What **fuel** for **GFRs?**

The challenges the fuel for a gas-cooled fast reactor must meet are on a par with the expectations vested in a technology that aims to combine the advantages of a fast neutron spectrum, ensuring optimum materials utilization, and those of high temperature, as the key to high energy efficiencies. While the "carbide" fuel option has been selected for the initial demonstration irradiations, the "nitride" option, which still requires further work, remains open for future developments.

P. Dumas/CEA

The TITANS line, at LEFCA (Laboratoires d'études et de fabrications expérimentales de combustibles nucléaires avancés: Advanced Nuclear Fuels Design and Experimental Fabrication Laboratories), at CEA/Cadarache, set up to carry out fabrication of fourth-generation reactor fuels. Actinide carbides, in particular, will be fabricated here, and a number of characterizations of these fuels will be carried out in the facility.

The concept of the gas-**cooled** reactor using **fast neutrons**, or fast reactor (**GFR**), stands both as an alternative to the **sodium**-cooled fast reactor (**SFR**), and as a "sustainable" variant of the very-high-temperature, gas-cooled, **thermal-neutron** reactor (**VHTR**) (see Box, *The six concepts selected by the Gen IV Forum*, p. 6). This **fourth-generation system** aims to combine the advantage afforded by use of a **fast spectrum**, as regards ensuring value-added, energy-yielding use for **natural uranium** resources, and the benefit of high temperatures, by employing a coolant that is inert, and transparent to **neutrons**: **helium**, that opens up the prospect of a very-high-efficiency **energy conversion cycle**.

Owing to the poor thermal energy removal characteristics exhibited by pressurized helium, compared to a liquid metal, a major fraction of the GFR **core**'s volume needs must be devoted to the coolant. Further, the purpose of that core being to achieve a slightly positive **internal breeding gain**, a condition acting as a restraint on **plutonium** enrichment, this makes the use inescapable, of a **fissile** material exhibiting a high **heavy-atom** density, and one that furthermore must be refractory. **Fuel** materials best meeting such specifications are **actinide carbides**, and **nitrides** (see Table, in *What fuel for SFRs?* p. 33).

Fuel specifications

The current reference boiler is based on a core delivering 1,200 **MWe** power, i.e. having a thermal power of some 2,400 **MWth**, for an efficiency estimated as standing at 50%. Neutronics and thermal–aeraulics criteria entail that the three main core entities must comply with the following volume fractions: 40% coolant; 38% **assembly** structures, and fuel element first **barrier**; 22% for the fissile phase itself.

The system considered being a high-efficiency system, coolant temperature at the core outlet will be very high, around 850 °C. To ensure satisfactory economic competitiveness, assembly in-core time will have to be optimized, to achieve, as a minimum, **burnup** of 5% **FIMA**,[1] with a more ambitious target set at 10% FIMA. Further, in order to downsize the **HLW–LL waste** stream in the **cycle**,

this fuel – in like manner to SFR fuel – must have the ability to incorporate **minor actinides**, up to a content of a few percent.
Structural materials involved in the makeup of fuel element and assembly components must have the ability to withstand temperatures reaching some 1,000 °C in nominal operating conditions, and liable to rise to 1,600 °C during accident **transients**, while concurrently proving capable of guaranteeing, in all circumstances, and through the item's entire lifetime, containment of the fissile material, and **fission products**, as well as core mechanical resistance. **Composite ceramic** materials would appear, at first blush, to be the only suitable materials, to meet such specifications (see *Putting the properties of nonmetallic materials to advantage*, p. 78).

