# What **fuel** for **SFRs?**

**Fuel for a fourth-generation sodium-cooled fast reactor will have to meet specifications involving novel requirements,** in terms of density, and thermic characteristics, and entailing further, new consequences as regards the core, assemblies, and fuel pins. Over the next two years, researchers are set to define fuels that will afford the capability to meet the specifications for reactors of the latest generation in this technology line.

P. Dumas/CEA

Cell 1 at LEFCA (Laboratoires d'études et de fabrications expérimentales de combustibles nucléaires avancés: Advanced Nuclear Fuels Design and Experimental Fabrication Laboratories), where experimental fuels are fabricated for future SFRs. View of the glovebox, allowing specific actinide powder mixing and grinding operations to be carried out.

The **fast reactor** (**FR**) concept calls for high **fissile** material density (see *Sodium-cooled fast reactors of the future,* p. 24). Of the various options for **fuel** materials (see Table), early designers quite logically went for a fuel exhibiting very high **heavy-atom** density: metallic fuel. Thus, the first models built, of liquid-metal-cooled fast reactors (using liquid **sodium** [Na], or a sodium–potassium [NaK] **eutectic**), in the United States, and later in the United Kingdom, burnt fuel of the metal **alloy** type. Mention must be made, in particular, of the US EBR-I (Experimental Breeder Reactor-I) reactor, initial criticality of which occurred in 1951, and which, furthermore, was the world's first **nuclear reactor** to generate electricity, with an output of 0.2 **MWe**! However, these early fuels, far from being optimized, exhibited extremely poor performance (**specific burnup** of a few **GW · d/t**), mainly owing to excessive **swelling** under **irradiation**.

C. Morgan/INL

The US EBR-I (Experimental Breeder Reactor-I) fast reactor, the world's first reactor to generate electricity, used fuel of the metal alloy type. Sited in Idaho (North West United States), this reactor operated from 1951 to 1964. In 1953, experiments showed the **breeding** process was feasible. The prototype reactor is visible in the foreground.

As early as the mid-1950s, urged on by Georges Vendryes, the "spiritual father" of the **breeder** reactor, CEA embarked on an R&D program, on the basis of **plutonium** (Pu) fuels, for the purposes of building an initial French fast reactor. Major work was done on the investigation of **uranium** (U) and plutonium metal alloys, involving an added third element, for the purposes of stabilizing the **cubic structure** of uranium. Investigations on the triptych "physical properties, fabrication process R&D, reactor behavior" proceeded apace, ultimately converging on the ternary uranium–plutonium–molybdenum (U–Pu–Mo) alloy. Unfortunately, as US researchers had found, for their part, such alloy fuels, when tested under the very harsh conditions of a fast reactor, would swell, owing to the growth of **fission-gas** bubbles, severely restricting in-reactor life-

| fuel materials | oxide $(U,Pu)O_2$* | carbide (U,Pu)C* | nitride (U,Pu)N* | metal alloy U-Pu-Zr* |
|---|---|---|---|---|
| theoretical heavy-atom density | 9.7 | 12.9 | 13.5 | 14.1 |
| melting temperature (°C) | 2,730 | 2,305 | 2,720** | 1,070 |
| thermal conductivity at 1,000 °C*** (W/m·K) | 2.1 | 12.8 | 13.5 | 17.5 |
| average thermal expansion coefficient 20→1,000 °C*** ($10^{-6}$/K) | 12.5 | 12.4 | 10 | 17 |
| countries having operational feedback on the fuel, on the scale of a fast reactor core | France, United States, United Kingdom, Germany, Russia, Japan | India, Russia | Russia**** | United States, United Kingdom |

* For 20% Pu. ** Partial breakdown may occur from 1,750 °C. *** At 500 °C for U–Pu–Zr. **** With enriched uranium only.

Table.
Characteristics of various fuel materials.

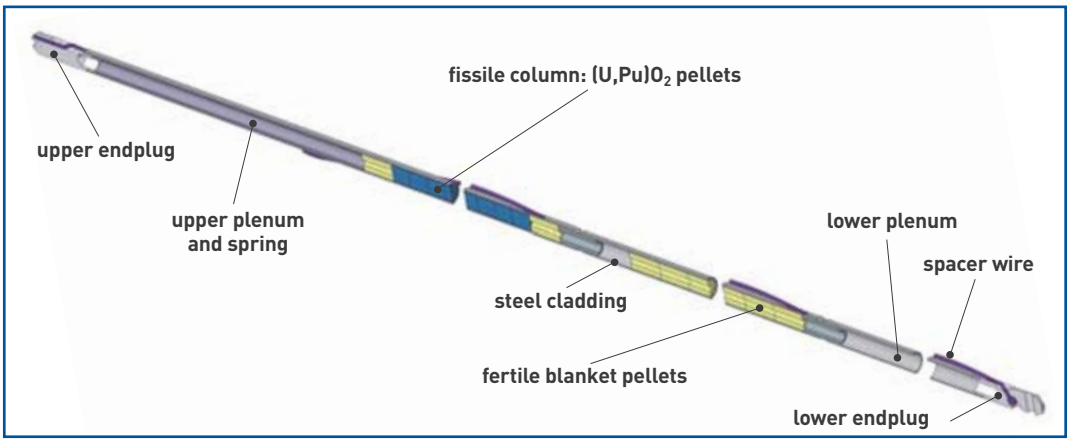


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Figure 1.
Exploded view of an SFR fuel pin, showing its main components.

time, and making frequent reprocessing of the fuel necessary.
Consequently, in the early 1960s, France, as indeed most of the other countries engaged in development of a first-generation fast reactor line, abandoned metallic fuel, turning instead to mixed uranium and plutonium oxides, exhibiting as they did far less temperamental behavior.(1) Although affording, in theory, poorer performance, $(U,Pu)O_2$ ceramic fuel soon made its mark as reference fuel for this reactor line, and development of this fuel provided the cornerstone for the programs conducted over more than three decades at CEA.

