **Michaël Lecomte****
Nuclear Energy Division
CEA Saclay and Marcoule** Centers*

Figure 7.
Co-conversion tests with actinides at various degrees of oxidation, carried out in ATALANTE.

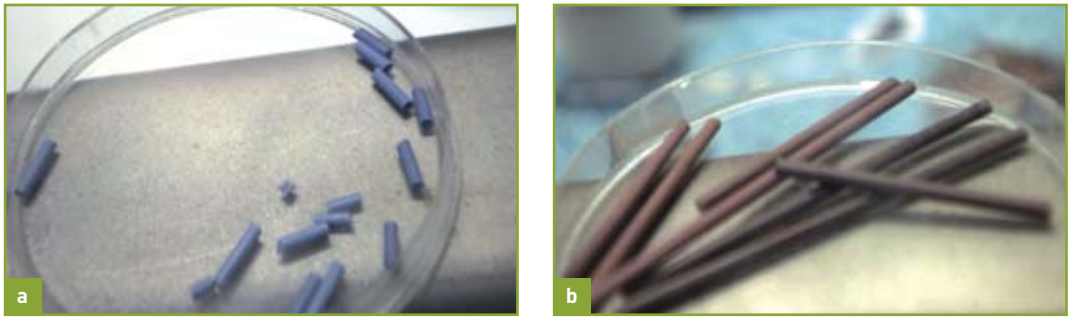


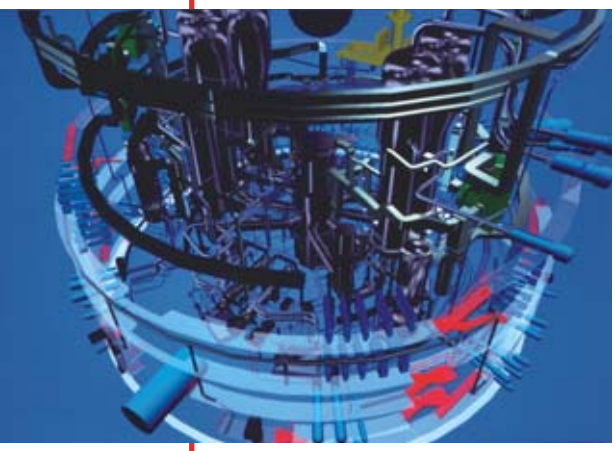
Figure 8.
a) The salt seen before contact [blue color, characteristic of Pu(III)].
b) Salt after contact (brown color, characteristic of lanthanides), after extraction of the actinides.

FOCUS A

The components of a nuclear system

A **nuclear system** comprises a **nuclear reactor** and the **fuel cycle** associated to it. It is the object of overall optimization, when industrially deployed – from raw materials to waste. In such a system, for which it forms the lynchpin, the reactor is given the ability to **recycle** fuel – so as to recover for value-added purposes **fissile** materials (**uranium**, **plutonium**), or even **fertile** materials (uranium, **thorium**) – and to minimize, through **transmutation**, production of **long-lived** waste, by **burning**, to a large extent, its own waste – namely, the **minor actinides** (**MAs**). Some systems may also feature online **reprocessing** plants.

The reactor itself, whichever **technology line** it may come under (see Focus B, *Reactor lines, generations, and neutron spectra*, p. 14), invariably comprises the same main components (as regards **fission** technology at any rate, since **fusion** reactors make use of altogether different nuclear processes).



Areva NP

Virtual 3D imagery of the components and circuits in a reactor of the PWR type.

The **core**, i.e. the area where **chain reactions** are sustained, holds the **fuel**, bearing fissile, energy-yielding materials (**heavy nuclei**), as well as fertile materials which, subjected to the action of **neutrons**, turn in part into fissile materials. The fuel may come in a number of forms (**pellets**, pebbles, particles), and **fuel elements** may be brought together in **rods**, **pins**, or plates, these in turn being grouped together in **assemblies**, as is the case, in particular, in water-cooled reactors.

The **moderator**, when required, plays an essential part. This is a material consisting in light **nuclei**, which slow down neutrons by way of *elastic scattering*. It must exhibit low **neutron-capture** capability, if neutron "wastage" is to be avoided, and sufficient density to ensure effective slowing down. **Thermal-spectrum** reactors (see Focus B) require a moderator – as opposed to **fast-spectrum** reactors (which, on the other hand, must compensate for the low probability of **fast-neutron**-induced fission through a steep rise in neutron numbers) – to slow down the neutrons, subsequent to the fission that yielded them, to bring them down to the optimum velocity, thus ensuring in turn further fissions. One example of a moderator is graphite, which was used as early as the first atomic "pile," in 1942, associated to a gas as coolant fluid.

The coolant fluid removes from the core the thermal energy released by fission processes, and transports the calories to systems that will turn this energy into useable form, electricity as a rule. The coolant is either water,(1) in "water reactors" (where it also acts as moderator), or a liquid metal (sodium, or lead), or a gas (historically, carbon dioxide, and later **helium**, in **gas-cooled reactors** [**GCRs**]), or yet **molten salts**. In the last-mentioned case, fuel and coolant are one and the same fluid, affording the ability to **reprocess** nuclear materials on a continuous basis, since the actinides are dissolved in it.

The choice of technology line has major repercussions on the choice of materials (see Focus E, *The main families of nuclear materials*, p. 76). Thus, the core of fast-neutron reactors may not contain neutron-moderating substances (water, graphite), and their coolant must be transparent to such neutrons.

Control devices (on the one hand, **control rods**, or **pilot** and **shutdown rods**, made of neutron-absorbent materials [boron, cadmium...], and, on the other hand, **neutron "poisons"**) allow the neutron population to be regulated and, in the process, by acting on its **reactivity**, to hold reactor power at the desired level, or even to quench the chain reaction. The rods, held integral and moving as one unit (known as a **cluster**) are inserted more or less deeply into the core. Poisons, on the other hand, may be adjusted in concentration within the cooling circuit.