Two fuel element concepts under scrutiny

Logically, GFR fuel features should stem straightforwardly from those of the fuel for the high-temperature reactor (HTR) family. These thermal-neutron reactors, using **graphite** as **moderator**, and helium as coolant, employ fuel in the form of kernels, of less than millimeter size, with coatings of carbon, and ceramic. These particles, dispersed in a graphite **matrix**, allow the construction of refractory, highly robust cores (see *High-temperature reactors: a recent past, a near future*, p. 51).
The initial idea was thus to seek to adapt this "magic particle," to meet the neutronic, and thermic constraints of the GFR core. Unfortunately, functional analysis of the fuel(2) that would allow specifications for the core to be met resulted in ruling out a number of concepts. All concepts based on a matrix dispersion of particles were ruled out, as being unable to meet the set conditions, in terms of **power density**, maximum operating temperature, or thermal–aeraulic behavior.
Ultimately, just two concepts to date show some potential, bearing in mind the stringent constraints imposed by GFR operating specifications. The first concept, of the *macrostructured* ***plate*** type, combines the advantages of a plane geometry, in terms of the optimization of thermal exchanges with the coolant, and the benefit of a honeycomb cell internal structure, with regard to element mechanical strength, and close confinement of fission products. The second concept, of the *cylindrical* ***pin*** type, taking its cue to some extent from **light-water reactor** (**LWR**) and SFR fuel elements, will nevertheless require some adjustment, to meet GFR core constraints. These two concepts carry a number of benefits, and drawbacks (see Table).

The cylindrical pin

As far as the pin is concerned, the lifetime limiting process is – once the as-fabricated gap has been taken up – a mechanical interaction between fuel and **cladding**, of exceptional amplitude. Thus, the satisfactory behavior of **pressurized-water reactor** (**PWR**) fuel pins (**rods**), in such conditions, is due to the outstanding mechanical properties under **irradiation** exhibited by the cladding material, throughout the irradiation phase subsequent to the gap being filled.(3)
As regards the SFR **pin**, the absence of any sizeable mechanical interaction is rather the outcome of a balanced tradeoff between a "hot" – and thus highly malleable – fuel, able to accommodate some of the distortion within its own internal voids, and cladding which, through irradiation **swelling** and **creep** mechanisms, can allow a substantial increase in pin internal volume to take place.
In the case of a GFR pin, there can be no question whatsoever of relying on any "benevolent" behavior in the various materials involved in its construction.

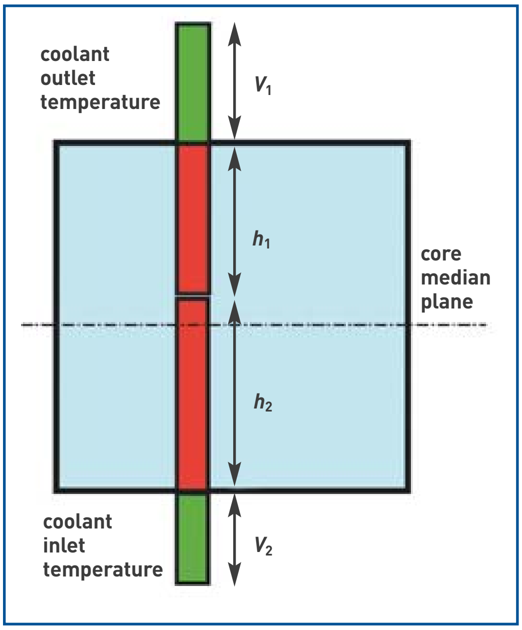


Figure 1.
Schematic of pin-type fuel elements in a GFR.
The fissile columns of the two, lower and upper, half pins are not of equal length, in order to equalize the pressure of the gaseous fission products inside the lower, and upper plenums, for which operating temperatures are the coolant temperatures at core inlet, and outlet, respectively.

Table.
Relative advantages (+) and deficiencies (-) or (--) of concepts of the macrostructured plate and cylindrical pin type.

criteria		plate	pin
mastery of fuel thermics		+	-
impact of a substantial release of gaseous fission products, and helium		-	+
mechanical resistance of fuel element to loading:	with applied **stress**	-	+
	with applied deformation	+	--

(1) In the present case, this refers to average burnup, across the entire core.