## Operational feedback regarding oxide fuel

In the French context, most findings under nominal operating conditions, for fuel pins (see Figure 1), were the outcome of experimental and monitoring programs conducted in the Rapsodie experimental fast reactor (1967–83), sited at Cadarache, and in the Phénix industrial prototype (1973–2009), built at CEA's Marcoule site, delivering an output power of 250 MWe. Further, many tests concerning behavior in accident or incident situations were carried out, under the aegis of international programs, over more than 25 years, in the Cabri and Scarabée experimental reactors, dedicated facilities for safety investigations.
SFR oxide fuel exhibits distinctively very high specific power, under nominal operating conditions (2 kW/cm³ in the case of Phénix(2)), which, combined with the mixed oxide's low thermal conductivity, results in the temperature, at the center of the pellets, commonly going beyond 2,000 °C. Indeed, the initial microstructure of the sintered ceramic pellets undergoes a remarkable evolution (see Figure 2).

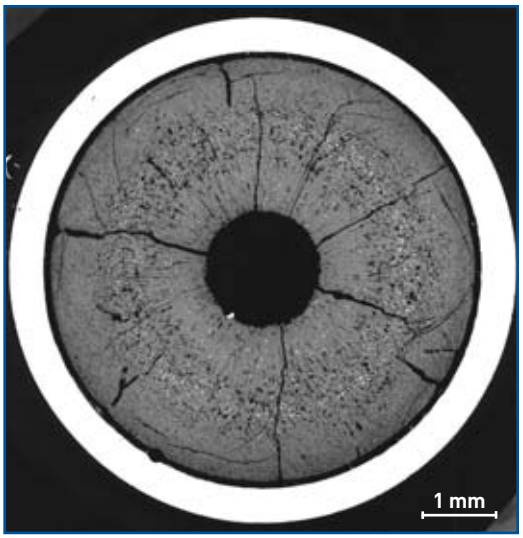


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Figure 2.
Transverse metallographic section of a Phénix fuel pin, showing the considerable microstructural evolution of the mixed-oxide fuel, during irradiation (main irradiation data: linear power at the plane of maximum neutron flux [section plane]: 390 W/cm at start of irradiation → 330 W/cm at end of irradiation; maximum burnup: 13.3% FIMA; nominal cladding temperature: 650 °C; irradiation time: 758 EFPD; cladding damage: 113.5 dpa). This examination was carried out at the Active Fuel Investigation Laboratory (LECA: Laboratoire d'examen des combustibles actifs, CEA/Cadarache).

(1) At the time, most power reactor projects were already considering use of uranium oxide as a fuel material.

(2) Maximum specific power stood at 1.3 kW/cm³ for the Superphénix industrial-scale reactor. By comparison, this only reaches 0.4 kW/cm³ for a pressurized-water reactor (PWR) in the EDF fleet.

A "restructuring" of the fuel occurs in its central region, this being due to as-fabricated porosity migrating to the center, and the "healing" of cracks, with the concomitant formation of columnar **grains**, and a central cavity. Such restructuring is accompanied by a major redistribution of plutonium concentration, as evidenced by a significant increase in the central region (see Figure 3).
The greater part of the gaseous **fission products** is released into the free space inside the fuel pin. Release rate may attain 90% of the products generated (see Figure 4). Only the outer region of the pellets retains some of this. Contrary what is the case for **pressurized-water reactor** (**PWR**) fuel, this in itself is not an issue, since this process is taken into account, from the dimensioning stage for the pins. At higher burnup (> 8% **FIMA**), volatile fission products (cesium [Cs], iodine [I], tellurium [Te]), or metallic fission products (molybdenum [Mo]) are partially released outside the fuel, ultimately forming an oxide–cladding bond named JOG (Joint Oxyde-Gaine), between the pellet outer regions and the inside wall of the stainless-steel cladding.
Concurrently, from about 5–6% FIMA burnup, internal corrosion of the cladding, to a depth which may be significant, occurs in the upper third of the fissile column (see Figure 5). The conditions for the onset of such corrosion are known. This involves the accumulation, at some locations on the cladding, of metallic tellurium, a volatile fission product, as a result of cesium telluride breakdown, owing to the fuel's increasing oxygen potential.(3) This internal corrosion process is seen as one of the limiting factors, as regards achieving high burnup. Remedies such as the use of getters(4) within the oxide, or the adoption of a protective coating for the inside surface of the cladding, may delay the onset of corrosion.
The other major drawback of the oxide lies in its chemical incompatibility with sodium. However, numerous out-of-reactor investigations, together with operational feedback from naturally-occurring cladding failures in the Rapsodie and Phénix reac-