A closed, leakproof, **primary circuit** contains the core, and channels and propels (by means of **circulators** – pumps or compressors) the coolant, which transfers its heat to a **secondary circuit**, by way of a **heat exchanger**, which may be a **steam generator** (this being the case equally in a pressurized-water reactor, or in the secondary circuit of a **fast reactor** such as Phénix). The **reactor vessel**, i.e. the vessel holding the core immersed in its cooling fluid, forms, in those cases when one is used, the main component of this primary circuit.

The secondary circuit extends out of the "nuclear island," to actuate, by way of a turbine, a turbo-alternator, or to feed a heat-distribution network. In **heavy-water** reactors,(1) and in some gas-cooled reactors, heat is transferred from gas to water in conventional heat exchangers.

A **tertiary circuit** takes off the unused heat, by way of a **condenser**, to a cold source (water in a river, or the sea), or the air in a cooling tower, or yet some other thermal device (e.g. for hydrogen production).

Other components are only found in certain reactor lines, such as the **pressurizer** in **pressurized-water reactors** (**PWRs**), where pressurization keeps the water in the liquid state by preventing it from boiling. On the other hand, boiling is put to work in **boiling-water reactors** (**BWRs**), the other line of **light-water reactors** (**LWRs**), where the primary circuit water comes to the boil, and directly actuates the turbine.

(1) *Heavy water*, in which **deuterium** is substituted for the **hydrogen** in ordinary water, was the first kind of moderator, used for reactor concepts requiring very low neutron absorption. *Light water* became the norm for operational, second-generation reactors. For the future, *supercritical water*, for which thermodynamic and transport properties are altered as it goes through the critical point (temperature of 374 °C, for a pressure higher than 22 MPa [221 bars, i.e. some 200 times atmospheric pressure]), may be used, to enhance the reactor's **Carnot efficiency** (see Focus C, *Thermodynamic cycles and energy conversion*, p. 23).

FOCUS B

Reactor lines, generations, and neutron spectra

M. Brigaud/ EDF Médiathèque

The four PWR units of EDF's Avoine power station, near Chinon (central France), belong to the second generation of nuclear reactors.

Nuclear reactor lines correspond to the many combinations of three basic components: **coolant**, **moderator** (when required), and **fuel** – almost invariably **uranium**, possibly mixed with **plutonium** (see Focus A, *The components of a nuclear system*, p. 10).

Numerous setups have been experimented with since the onset of the industrial nuclear energy age, in the 1950s, though only a few of these were selected, for the various generations of operational power generating reactors.

The term **technology line**, or **reactor line**, is thus used to refer to one possible path for the actual construction of nuclear reactors having the ability to function under satisfactory safety and profitability conditions, and defined, essentially, by the nature of the **fuel**, the energy carried by the **neutrons** involved in the **chain reaction**, the nature of the moderator, and that of the **coolant**.

The term is used advisedly, implying as it does that this combination stands as the origin of a succession of reactors, exhibiting characteristics of a technological continuum. More or less directly related to this or that line are research and trials reactors, which are seldom built as a series.

Such reactor lines are classified into two main families, depending on the **neutron spectrum** chosen: **thermal**, or **fast** (an operating range partly straddling both domains is feasible, for research reactors), according to whether neutrons directly released by **fission** are allowed to retain their velocity of some 20,000 km/s, or whether they are slowed down to bring them into thermal equilibrium (**thermalizing** them) with the material through which they scatter. The neutron spectrum, i.e. the energy distribution for the neutron population present within the **core**, is thus a **thermal spectrum** in virtually all reactors in service around the world, in particular, in France, for the 58 **PWRs** (**pressurized-water reactors**) in the **EDF** fleet. In these reactors, operating with **enriched uranium** (and, in some cases, **plutonium**), heat is transferred from the core to **heat exchangers** by means of water, kept at high pressure in the **primary circuit**.

Together with **BWRs** (**boiling-water reactors**), in which water is brought to the boil directly within the core, PWRs form the major family of **light-water reactors** (**LWRs**), in which ordinary water plays the role both of coolant, and moderator.

Use of the **fast spectrum** is, currently, restricted to a small number of reactors, operated essentially for experimental purposes, such as Phénix, in France, Monju and Joyo, in Japan, or BOR-60, in Russia. In such **fast reactors** (**FRs**), operating as they do without a moderator, the greater part of **fission** processes are caused by neutrons exhibiting energies of the same order as that they were endowed with, when

yielded by fission. A few reactors of this type have been built for industrial production purposes (Superphénix in France, BN600 in Russia), or investigated with such a purpose in mind (mainly EFR, a European endeavor, in the 1980s and 1990s, BN800 in Russia, CEFR in China, PFBR in India).

Electrical power generation reactors fall into four generations. The *first generation* covers reactors developed from the 1950s to the 1970s, which made possible the takeoff of nuclear electricity production in the various developed countries, comprising in particular the UNGG (or NUGG: natural uranium–graphite–gas) line, using graphite as moderator, and carbon dioxide as coolant, in France; the Magnox line, in the United Kingdom; and, in the United States, the first land-based[1] pressurized-water reactor (PWR), built at Shippingport.

While comparable in some respects to first-generation reactors, the Soviet Union's RBMK line (the technology used for the reactors at Chernobyl) is classed under the second generation, owing, in particular, to the time when it came on stream. RBMK reactors, using graphite as moderator, and cooled with ordinary water, brought to boil in pressure tubes, or channels, were finally disqualified by the accident at Chernobyl, in 1986.

The *second generation* covers those reactors, currently in service, that came on stream in the period from the 1970s to the 1990s. Solely built for electricity generation purposes, most of these (87% of the world fleet) are water-cooled reactors, with the one outstanding exception of the British-built AGRs (advanced gas-cooled reactors). The standard fuel they use consists of sintered enriched uranium-oxide pellets, to about 4% uranium-235 enrichment, stacked in impervious tubes (rods), which, held together in bundles, form assemblies. PWRs hold the lion's share of the market, accounting for 3 nuclear reactors out of 5 worldwide. This line includes the successive "levels" of PWR reactor models built, in France, by Framatome (now trading as Areva NP) for national power utility EDF. Russian reactors from the VVER 1000 line are comparable to the PWRs in the West. While operated in smaller numbers than PWRs, BWRs (boiling-water reactors) are to be found, in particular, in the United States, Japan, or Germany. Finally, natural-uranium powered reactors of the CANDU type, a Canadian design, and their Indian counterparts, form a line that is actively pursued. These are also pressurized-water reactors, however they use heavy water (D_2O) for their moderator, and coolant, hence the term PHWR (pressurized-heavy-water reactor) used to refer to this line.