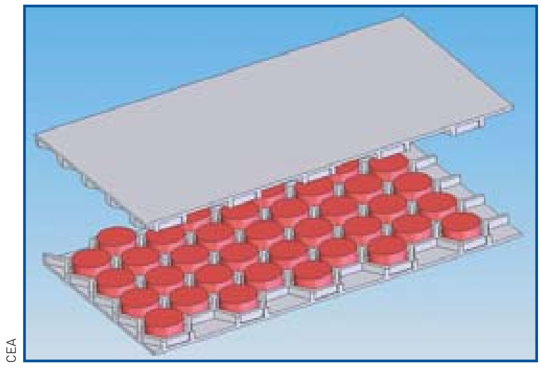
(2) The functions that must be ensured are: energy generation, meeting the targets set in terms **power density**; transfer of the heat generated to the coolant, involving minimization of thermal resistance, and reductions in **pressure drop**; containment of fission products, through minimization of containment barrier permeability.

(3) PWR fuel rod claddings are nonetheless liable to be subjected to very high stresses, during power transients in which cladding creep cannot occur to an extent sufficient to ensure stress relief.

The somewhat "cool" fuel does not readily creep, while it swells at at least twice the rate of an **oxide fuel**. The cladding, made as it is of a ceramic material, exhibits neither plastic nor creep **ductility**, nor fracture **toughness**, sufficient to allow it to accommodate such interaction without sustaining unacceptable damage.
Thinking as to the design of a GFR pin must therefore address that pin's ability to function out of mechanical interaction conditions, for as long as possible. This comes down to finding the best trade-off between an ensemble of fabrication and operating parameters, such as diameter, initial fuel–cladding gap, as-fabricated fuel porosity, maximum **linear power**… Further, preliminary dimensioning studies have shown that using cladding made of ceramic materials restricts to one meter the maximum permissible length for the cladding tube. Thus, designers are currently looking to a structure comprising two "half pins" joined end on end, featuring **plenums**, to accommodate gaseous fission products, extending above, and below the core (see Figure 1).

The macrostructured plate: an innovative fuel element

Of the two fuel elements potentially able to provide a solution equal to meeting the demands of operation at very high temperatures (with cladding temperature higher than 1,000 °C), and high fast-neutron **fluences**, as required by GFR systems, only the macrostructured plate element stands as a conceptually original solution.
These fuel plates are intended to be inserted, in three stacks, set at 120 degrees from each other, into the hexagonal-section assembly casing (see *Gas-cooled fast reactors*, p. 38). The plate's fuel core consists of a macrostructured **cercer**, comprising the ceramic matrix, exhibiting a honeycomb structure; and the fissile phase, in the form of cylindrical compacts, positioned in each cell of the honeycomb. This fissile core is clad on either side by two plates, also made of a ceramic material, ensuring cell closure, and thermal exchange with the gas coolant (see Figure 2). Preliminary dimensioning of these plates currently results in a thickness lower than 10 mm, width of 120 mm, and length of about 250 mm.
What are the theoretical advantages afforded by this concept? Heat transfer, from power source to coolant, is optimized, with minimal thermal resistance between fuel compact and cladding, and maximal exchange surface between cladding and coolant. Such good heat-exchange management, throughout irradiation, allows fuel temperature to be controlled, whatever the level reached, of gaseous fission product release into the cell (see Figure 3).
The presence, in every cell, of a space surrounding the fuel compact, due to the bringing together of cylindrical, and hexagonal geometries, allows gaseous fission products to collect. This space is so dimensioned as to limit any pressurization of these products, in all operating conditions (see Figure 4). The plate exhibits good mechanical strength, with respect to pressure loadings, and mechanical interactions between fuel and cladding. Interaction between fuel compact and cell is restricted to the sole dimension perpendicular to the plane of the plate, which makes it possible to minimize stresses, by accommodating the distortion imposed by fuel swelling, through cladding bending, and radial deformation of the compact, by way of an irradiation-induced creep mechanism. Further, using fuel compacts of a cambered, bulging shape allows gradual contact to take place, thus precluding stress concentration.



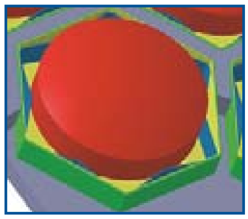


Figure 2.
Left, exploded view of a GFR macrostructured fuel plate; right, detail view of a cell. The fuel pellets are uniformly distributed in the hexagonal cells, made of a ceramic material, to optimize fissile material packing density. Each cell is leaktight, and independent from its neighbors. In the event of containment being breached, for one of the cells, release of gaseous fission products is restricted to that cell, and remains at a low level.