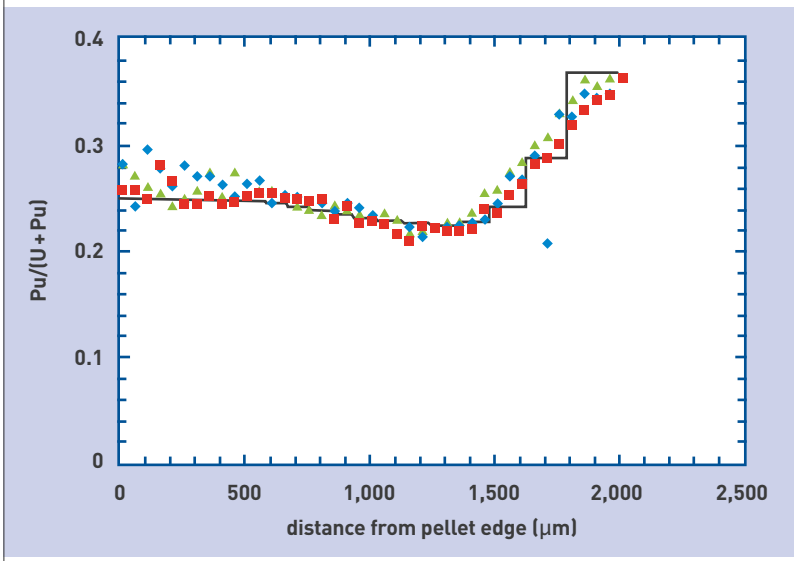



Figure 3.
**Castaing microprobe** measurement of plutonium radial redistribution in a pin that has undergone restructuring (main irradiation data: linear power at section plane: 370 W/cm at start of irradiation → 270 W/cm at end of irradiation; maximum burnup: 15.8% FIMA; nominal cladding temperature: 630 °C; irradiation time: 983 EFPD; cladding damage: 117.5 dpa). This examination was carried out at the Active Fuel Investigation Laboratory (LECA, CEA/Cadarache).

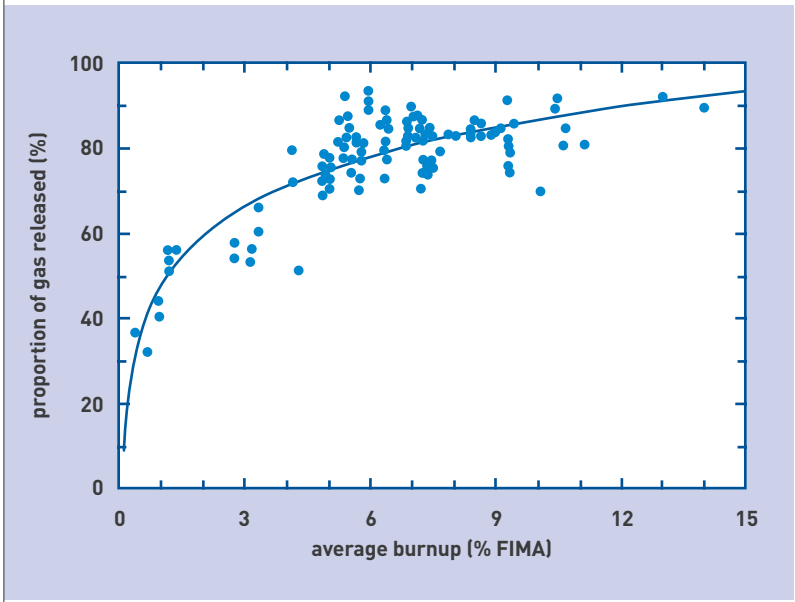



Figure 4.
Evolution of gaseous fission product release rate (volume released into pin free spaces/volume generated by fission), as a function of average burnup (taking into account axial neutron flux factor) in pins irradiated in Phénix. The dispersion of values around the 6% FIMA mark is due to the presence, or absence of cladding diametral strain, due to swelling. Cladding strain results in reopening the gap between fuel and cladding, and, consequently, exacerbating oxide thermics, and thus gaseous fission product release.

(3) Oxygen potential: the oxygen pressure, at a given temperature, that sustains a state of equilibrium over a solid, or liquid oxide, indicating its oxidation potential, i.e. its capacity to strip **electrons** from an **atom**, or **ion**.

(4) Getters: oxygen "traps," which may be used to restrict the rise in oxygen potential, in irradiated fuel. Consisting of refractory, highly reducing metals, or ceramics, such oxygen buffers come in the form of sintered disks, generally positioned at the top end of the fissile column, where the risk of cladding corrosion is highest.

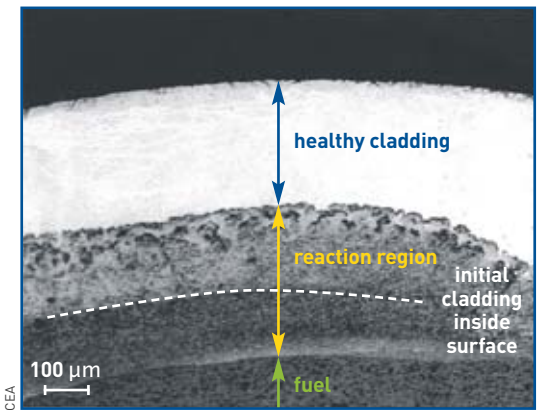
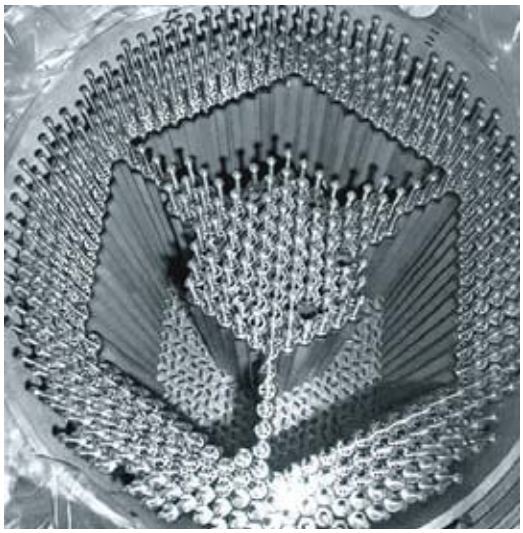



Figure 5.
Optical micrograph showing internal corrosion on the cladding of a highly irradiated Phénix fuel pin. Depth of corrosion, in this case, is quite large (~ 200 µm, i.e. 40% initial cladding thickness), affecting almost the entire inside wall at this location (main irradiation data: maximum burnup: 16.3% FIMA; nominal cladding temperature: 640 °C; irradiation time: 1,122 EFPD; cladding damage: 152 dpa). This examination was carried out at the Active Fuel Investigation Laboratory (LECA, CEA/Cadarache).