The *third generation* corresponds to installations that are beginning to enter construction, scheduled to go on stream from around 2010. This covers, in particular, the French–German EPR, designed by Areva NP (initially: Framatome and Siemens), which company is also putting forward a boiling-water reactor, the SWR-1000, at the same time as it has been coming together with Japanese firm Mitsubishi Heavy Industries. This generation further includes the AP1000 and AP600 types from Westinghouse, a firm now controlled by Toshiba; the ESBWR and ABWR II from General Electric, now in association with Hitachi; the Canadian ACRs, and the AES92 from Russia; along with projects for smaller integral reactors.

Programs for modular high-temperature reactors, of the GT–MHR (an international program) or PBMR (from South African firm Eskom) type, belong to the third generation, however they may be seen as heralding *fourth-generation* reactors.

The fourth generation, currently being investigated, and scheduled for industrial deployment around 2040, could in theory involve any one of the six concepts selected by the Generation IV International Forum (see Box, in *The challenges of sustainable energy production*, p. 6). Aside from their use for electricity generation, reactors of this generation may have a cogeneration capability, i.e. for combined heat and power production, or even, for some of models, be designed solely for heat supply purposes, to provide either "low-temperature" (around 200 °C) heat, supplying urban heating networks, or "intermediate-temperature" (500–800 °C) heat, for industrial applications, of which seawater desalination is but one possibility, or yet "high- (or even very-high-) temperature" (1,000–1,200 °C) heat, for specific applications, such as hydrogen production, biomass gasification, or hydrocarbon cracking.

(1) In the United States, as in France, the first pressurized-water reactors were designed for naval (submarine) propulsion.

FOCUS C

Thermodynamic cycles and energy conversion

In the large-scale conversion of heat into electricity, a **thermodynamic cycle** must be involved. Conversion efficiency η is always lower than the **Carnot efficiency**:

$$\eta = 1 - \frac{T_c}{T_h}$$

where T_h is the temperature of the hot source, and T_c is the temperature of the cold source.

Generally speaking, a distinction is made, for energy conversion, between the **direct cycle**, whereby the fluid originating in the hot source directly actuates the device using it (a turbo-alternator, for instance), and, conversely, the **indirect cycle**, whereby the cooling circuit is distinct from the circuit ensuring the energy conversion itself. The **combined indirect cycle** may complement this setup by adding to it a gas turbine, or, by way of a steam generator, a steam turbine.

Any system built around a nuclear generator is a heat engine, making use of the principles of thermodynamics. Just as fossil-fuel- (coal-, fuel oil-) burning thermal power plants, nuclear power plants use the heat from a "boiler," in this case delivered by **fuel elements**, inside which the **fission** processes occur. This heat is converted into electric energy, by making a fluid (water, in most reactors currently in service) go through an *indirect* thermodynamic **cycle**, the so-called **Rankine** (or **Hirn–Rankine**) cycle, consisting of: water vaporization at constant pressure, around the hot source; expansion of the steam inside a turbine; condensation of the steam exiting the turbine at low pressure; and compression of the condensed water to bring that water back to the initial pressure. In this arrangement, the circuit used for the water circulating inside the core (the **primary circuit**; see Focus A, ***The components of a nuclear system***, p. 10) is distinct from the circuit ensuring the actual energy conversion. With a maximum steam temperature of some 280 °C, and a pressure of 7 **MPa**, the net energy efficiency (the ratio of the electric energy generated, over the thermal energy released by the reactor core) stands at about one third for a second-generation pressurized-water reactor. This can be made to rise to 36–38% for a third-generation PWR, such as **EPR**, by raising the temperature, since the Carnot equation clearly shows the advantage of generating high-temperature heat, to achieve high efficiency. Indeed, raising the core outlet temperature by about 100 degrees allows an efficiency improvement of several points to be achieved.

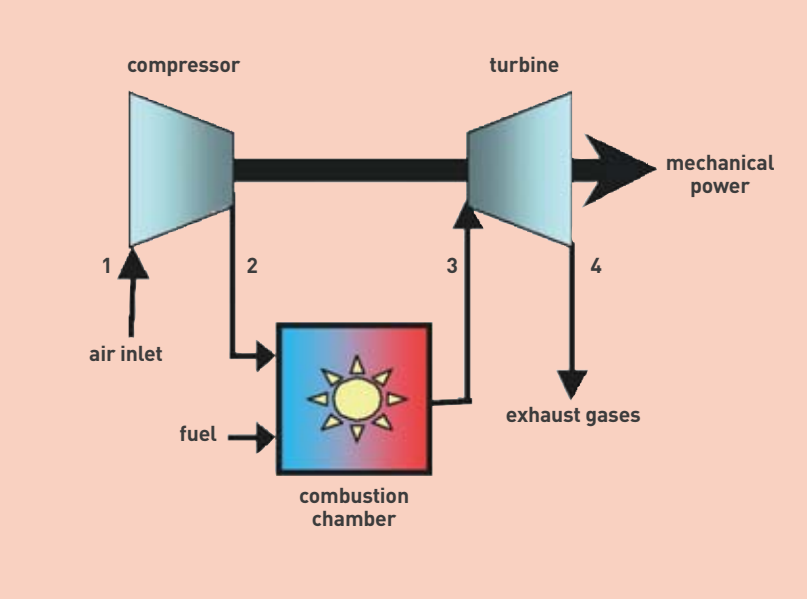


Figure.
Brayton cycle, as implemented in an open-cycle gas turbine.

The thermodynamic properties of a coolant gas such as helium make it possible to go further, by allowing a target core outlet temperature of at least 850 °C. To take full advantage of this, it is preferable, in theory, to use a **direct** energy conversion **cycle**, the **Joule–Brayton cycle**, whereby the fluid exiting the reactor (or any other "boiler") is channeled directly to the turbine driving the alternator, as is the case in natural-gas, **combined-cycle** electricity generation plants, or indeed in a jet aero-engine. Using this cycle, electricity generation efficiency may be raised from 51.6% to 56%, by increasing Tc from 850 °C to 1,000 °C.