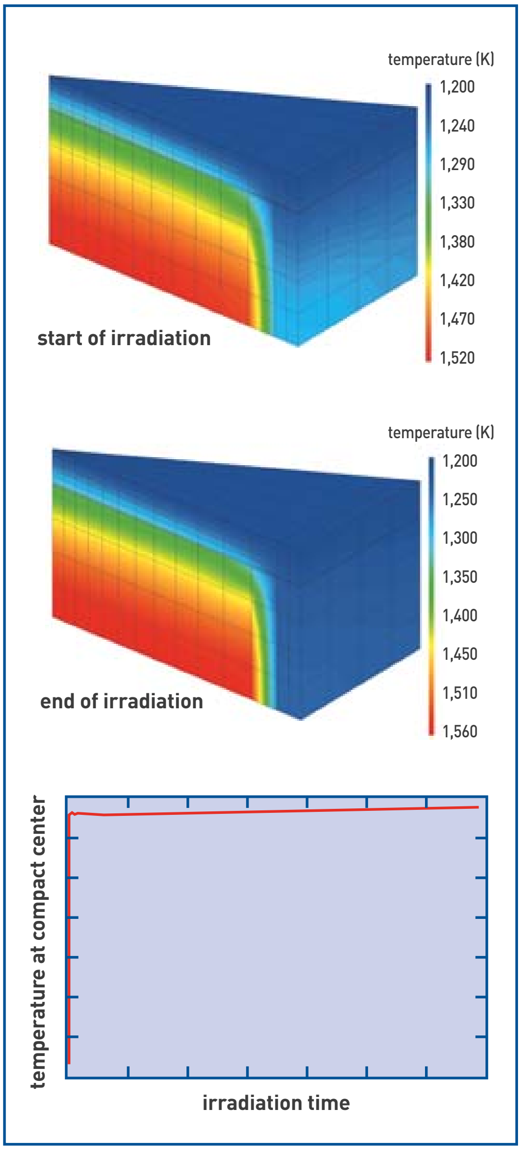


Figure 3.
Thermics **modeling** of a wedge-shaped cell segment in a macrostructured fuel plate. The 3D representations show the temperature distributions inside the fuel compact, in the gaps collecting gases, and in the plate structures. The schematic shows that temperature at compact center point stays relatively stable, throughout irradiation.