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Layout of the fuel assemblies in the Rapsodie reactor core. Built at Cadarache, this experimental reactor, belonging to the sodium-cooled fast reactor line, was operated from 1967 to 1983. Numerous experiment programs were carried out in this reactor, to develop the technology line.

tors, have made it possible to narrow down the thermic, physical–chemical, and mechanical conditions making for the initial defect developing into an open cladding failure, and thus to suggest to the operator an adequate management mode for failing assemblies.

## Operational feedback regarding carbide, nitride, and metallic fuels

While the mixed oxide's satisfactory behavior has been proven up to very high burnup values (15–20% FIMA), and during the various steps in the **cycle** (fabrication, storage, and reprocessing), it does not carry advantages only. Its main drawbacks – to wit, low heavy-atom density, and poor sodium compatibility – have led the various R&D organizations, CEA in particular, to look to fuels potentially more advantageous than oxide fuel: **carbide** and **nitride** ceramics, and **metal alloys** (see Table).

These three fuel materials afford the benefits of higher density, allowing higher **internal breeding gain** (increased cycle duration, and greater Pu yield), and full sodium compatibility, thus avoiding inconvenient reactor stoppages, to discharge a cladding-failure assembly.

Carbide and nitride ceramic fuels further afford the advantage of exhibiting better thermal conductivity than the oxide, allowing lower operating temperatures, and a greatly increased margin with respect to melting point. However, these so-called "cool" fuels do exhibit swelling that is more severe, by a factor 2–3, than for oxide. Moreover, release of gaseous fission products, while remaining lower than is the case with oxide, may nonetheless reach significant levels (> 50%) at high burnup (10% FIMA). The high swelling rate, combined with the lower viscoplasticity(5) exhibited by these fuels (lower by a factor 10, compared to the oxide's), sets up strong mechanical interaction between the fuel and cladding, once the as-fabricated gap has been taken up. The experience gained on this process shows it is imperative to bring down, at the fabrication step, the packing density(6) of such fuels (this must be lower than 75% theoretical density), if it is intended to achieve, without incurring undue risks of cladding failure, the same burnup values as with the oxide. Further, compared to the experience accumulated with the oxide, carbide, and – even more emphatically – nitride do leave, currently, far more unanswered questions as to their actual potential,

(5) Viscoplasticity: a reaction to **stress** exhibited by fuel, which behaves as though it consisted of a plastic solid, and a viscous fluid, only flowing above a certain threshold, flow being governed by parameters such as temperature, fission density, load, or applied load gradient.

(6) A fuel pin's packing density is equal to the ratio of fuel pellet cross-section, over cladding internal cross-section, multiplied by fuel density, expressed as a percentage of theoretical density.

C. Morgan/INL

The EBR-II (Experimental Breeder Reactor-II) sodium-cooled fast reactor took over from EBR-I in 1964. A large number of fuels, other than oxides (carbides, and metal alloys), and structural materials (claddings, hexagonal wrappers) have been tested in this reactor.

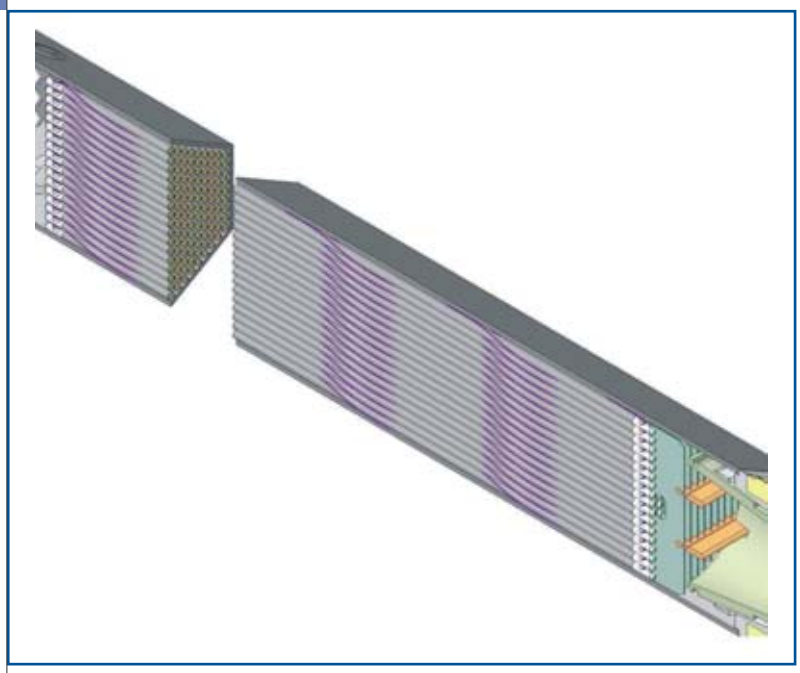

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Figure 6.
Exploded view of the central region of a standard Phénix fuel assembly, showing the "bundle" of 217 pins, each fitted with its own spacer wire.

as fuels for the **SFR** of the future (see Box, *The six concepts selected by the Gen IV Forum*, p. 6).
As regards metallic fuel, after the disappointing initial attempts, as recorded above, the US teams at **Argonne National Laboratory** (**ANL**) conducted a significant R&D effort, up to the early 1990s. Extensive investigations of stabilized fuel alloys, carried out with the aid of the EBR-II experimental reactor, enabled them to achieve, ultimately, using alloys of the U–Pu–10Zr (10% zirconium) type, performance comparable to that for oxide fuel, in terms of burnup. Nor was this experimentation with metallic fuel held as a hole-and-corner affair. It was grounded on a substantial base of some 150,000 pins, irradiated in EBR-II. To be sure, this type of fuel also carries a number of drawbacks, the main ones being the risk of chemical reaction with the cladding (with possible formation of a eutectic), bringing down the maximum allowable **coolant** temperature, under nominal operating conditions (by about – 40 °C, with respect to ceramic fuels); and a cycle relying on an electrochemical process, deemed non**proliferating** by US researchers, and suited to onsite operation – though one that has yet to be demonstrated on an industrial scale (plutonium and **minor actinide recycling**, waste management)…