Indeed, over the past half-century, use of natural gas as a fuel has resulted in a spectacular development of gas turbines (GTs) that can operate at very high temperatures, higher than around 1,000 °C. This type of energy conversion arrangement stands, for the nuclear reactors of the future, as an attractive alternative to steam turbines. GT thermodynamic cycles are in very widespread use, whether for propulsion systems, or large fossil-fuel electricity generation plants. Such cycles, known as **Brayton cycles** (see Figure) simply consist of: drawing in air, and compressing it to inject it into the combustion chamber (1 → 2); burning the air–fuel mix inside the combustion chamber (2 → 3); and allowing the hot gases to expand inside a turbine (3 → 4). On exiting the turbine, the exhaust gases are discharged into the atmosphere (this forming the cold source): the cycle is thus termed an *open* cycle. If the hot source is a nuclear reactor, open-cycle operation, using air, becomes highly problematical (if only because of the requisite compliance with the principle of three confinement barriers between nuclear fuel and the ambient environment). In order to *close* the cycle, all that is required is to insert a heat exchanger at the turbine outlet, to cool the gas (by way of a heat exchanger connected to the cold source), before it is reinjected into the compressor. The nature of the gas then ceases to be dictated by a combustion process.

What is multiphysics, multiscale modeling?

Multiphysics, multiscale modeling is a relatively recent R&D approach, arising out of the requirement to take into account, when modeling a system for which behavior is to be predicted, all processes – these in practice being coupled one with another – acting on (or prevailing in) that system. This is the most complete form of modeling, for a concatenation of various processes, of highly diverse scales, bringing together as it does all of the relevant knowledge, whether theoretical or empirical, at a variety of scales, into elementary building blocks, which then have to be assembled.

In physical terms, this takes into account the couplings arising between basic processes of diverse nature. In the area of reactor physics, for instance, coupling occurs between structural mechanics, neutronics, and thermal–hydraulics.

This kind of modeling further aims to provide a description of processes at different scales. In the area of materials physics, the aim will be, e.g., to derive the macroscopic properties of a polycrystalline material, from its description at the most microscopic scale (the atom), by way of nested levels of description (molecular dynamics, dislocation dynamics).

The issue is that of connecting these various levels of description, by using the correct information to pass from one scale to the next with no break in continuity, and of handling in modular fashion such behavior laws, valid as these are at diverse scales (see Figure).

Thus it is numerical computation of a composite character, depending on the spatial scale being considered, that "drives" the overall model. All the more composite, since researchers are led to "chain" *deterministic*, and *probabilistic* models, whether it be for lack of an exhaustive knowledge of the basic processes involved, or because the numerical resolution of the deterministic equations would prove too difficult, or too heavy a task. Hence the adoption of such methods as the Monte-Carlo method, in particular.

Finally, multiscale modeling joins up, through superposition techniques, numerical models at different scales. This makes it possible – to stay with the example of materials – to "zoom in" on regions that are particularly sensitive to stresses, such as fissures, welds, or supporting structures.

Multiphysics, multiscale modeling thus raises, in acute fashion, the issue of the compatibility, and consistency of the computation codes making up the elementary building blocks in the description. However, the outcomes are on a par with the difficulty: in the area of metallic materials, in particular, it is now possible to implement an approach predicting macroscopic properties from "first principles," of atomic physics and molecular dynamics (*ab-initio* method, see note (1) p. 79), by way of the physical description of microstructures. In the nuclear energy context, the investigation of materials subjected to irradiation provides a good illustration of this approach, since it has now become feasible to bridge the gap between knowledge of defects at the macroscopic scale, and modeling of point defect formation processes, at the atomic scale.

While physics naturally provides the first level, in this type of modeling, the two other levels are mathematical, and numerical, insofar as the point is to connect findings from measurements, or computations, valid at different scales, going on to implement the algorithms developed. Multiphysics, multiscale modeling has thus only been made possible by the coming together of two concurrent lines of advances: advances in the knowledge of basic processes, and in the power of computing resources.

CEA is one of the few organizations around the world with the capability to develop such multiphysics, multiscale modeling, in its various areas of research and development activity, by bringing together a vast ensemble of modeling, experimental, and computation tools, enabling it to demonstrate, at the same time, the validity of theories, the relevance of technologies, and bring about advances in component design, whether in the area of nuclear energy (in which context coupling is effected between partial codes from CEA and EDF), or, for example, in that of the new energy technologies.

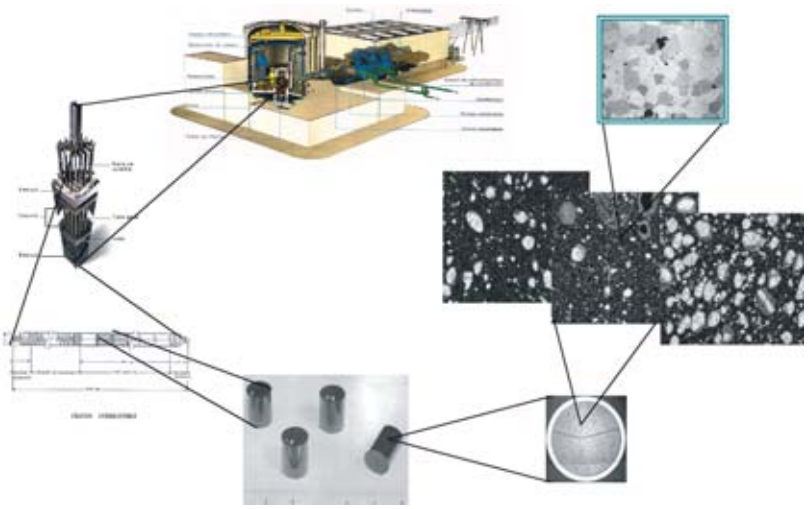


Figure.
Improving nuclear fuel reliability, and cost-effectiveness calls for finescale modeling of that fuel, through a multiscale approach, from reactor to fuel microstructure (in this instance, MOX fuel). Microstructural characteristics (porosity, cluster size and distribution, grain size...) have a direct impact on fuel rod behavior under irradiation, and thus on reactor ease of operation, and on that rod's lifespan.