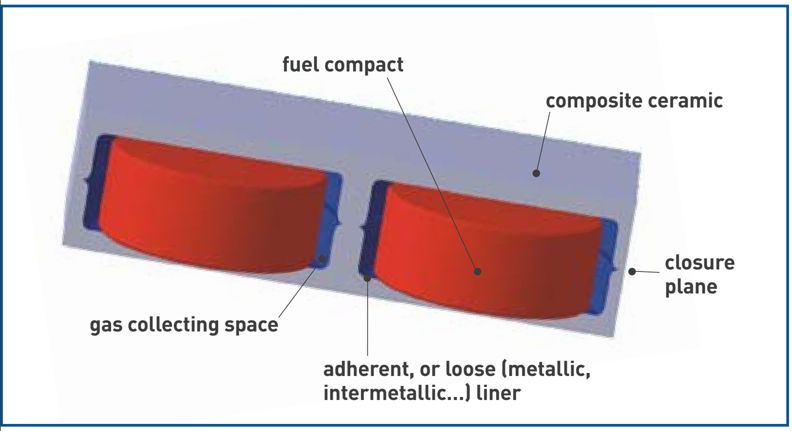


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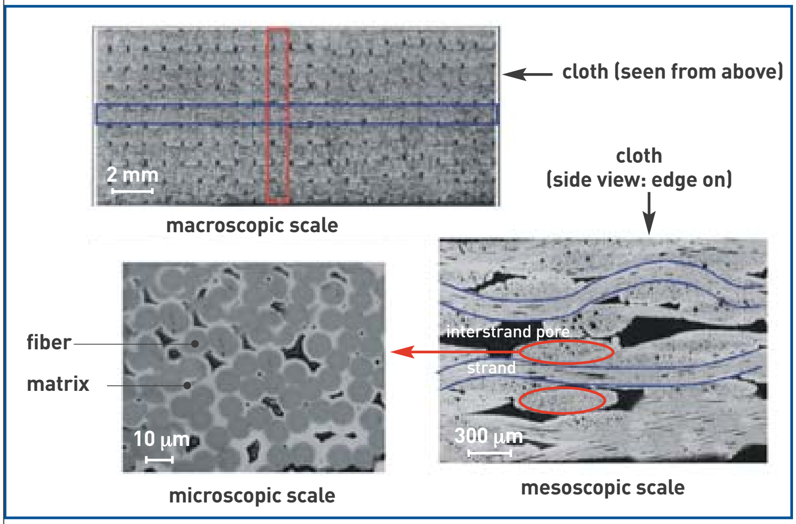
Figure 4.
Schematic of macrostructured plate cell design, showing cells holding their fuel compacts. Cell main dimensions, along with compact shape, were determined as the outcome of a parameter study.

Reinforced ceramic materials for the first barrier

Strict compliance with the specifications imperatively entails using ceramic materials for the plate structure. Indeed, metal **alloys** involving refractory metals (molybdenum [Mo],(4) niobium [Nb], tantalum [Ta], tungsten [W]), exhibiting as they do very high **neutron absorption** capacities, may not be allowed without incurring a considerable downturn in core performance. At the same time, use of monolithic ceramics, such as silicon carbide (SiC), cannot even begin to be entertained, owing to their very low toughness (< 5 MPa · $m^{1/2}$). As a result, it becomes necessary to turn to reinforced ceramic materials, these being, at first blush, the only materials liable to achieve a "consensus," between thermic, neutronic, and mechanical properties, and the demanding operating conditions for plate, and pin fuel elements.
Use of long-fiber-based composites may make a decisive contribution to the sought-for solution (see Figure 5). These materials would contribute higher distortion and damage tolerance,(5) compared to monolithic ceramics, due to the characteristics of their fiber reinforcement (see *Putting the properties of nonmetallic materials to advantage*, p. 78). Be that as it may, in the current state of know-how, they do not exhibit sufficient impermeability, to rare gases(6) (gaseous fission products generated during irradiation; and coolant helium), at any rate. As a result, it will prove indispensable, in order to vouchsafe the containment function assigned to the first barrier, to resort either to impervious coatings (adhering to the composite), or to (nonadherent) liners, consisting of a small thickness of refractory metal alloys. This requirement makes for singularly more complex fabrication for such fuel elements, compared with that of rods, or pins for the PWR, or SFR reactor lines, involving as these do metallic claddings ensuring both a total retention of fission products and mechanical integrity of fuel pin.
At the same time, the fissile phase equally requires major optimization, if it is to meet the operating requirements for the two GFR fuel element concepts.



CNRS-SNECMA-CEA-Bordeaux-I University

Figure 5.
Micrographs, obtained by scanning microscopy, of an SiC–SiC composite ceramic. This material consists of SiC fibers, coated with an interphase material, immersed in an SiC matrix. The composite is the material resulting from assembling the fibers into yarns (microscopic scale), which are then woven into plies (mesoscopic scale). Final assembling of the plies (macroscopic scale) forms the composite's reinforcement. The presence should be noted of porosities of considerable size. (Investigation carried out at the **LCTS** [Laboratoire des composites thermostructuraux: Thermostructural Composites Laboratory], UMR 5801, CNRS–SNECMA–CEA–Bordeaux-I University).

Between carbide and nitride, the GFR core is still unresolved

A comparative evaluation, in terms of core performance, safety, and level of industrialization was conducted, comparing actinide carbide, and nitride, as being the two most credible fuel materials for GFR use.