## Fuel definition

Meeting the specifications for the fourth-generation SFR will entail strong constraints on core design, and design of the assemblies and fuel elements. The various avenues to achieve core improvements being far from complementary, the issue will be that of finding, among the many options, the best trade-off (see *Sodium-cooled fast reactors of the future*, p. 24).
Savings in terms of resources, and nonproliferation of nuclear materials entail a conversion of uranium into plutonium, inside the fuel itself. This can only be achieved by using a sufficient proportion of **fertile** uranium 238 to make up for the fission of the plutonium initially brought in.[7] In effect, this type of core, involving as it does a slightly positive internal breeding gain, will necessarily exhibit poorer performance than first-generation SFR cores; however, such lower specific power is altogether imperative, to ensure consistency with the aim of bringing down the **sodium void coefficient**, by way of an increased fuel-to-sodium volume ratio.
To limit **pressure drop** across such a core, designers are looking to a core featuring "bundle" fuel assemblies, comprising fewer pins, of a larger diameter than those used in Phénix-type assemblies. Each of these is fitted with a small-diameter spacer wire (see Figure 6). Obviously, the thermal–mechanical behavior under irradiation of such an assembly will have to be **modeled**, to assess its ability to meet the lifetime criterion. Further, its behavior must be established in situations departing from nominal, in particular with respect to the risk of a blockage of cooling channels, in the event of excessive distortions in the pins and their associated wires.

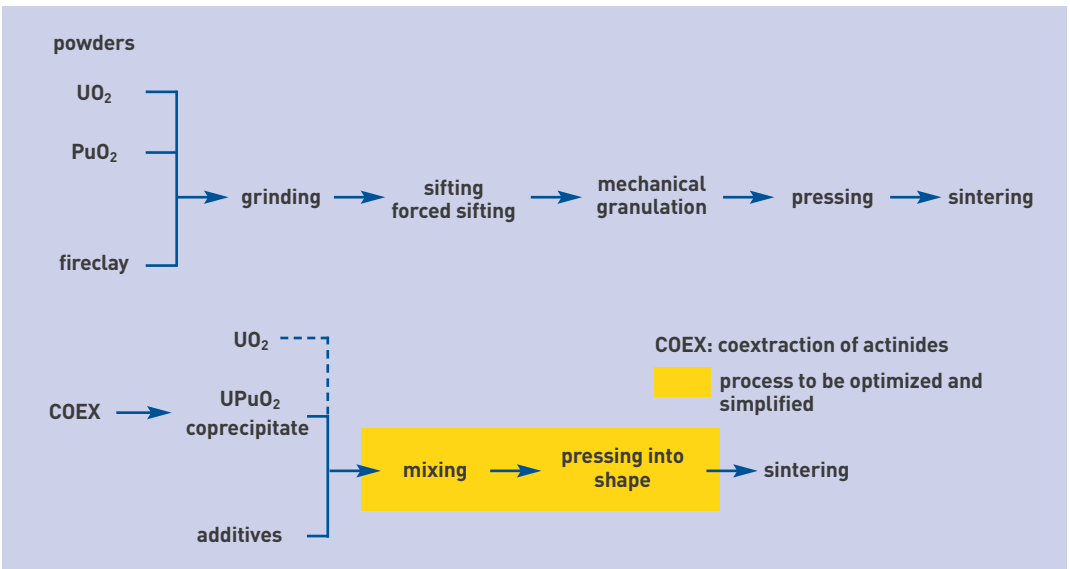


Figure 7.
Top, the steps in the COCA process, a conventional SFR mixed-oxide pellet fabrication process. Bottom, an innovative alternative, involving a $(U,Pu)O_2$ powder obtained by coprecipitation. Development of this process, which could facilitate use of minor-actinide-bearing fuels, is under way at CEA's Fuel Design Department (DEC: Département d'études des combustibles).

As regards the fuel itself, the preferred material option remains the oxide one, at any rate for the initial cores. Operational feedback has shown that the oxide, and – to a lesser extent – the metal option are the only fuel materials that will allow vouchsafing, presently, high burnup.

Blending minor actinides into the fuel, to be diluted in every assembly (**homogeneous-path transmutation**), entails a rethinking of the fabrication process for such a fuel. The COCA process,(8) involving the joint grinding of uranium dioxide ($UO_2$) and plutonium dioxide ($PuO_2$) powders, previously used for the fabrication of fuels for the SFR reactor line, might be supplanted by an innovative process, involving powders obtained by co**precipitation** of a uranium–plutonium–minor actinide mixture, as yielded by **spent fuel** (see Figure 7). This process would allow curtailing the handling of minor-actinide-bearing powders, and reduce contamination in the fabrication workshop. These design studies have, as yet, only reached the R&D stage. An irradiation program will doubtless be required, to validate the most promising options. In any event, owing to the very-high-temperature operating conditions for an SFR oxide fuel, having the effect of swiftly obliterating initial differences in the microstructure, researchers are anticipating, at first blush, but little impact on the irradiation behavior of this novel fuel, as compared to the operational feedback from the COCA process.

Using large-diameter pins, to increase the proportion, in volume terms, of fuel inside the core, and lower the pressure drop – designers are currently looking to a fuel pellet diameter of around 10 mm(9) – will, in the preferred option, involve fuel exhibiting an annular geometry, to allow for an adequate margin, with respect to fuel melting temperature, particularly at start of life, when the gap between pellet and cladding is still wide open (see Figure 8). Employment of pellets of this type in experimental irradiations in Phénix has not evidenced any significantly different behavior from that of the initially solid, standard fuel(10) (see Figure 9). Nevertheless, it will be indispensable to check the long-term behavior of such a fuel element. For instance, the association of high fuel thermics, and a larger volume of fuel than for a Phénix-type pin means a potentially stronger risk may be anticipated, of internal corrosion at high burnup values, for the larger-diameter pin.