The main families of nuclear materials

The specific conditions attributable to radiation conditions prevailing inside nuclear reactors mean it is imperative to look to materials exhibiting special characteristics, which may be grouped under two main categories: **cladding and structural materials**, on the one hand, and **fuel materials**, on the other. For either group, the six concepts for fourth-generation systems selected by the **Generation IV International Forum** mostly require going for innovative solutions, as the favored option (see Table, p. 71).

The characteristics, in terms of resistance to temperature, pressure, fatigue, heat, corrosion, often under stress, that should be exhibited, as a general rule, by materials involved in any industrial process must, in the nuclear energy context, be virtually fully sustained, notwithstanding the effects of irradiation, due in particular to the **neutron** flux. Indeed, irradiation speeds up, or amplifies processes such as **creep** (**irradiation creep**), or causes other ones, such as **swelling**, or **growth**, i.e. an **anisotropic** deformation occurring under the action of a neutron flux, in the absence of any other stress.

Structural materials in the reactor itself are subject, in particular, to the process of **activation** by neutron bombardment, or bombardment by other particles (**photons, electrons**).

Materials employed for fuel structures (**assemblies**, **claddings**, **plates**, and so on) are further subjected to yet other stresses. Finally, the **fuel** itself is a material, taking the form, in current **light-water reactors**, for instance, of **sintered uranium** and/or **plutonium ceramics**, in the form of pellets.

Neutron **irradiation** can cause a major alteration in the properties exhibited by the materials employed in the various components of a reactor. In metals, and metal alloys, but equally in other solid materials, such as ceramics,[1] such alterations are related to the evolution of the **point defects** generated by this irradiation, and to the extraneous **atoms** generated by nuclear reactions, substituting for one of the atoms in the **crystal** lattice. The nature, and number of such defects depends both on the neutron flux, and neutron energies, however the neutrons that cause appreciable structural evolutions are, in **thermal-neutron reactors** as in **fast-neutron reactors** (**fast reactors**), the **fast neutrons**.

A crystal invariably exhibits some defects, and irradiation may generate further defects. Point defects fall under two types: **vacancies** (one atom being expelled from its location in the crystal), and **interstitials** (one extra atom positioning itself at a supernumerary site, between the planes of the crystal lattice).

Dislocations, marking out a region where the crystal stack is disturbed by local slipping, affecting a single atomic plane, in turn act as *sources*, or *sinks* of point defects. Vacancies may come together to form **vacancy clusters**, **loops**, or **cavities**, while interstitials may form interstitial clusters, or *dislocation loops*. At the same time, copper, manganese, and nickel atoms, e.g. in a vessel steel alloy, tend to draw together, to form **clusters**, resulting in hardening of the steel. Finally, **grain boundary** are defects bounding two crystals exhibiting different orientations, and thus act as potential factors of embrittlement. Many of the metal's properties are subject to alteration at these boundaries.

The damage occasioned to such materials is expressed in terms of displacements per atom (**dpa**), with *n* dpa implying that every atom in the material has been displaced *n* times, on average, during irradiation.

Crystal structures

Metallic materials exhibit a crystal structure: they are formed by an elementary unit, periodically repeating across space, known as a unit **cell**, consisting of **atoms**, in precise, definite numbers and positions. Repetition of such structures endows them with specific properties. Three of these structures, defining the position of the atoms, are of importance:

- the **body-centered cubic structure** (that found in iron at ambient room temperature, chromium, vanadium); such materials as a rule exhibit a ductile–brittle behavior transition, depending on temperature;
- the **face-centered cubic structure** (nickel, aluminum, copper, iron at high temperature);
- the **hexagonal structure** (that of zirconium, or titanium).

Depending on temperature and composition, the metal will structure itself into elementary crystals, the **grains**, exhibiting a variety of microstructures, or *phases*. The way these arrange themselves has a major influence of the properties exhibited by metals, steels in particular. The **ferrite** of pure iron, with a *body-centered cubic structure*, turns into **austenite**, a *face-centered cubic structure*, above 910 °C. **Martensite** is a particular structure, obtained through *tempering*, which hardens it, followed by *annealing*, making it less brittle. **Bainite** is a structure intermediate between ferrite and martensite, likewise obtained through tempering followed by annealing.

Among metals, high-chromium-content (more than 13%) stainless steels, exhibiting as they do a corrosion and oxidation resistance that is due to the formation of a film of chromium oxide on their surface, take the lion's share. If the criterion for stainless ability (rustproofness) is taken to be chromium content, which should be higher than 13%, such steels fall into three main categories: ferritic steels, austenitic steels, and austenitic–ferritic steels.

Steel families

Ferritic steels, exhibiting a *body-centered* cubic structure (e.g. F17), are characterized by a low carbon concentration (0.08–0.20%), and high chromium content. As a rule containing no nickel, these are iron–chromium, or iron–chromium–molybdenum alloys, with a chromium content ranging from 10.5% to 28%: they exhibit no appreciable hardening when tempered, only hardening as a result of work hardening.

They exhibit a small expansion coefficient, are highly oxidation resistant, and prove suitable for high temperatures. In the nuclear industry, 16MND5 **bainitic steel**, a low-carbon, low-alloy (1.5% manganese, 1% nickel, 0.5% molybdenum) steel, takes pride of place, providing as it does the vessel material for French-built **PWRs**, having been selected for the qualities it exhibits at 290 °C, when subjected to a **fluence** of $3 \cdot 10^{19}$ n · cm^{-2}, for neutrons of energies higher than 1 **MeV**.

Martensitic steels, exhibiting a *body-centered cubic* structure, are ferritic steels containing less than 13% chromium (9–12% as a rule), and a maximum 0.15% carbon,

(1) Ceramics are used on their own, or incorporated into composites, which may be of the **cercer** (a ceramic held in a matrix that is also a ceramic) or **cermet** (a ceramic material embedded in a metallic matrix) types. With regard to nuclear fuel, this takes the form of a closely mixed composite of metallic products, and refractory compounds, the fissile elements being held in one phase only, or in both.