Performance

Unquestionably, carbide affords the best performance, particularly in terms of core volume, and fissile material immobilization. To achieve an equivalent performance level, nitride requires the inclusion, as part of the cycle, of **enrichment** to at least 50 at% nitrogen 15 (the ^{15}N isotope), a step for which a technico–economic assessment has yet to be carried out.(7) With respect to waste, analysis shows that, in any event, the **potential radiotoxicity** induced by carbon 14 (^{14}C), yielded during irradiation by (n,**p**) reactions in ^{14}N, remains much lower than that coming from 0.1% of the actinides

(4) Natural molybdenum, which has 7 **isotopes**, can be rid, through an enrichment process, of its more absorbent isotopes (> ^{96}Mo), thus yielding a "light" form of molybdenum, involving much smaller losses, in terms of neutronics. However, the technico–economic benefits from such a process, on an industrial scale, would appear, as of now, somewhat improbable.

(5) A composite material exhibits higher capabilities for deformation than a conventional ceramic, owing to its seeming ductility, though this is in fact due to gradual internal damage, occurring between the fibers composing it, and the matrix. However, this damage in the material also results, regrettably, in an increased permeability to gases.

(6) Rare gases: **elements** from column 18 in Mendeleev's periodic table (helium, neon, argon, krypton, xenon, and radon).

(7) Natural nitrogen contains less than 0.5 at% of isotope ^{15}N, exhibiting a much smaller **capture** capability than the majority isotope, ^{14}N. Some 6–10 **MSWU** of the ^{15}N isotope would have to be planned for, depending on the process, and targeted content value, for a 400-**TWh** GFR fleet (comparable to the current French reactor fleet).

present;(8) and that this volume of waste, generated as a result of the presence of an inert material, is of the same order of magnitude as that due to fission products. At the same time, with nitride, further generation of gas has to be considered: that of helium, from (n,α) reactions in ^{14}N, accounting for about 20% of gaseous fission product volume; and twice that amount of **tritium**. Such generation of helium has the direct consequence of bringing down fuel element burnup, all other things being equal. Tritium generation, in turn, has an impact in terms of circuit and coolant contamination.

Behavior under irradiation

The investigations, and irradiation programs conducted over the years 1960–90, particularly in the United States, in Europe, and, more recently, in Japan, have made it possible to evidence the importance of certain fabrication parameters for carbide and nitride fuels, as regards in-reactor behavior (swelling, release of gaseous fission products, mechanical and chemical interactions with cladding). Of late, investigations of irradiations, carried out in the Phénix (see *Phénix, a unique instrument in the area of fuel,* p. 98) and Joyo (Japan) SFRs, have confirmed that, for certain closed-system operating conditions at high temperatures, nitride could exhibit signs of dissociation of the (U,Pu)N phase, with concomitant plutonium-enriched metallic phase **precipitation** in the gap between fuel and cladding (see Figure 6). On the basis of such experimental feedback, and placing these findings in the context of GFR fuel element operation, recommendations regarding chemical content (impurity content, oxygen content in particular), and fissile phase microstructure (**grain** size, proportion and nature of as-fabricated porosity) have been made, to optimize the behavior under irradiation for such fuels.

(8) This value is consonant with the loss rate deemed to be acceptable, over the cycle as a whole.

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The Joyo sodium-cooled fast reactor, in Japan. Development of the GFR reactor line calls for irradiations in experimental fast reactors, such as Joyo, and Phénix, to test innovative fuels and materials.