The specifications for a fourth-generation SFR core thus entail a number of consequences, as regards definition of the "fuel" – in the broader sense. Recommendations concerning the core, assemblies, and fuel pins, along with the type of fuel, and associated fabrication process, are expected to be made by 2009. Mechanical and chemical interactions with structural, cladding, and hexagonal wrapper materials have barely been gone into. The materials likely to meet the core specifications for the SFR of the future are surveyed in Chapter III of the present issue.

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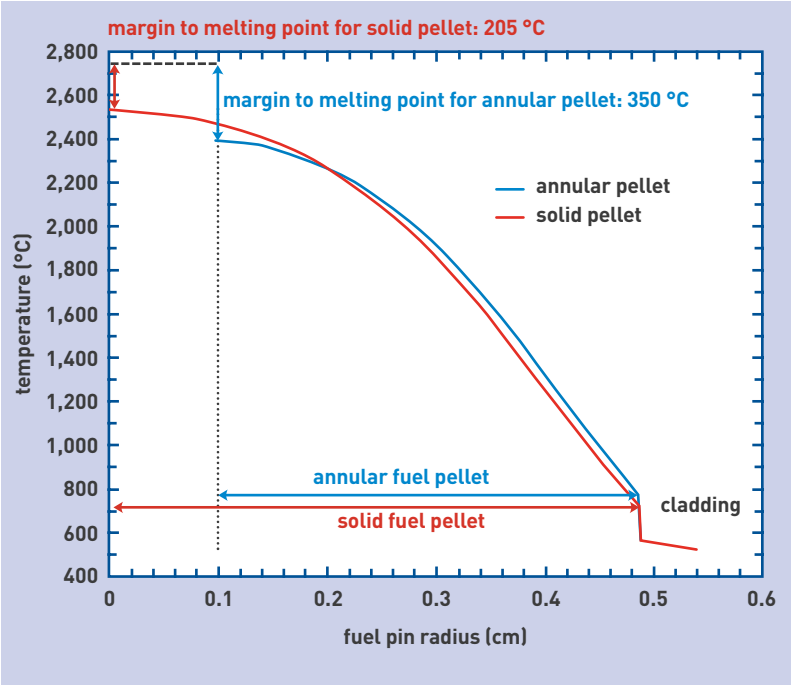


Figure 8.
Results of a computation carried out using the Germinal 1.4 **software**, modeling the irradiation behavior of an SFR mixed-oxide fuel pin. For a large-diameter pin (pellet diameter = 9.5 mm), comparison of radial temperature profiles for solid-pellet, and annular-pellet fuels, under similar operating conditions for both fuels. A significantly enhanced margin may be achieved at start of life, with respect to fuel melting point, for the annular concept, compared to the solid pellet.

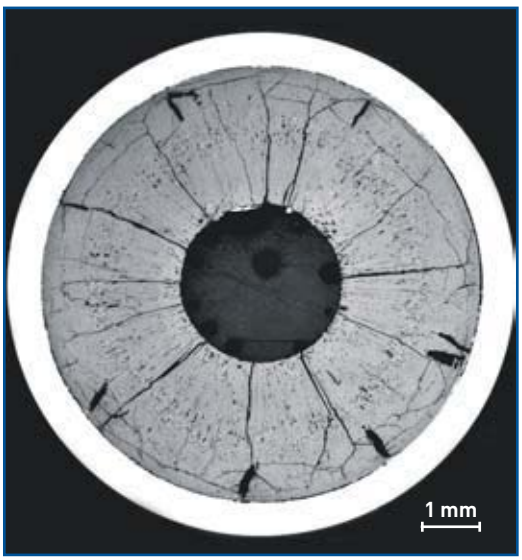


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Figure 9.
Transverse metallographic section of an annular, Superphénix-geometry fuel pin, irradiated in Phénix, showing that fuel microstructural evolution is similar to that for an initially solid fuel (main irradiation data: linear power at the section plane: 555 W/cm at start of irradiation → 585 W/cm at end of irradiation; maximum burnup: 9% FIMA; nominal cladding temperature: 620 °C; irradiation time: 487 EFPD; cladding damage: 73.5 dpa). This examination was carried out at the Active Fuel Investigation Laboratory (LECA, CEA/Cadarache).

(7) Mean core enrichment, as currently envisaged, would stand at around 15% Pu. **Neutron captures** in $^{238}U$ yield, by way of successive **$\beta^-$ decays**, $^{239}Pu$.

(8) COCA process: a **MOX** fuel fabrication process, developed by COGEMA (now trading as **Areva NC**) at Cadarache, whereby all of the $UO_2$ and $PuO_2$ powders are ground together, resulting in very high mixture homogeneity.

(9) By way of comparison, fuel pellet diameter stands, respectively, at 5.42 mm for the Phénix reactor, 6.94 mm for the erstwhile European **EFR** project, 7.14 mm for Superphénix fuel, and 8.19 mm for a standard EDF PWR fuel rod.

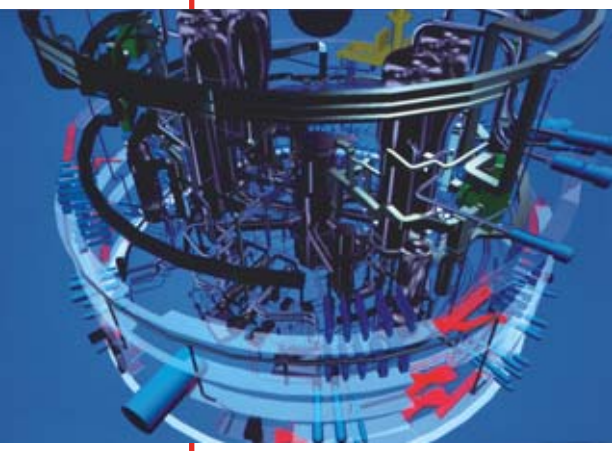
(10) It should be remembered that initially solid fuel pellets undergo restructuring, under the effects of the high thermics in the oxide, thus turning into – more or less rapidly, and over part of the fissile column – annular pellets!