Areva NP

Pressure-vessel nozzle shell for EDF's Flamanville 3 reactor, the first EPR to be built on French soil.

which have been subjected to *annealing*: they become martensitic when quenched, in air or a liquid, after being heated to reach the austenitic domain. They subsequently undergo softening, by means of a heat treatment. They may contain nickel, molybdenum, along with further addition elements. These steels are magnetic, and exhibit high stiffness and strength, however they may prove brittle under impact, particularly at low temperatures. They have gained widespread use in the nuclear industry (fastenings, valves and fittings...), owing to their good corrosion resistance, combined with impressive mechanical characteristics.

Austenitic steels, characterized by a *face*-centered cubic structure, contain some 17–18% chromium, 8–12% nickel (this enhancing corrosion resistance: the greater part, by far, of stainless steels are austenitic steels), little carbon, possibly some molybdenum, titanium, or niobium, and, mainly, iron (the remainder). They exhibit remarkable **ductility**, and **toughness**, a high expansion coefficient, and a lower **heat conductivity** coefficient than found in ferritic–martensitic steels. Of the main grades (coming under US references AISI[2] 301 to 303, 304, 308, 316, 316L, 316LN, 316Ti, 316Cb, 318, 321, 330, 347), 304 and 316 steels proved particularly important for the nuclear industry, before being abandoned owing to their excessive swelling under irradiation. Some derivatives (e.g. 304L, used for internal structures and fuel assembly end-caps, in PWRs; or 316Tiε, employed for claddings) stand as reference materials. In fast reactors, they are employed, in particular, for the fabrication of **hexagonal tubes** (characteristic of reactors of the Phénix type) (316L[N] steel), while 15/15Ti austenitic steel has been optimized for fuel **pins** for this reactor line, providing the new cladding reference for fast reactors.

Austenitic–ferritic steels, containing 0%, 8%, 20%, 32%, or even 50% ferrite, exhibit good corrosion resistance, and satisfactory weldability, resulting in their employment, in molded form, for the ducts connecting vessels and steam generators.

One class of alloys that is of particular importance for the nuclear industry is that of **nickel alloys**, these exhibiting an austenitic structure. Alloy 600 (Inconel 600, made by INCO), a nickel (72%), chromium (16%), and iron (8%) alloy, further containing cobalt and carbon, which was employed for PWR steam generators (along with alloy 620) and vessel head penetrations, was substituted, owing to its poor corrosion resistance under stress, by alloy 690, with a higher chromium content (30%). For certain components, Inconel 706, Inconel 718 (for PWR fuel assembly grids), and Inconel X750 with titanium and aluminum additions have been selected, in view of their swelling resistance, and very high mechanical strength. For steam generators in fast reactors such as Phénix, alloy 800 (35% nickel, 20% chromium, slightly less than 50% iron) was favored. Alloy 617 (Ni–Cr–Co–Mo), and alloy 230 (Ni–Cr–W), widely employed as they are in the chemical industry, are being evaluated for gas-cooled **VHTRs**.

Ferritic–martensitic steels (F–M steels) exhibit a *body-centered cubic* structure. In effect, this category subsumes the martensitic steel and ferritic steel families. These steels combine a low thermal expansion coefficient with high heat conductivity. Martensitic or ferritic steels with chromium contents in the 9–18% range see restricted employment, owing to their lower creep resistance than that of austenitic steels. Fe–9/12Cr martensitic steels (i.e. steels containing 9–12% chromium by mass) may however withstand high temperatures, and are being optimized with respect to creep. For instance, Fe–9Cr 1Mo molybdenum steel might prove suitable for the hexagonal tube in **SFR** fuel assemblies. Under the general designation of AFMSs (advanced ferritic–martensitic steels), they are being more particularly investigated for use in gas-cooled fast reactors.

Oxide-dispersion-strengthened (ODS) **ferritic and martensitic steels** were developed to combine the swelling resistance exhibited by ferritic steels, with a creep resistance in hot conditions at least equal to that of austenitic steels. They currently provide the reference solution for fuel cladding, for future sodium-cooled reactors.

The **cladding material** in light-water reactors, for which stainless steel had been used initially, nowadays consists of a **zirconium alloy**, selected for its "transparency" to neutrons, which exhibits a *compact hexagonal* crystal structure at low temperature, a *face-centered cubic* structure at high temperature. The most widely used zirconium–iron–chromium alloys are tin-containing **Zircaloys** (Zircaloy-4 in PWRs, Zircaloy-2 in BWRs, ZrNb – containing niobium – in the Russian VVER line), owing to their outstanding behavior under radiation, and capacity with respect to creep in hot conditions.

After bringing down tin content, in order to improve corrosion resistance, a zirconium–niobium alloy (M5®) is presently being deployed for such cladding.

Among nuclear energy materials, **graphite** calls for particular mention: along with heavy water, it is associated with reactors that must operate on **natural uranium**; it proves advantageous as a **moderator**, as being a low neutron absorber.

For **GFRs**, novel ceramics, and new alloys must be developed, to the margins of high fluences. Researchers are storing high hopes on refractory materials containing no metals.

In particle fuels, uranium and plutonium oxides are coated with several layers of insulating **pyrocarbons**, and/or silicon **carbide** (**SiC**), possibly in fibrous form (**SiCf**). These are known as coated particles (CPs). While SiC-coated UO_2, or **MOX** balls stand as the reference, ZrC coatings might afford an alternative.

At the same time, conventional **sintered** uranium oxide (and plutonium oxide, in **MOX**) pellets might be supplanted by advanced fuels, whether featuring chromium additions or otherwise, with the aim of seeking to overcome the issues raised by **pellet–cladding interaction**, linked as this is to the ceramic fuel pellet's tendency to swell under irradiation.

Oxides might be supplanted by **nitrides** (compatible with the **Purex** reprocessing process), or **carbides**, in the form e.g. of uranium–plutonium alloys containing 10% zirconium.

(2) This being the acronym for the American Iron and Steel Institute.