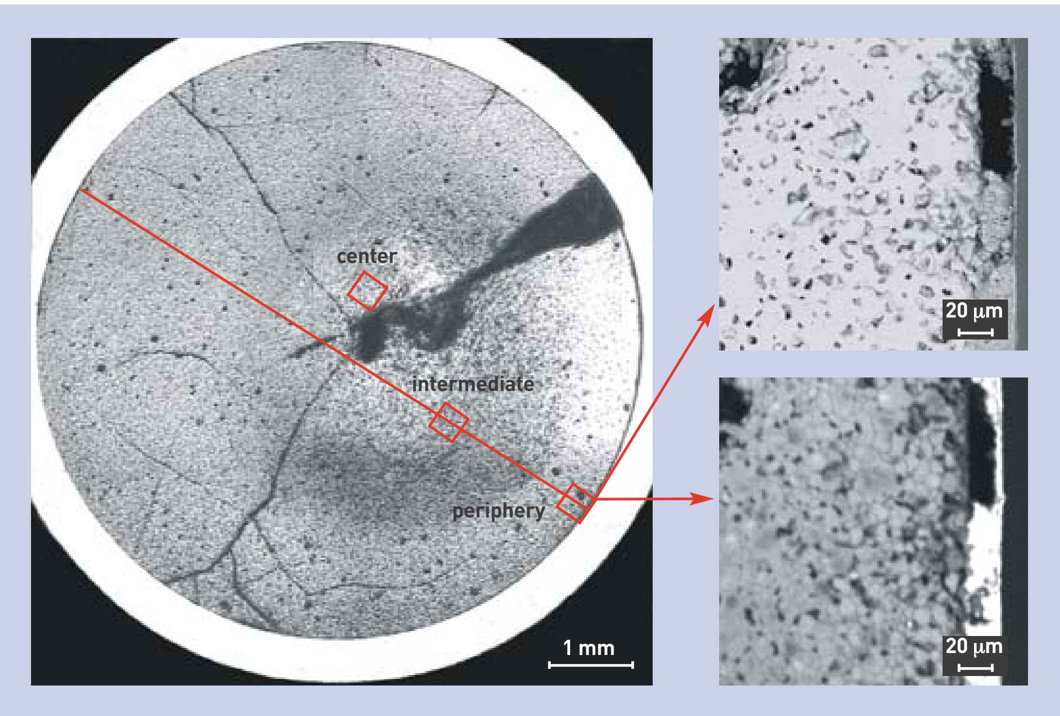

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Figure 6.
At right, **Castaing microprobe** mappings of the peripheral region of a section from a nitride fuel pin (shown at left), irradiated in Phénix (main irradiation data: maximum burnup: 5.8% FIMA; irradiation time: 362 **EFPD**; cladding damage: 47.5 **dpa**). This examination was carried out at the Active Fuel Investigation Laboratory (LECA: Laboratoire d'examen des combustibles actifs, CEA/Cadarache). Top: electronic, and bottom: X-ray imaging of Pu, clearly showing the presence of a plutonium-rich metallic phase at the pellet–cladding interface (lighter area in the image).

IGCAR

The FBTR (Fast Breeder Test Reactor), a sodium-cooled fast reactor, sited at Kalpakkam (India), delivers power of 40 MWth, using actinide carbide as fuel. The difficulties encountered in fabricating this fuel have led the Indian teams to switch away from this technology line, for their 500-MWe SFR, currently being built, at Kalpakkam also: the PFBR (Prototype Fast Breeder Reactor).

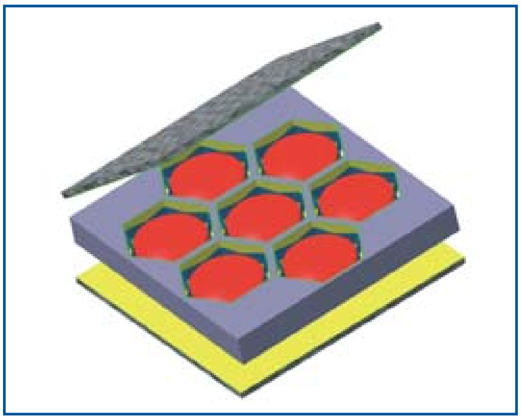

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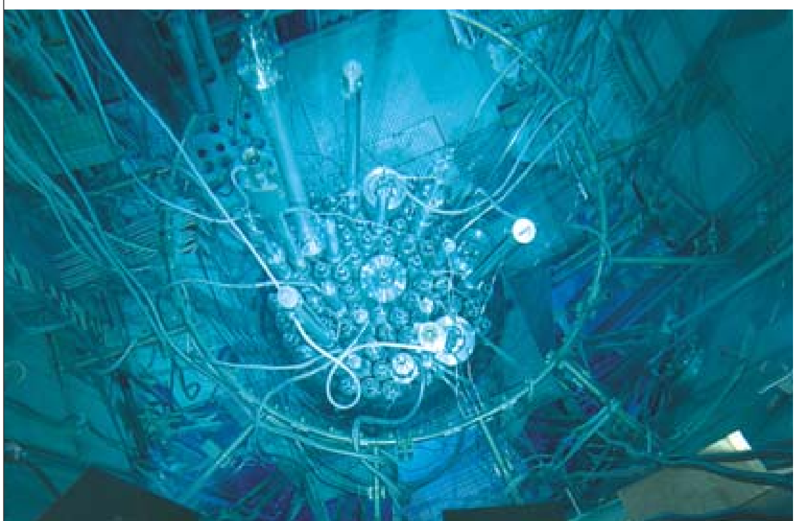
Figure 7.
"IRRDEMO" project, for the irradiation of fuel elements of the macrostructured miniplate type, in the BR2 experimental reactor, at Mol (Belgium).