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 $\eta$ 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.

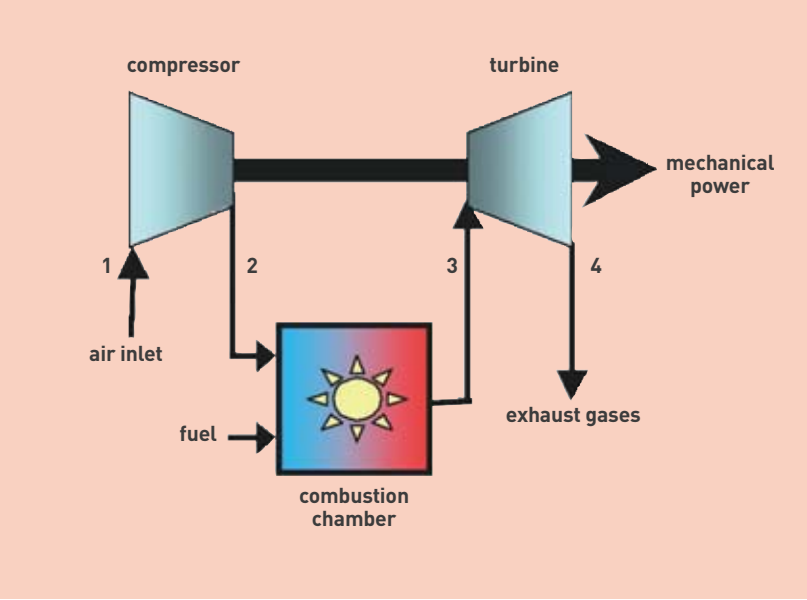



Figure.
Brayton cycle, as implemented in an open-cycle gas turbine.

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.

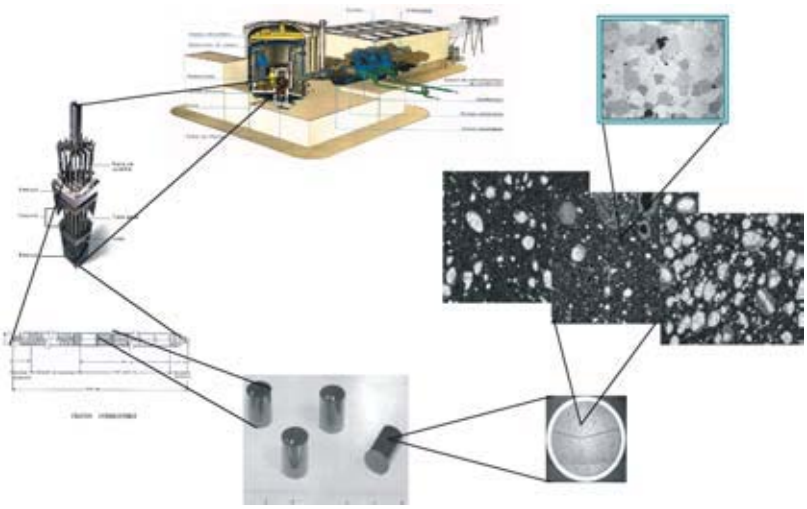


Figure.
Improving nuclear fuel reliability, and cost-effectiveness calls for finescale modeling of that fuel, through a multiscale approach, from reactor to fuel microstructure (in this instance, MOX fuel). Microstructural characteristics (porosity, cluster size and distribution, grain size...) have 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.

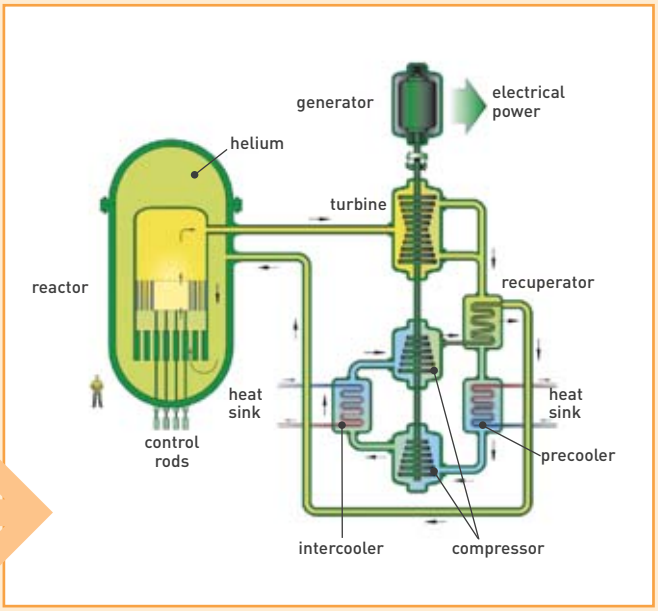
# The six concepts selected by the Gen IV Forum

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

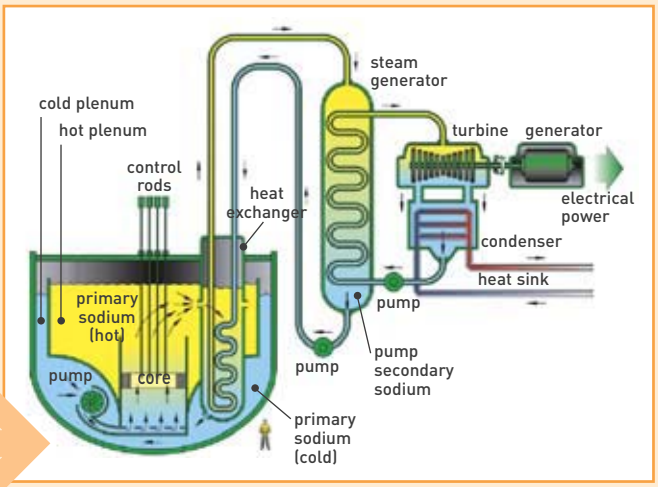
The gas-cooled fast reactor system (GFR) is a high-temperature, gas-cooled (helium-cooled as a rule), fast-neutron reactor allowing actinide recycle (homogeneous, or heterogeneous), while sustaining a breeding capability greater than unity. The reference concept is a helium-cooled, direct- or indirect-cycle 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.