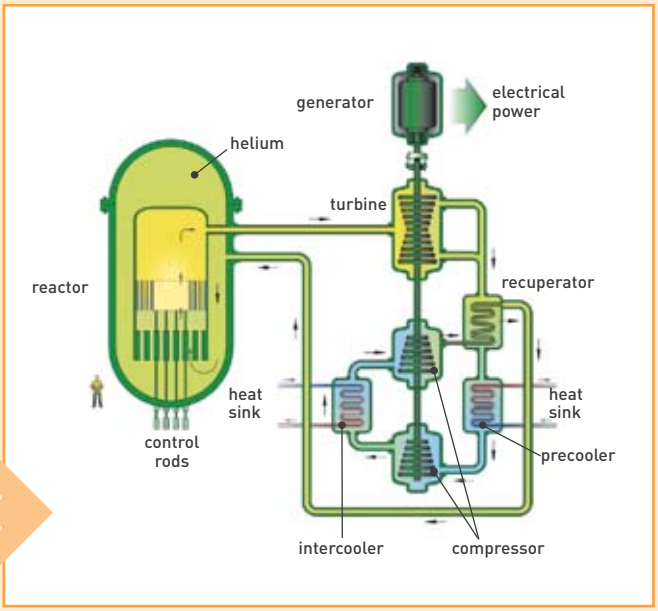
The six concepts selected by the Gen IV Forum

Of the six concepts selected by the Generation IV International Forum for their ability to meet the criteria outlined, three – and ultimately four – make use of fast neutrons, while three (ultimately two) use thermal neutrons. At the same time, two of the six concepts use gas as a coolant (they are thus gas-cooled reactors [GCRs]). The six concepts are the following:

GFR

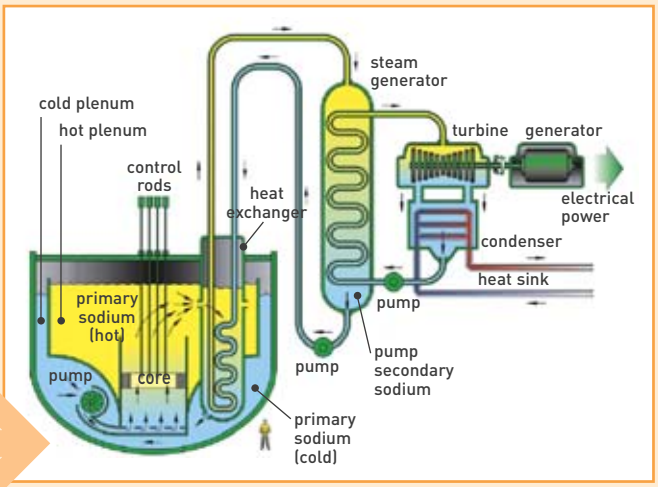
The gas-cooled fast reactor system (GFR) is a high-temperature, gas-cooled (helium-cooled as a rule), fast-neutron reactor allowing actinide recycle (homogeneous, or heterogeneous), while sustaining a breeding capability greater than unity. The reference concept is a helium-cooled, direct- or indirect-cycle reactor, exhibiting high efficiency (48%). Decay heat removal, in the event of depressurization, is feasible through natural convection a few hours after the accident. Maintaining forced circulation is a requisite, during the initial accident stage. Core power density is set at a level such as to restrict fuel temperature to 1,600 °C during transients. The innovative fuel is designed to retain fission products (at temperatures below the 1,600 °C limit), and preclude their release in accident conditions. Reprocessing of spent fuel for recycling purposes may be considered (possibly on the reactor site), whether by means of a pyrochemical or a hydrometallurgical process. The GFR is a high-performance system, in terms of natural resource utilization, and long-lived waste minimization. It comes under the gas-cooled technology line, complementing such thermal-spectrum concepts as the GT–MHR,(1) PBMR,(2) and VHTR.

(1) GT–MHR: Gas-Turbine Modular Helium Reactor.
(2) PBMR: Pebble-Bed Modular Reactor.



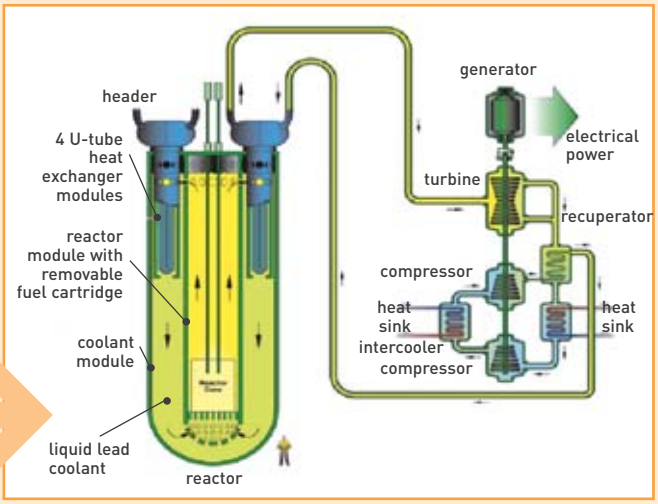
Le SFR

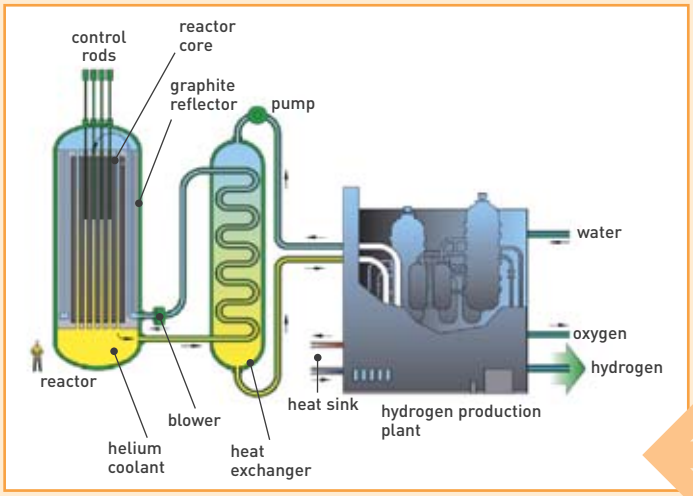
The sodium-cooled fast reactor system (SFR) is a liquid-sodium-cooled, fast-neutron reactor, associated to a closed cycle, allowing full actinide recycle, and plutonium breeding. Owing to its breeding of fissile material, this type of reactor may operate for highly extended periods without requiring any intervention on the core. Two main options may be considered: one that, associated to the reprocessing of metallic fuel, results in a reactor of intermediate unit power, in the 150–500 MWe range; the other, characterized by the Purex reprocessing of mixed-oxide fuel (MOX), corresponds to a high-unit-power reactor, in the 500–1,500 MWe range. The SFR presents highly advantageous natural resource utilization and actinide management features. It has been assessed as exhibiting good safety characteristics. A number of SFR prototypes are to be found around the world, including Joyo and Monju in Japan, BN600 in Russia, and Phénix in France. The main issues for research concern the full recycling of actinides (actinide-bearing fuels are radioactive, and thus pose fabrication difficulties), in-service inspection (sodium not being transparent), safety (passive safety approaches are under investigation), and capital cost reduction. Substitution of water with supercritical CO_2 as the working fluid for the power conversion system is also being investigated