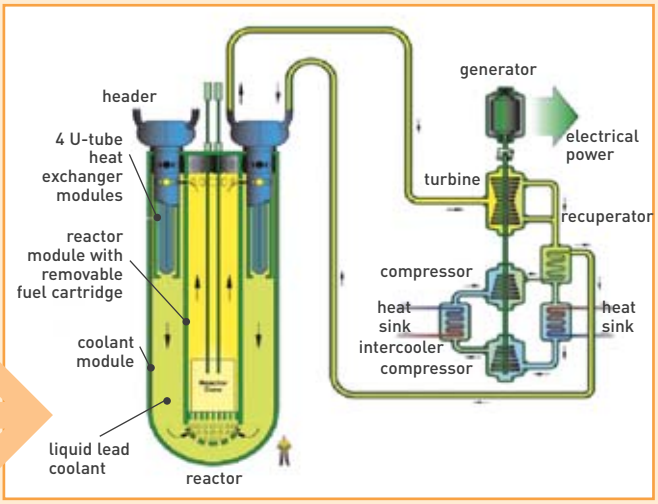

## Le SFR

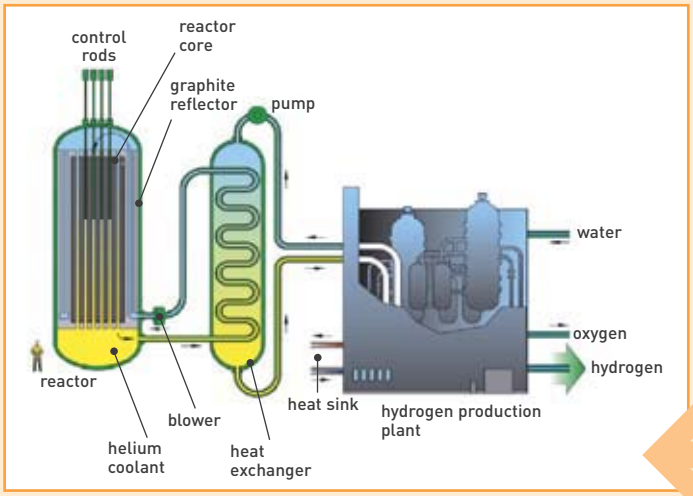
The sodium-cooled fast reactor system (SFR) is a liquid-sodium-cooled, fast-neutron reactor, associated to a closed cycle, allowing full actinide recycle, and plutonium breeding. Owing to its breeding of fissile material, this type of reactor may operate for highly extended periods without requiring 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

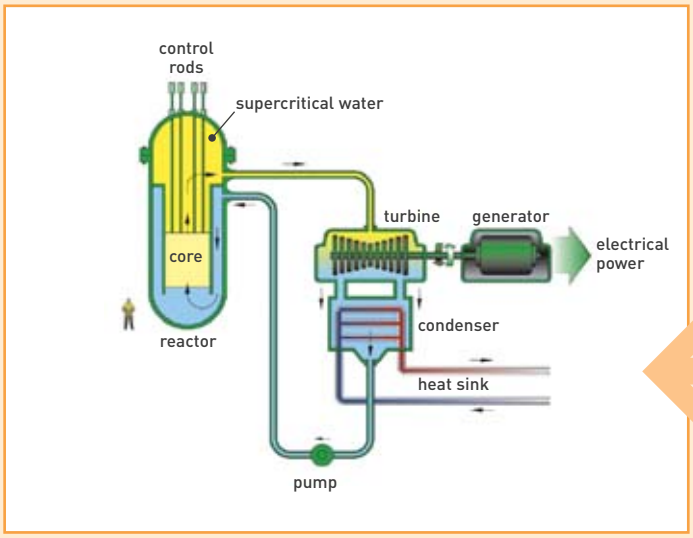
The lead-cooled fast reactor system (LFR) is a lead- (or lead–bismuth alloy-) cooled, fast-neutron reactor, associated to a closed fuel cycle, allowing optimum uranium utilization. A number of reference systems have been selected. Unit power ranges from the 50–100 MWe bracket, for so-called battery concepts, 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.





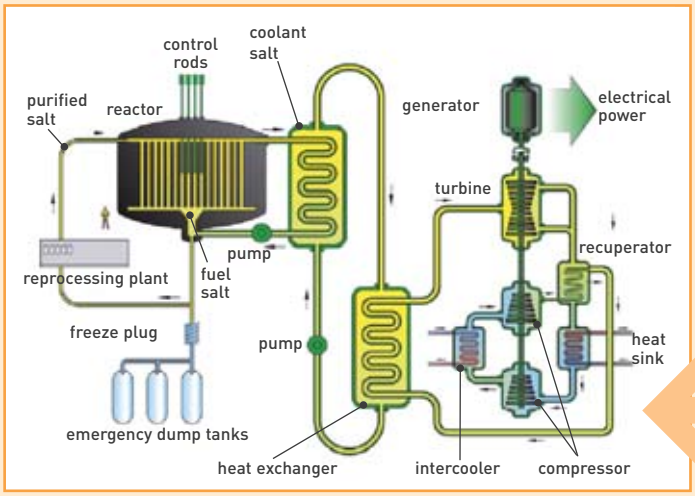

## VHTR

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.