LFR

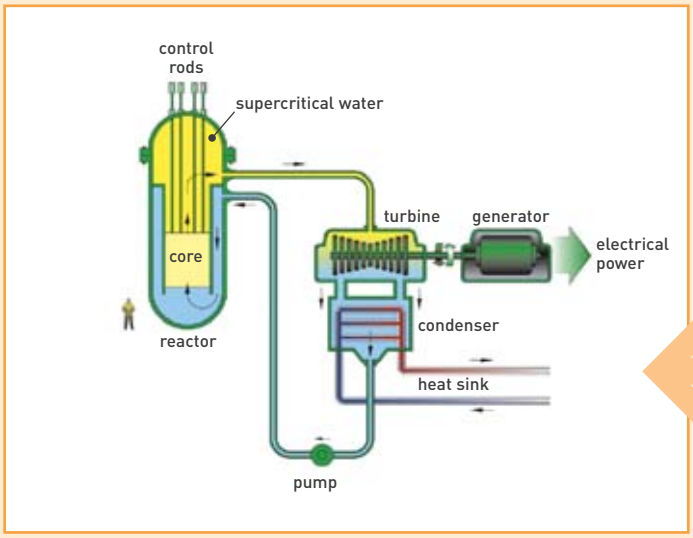
The lead-cooled fast reactor system (LFR) is a lead- (or lead–bismuth alloy-) cooled, fast-neutron reactor, associated to a closed fuel cycle, allowing optimum uranium utilization. A number of reference systems have been selected. Unit power ranges from the 50–100 MWe bracket, for so-called battery concepts, up to 1,200 MWe, including modular concepts in the 300–400 MWe bracket. The concepts feature long-duration (10–30 years) fuel management. Fuels may be either metallic, or of the nitride type, and allow full actinide recycle.





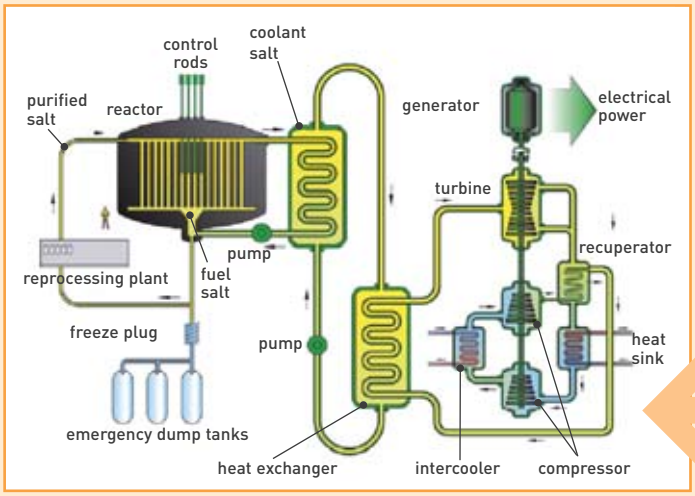
VHTR

The very-high-temperature reactor system (VHTR) is a very-high-temperature, helium-gas-cooled, thermal-neutron reactor, initially intended to operate with an open fuel cycle. Its strong points are low costs, and most particularly safety. Its capability, with regard to sustainability, is on a par with that of a third-generation reactor, owing to the use of an open cycle. It may be dedicated to hydrogen production, even while also allowing production of electricity (as sole output, or through cogeneration). The specific feature of the VHTR is that it operates at very high temperature (> 1,000 °C), to provide the heat required for water splitting processes, by way of thermochemical cycles (iodine–sulfur process), or high-temperature electrolysis. The reference system exhibits a unit power of 600 MWth, and uses helium as coolant. The core is made up of prismatic blocks, or pebbles.



SCWR

The supercritical-water-cooled reactor system (SCWR) is a supercritical-water-cooled, thermal-neutron reactor, in an initial stage (open fuel cycle); a fast-neutron reactor in its ultimate configuration (featuring a closed cycle, for full actinide recycle). Two fuel cycles correspond to these two versions. Both options involve an identical operating point, with regard to supercritical water: pressure of 25 MPa, and core outlet temperature of 550 °C, enabling a thermodynamic efficiency of 44%. Unit power for the reference system stands at 1,700 MWe. The SCWR has been assessed as affording a high economic competitiveness potential.



MSR

The molten salt reactor system (MSR) is a molten salt (liquid core, with a closed cycle, through continuous online pyrochemical reprocessing), thermal-neutron – more accurately epithermal-neutron – reactor. Its originality lies is its use of a molten salt solution, serving both as fuel, and coolant. Fissile material breeding is feasible, using an optional uranium–thorium cycle. The MSR includes as a design feature online fuel recycling, thus affording the opportunity to bring together on one and the same site an electricity-generating reactor, and its reprocessing plant. The salt selected for the reference concept (unit power of 1,000 MWe) is a sodium–zirconium–actinide fluoride. Spectrum moderation inside the core is effected by placing graphite blocks, through which the fuel salt flows. The MSR features an intermediate fluoride-salt circuit, and a tertiary, water or helium circuit for electricity production.