Safety

Actinide carbide, and actinide nitride exhibit fairly similar properties, as regards ignition temperature, and kinetics. However, the carbide, owing to its morphological instability in the presence of traces of oxygen, or of water, has a much lower ignition threshold. For the same reasons, it is difficult to produce free of impurities, which impurities may have a deleterious impact on in-reactor behavior. Handling carbide fuel, during the fuel fabrication steps, demands perfect control of experimental conditions, which may prove extremely difficult to achieve in an industrial unit. In this respect, nitride exhibits greater tolerance. As regards in-reactor behavior, accident scenario analyses show that the actinide compound is involved only in the event of a breach of local containment, and that, in such conditions, its pyrophoric character (i.e. its ability to ignite spontaneously in air), whichever kind of actinide compound is involved, causes effects that are negligible, compared to the other processes that have to be taken into account in such a situation.



SCK-CEN

The BR2 experimental reactor, at Mol (Belgium), in which a number of demonstration irradiations are to be carried out, of the plate- and pin-type concepts.

Industrialization

Experimental feedback, as a whole, does not come down decisively on one side or the other, as regards carbide or nitride. One should take note, on the other hand, of India's switching away from the carbide technology line for its future SFR, owing, it would appear, to the many difficulties encountered in fabricating the fuel for the FBTR (Fast Breeder Test Reactor) reactor, at Kalpakkam. Compared to existing fabrication processes, a technological leap is required, in the carbide case, if pyrophoricity and impurity issues are to be overcome. As for nitride fuel, a technological leap is equally required, whether it be to achieve adequate ^{15}N enrichment, at a reasonable cost, or to trap the ^{14}C.

The options remain open

The final choice, between carbide and nitride, to provide the fissile phase for the fuel element in an industrial GFR, has not yet been decided. In particular, the "ease" with which the planned introduction of minor actinides may be effected, even in small amounts, into the fissile compound may play a part in material selection. Presently, on the one hand, for the purposes of viability irradiations (FUTURIX–Concepts irradiation in Phénix, in 2007–9), and, on the other hand, for demonstration irradiations of the plate- and pin-type concepts, currently at the definition stage, and initial results from which are expected by 2012 (see Figure 7), the carbide fuel option, carrying as it does fewer uncertainties than the nitride option, is the preferred one.

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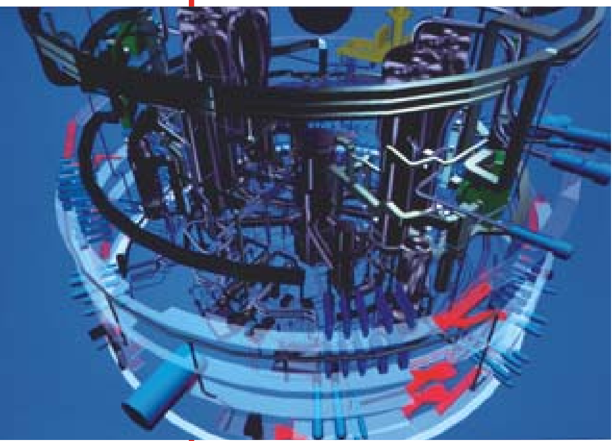
The components of a nuclear system

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

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

FOCUS (Cond't) B

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.

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 ($1 \rightarrow 2$); burning the air–fuel mix inside the combustion chamber ($2 \rightarrow 3$); and allowing the hot gases to expand inside a turbine ($3 \rightarrow 4$). 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.

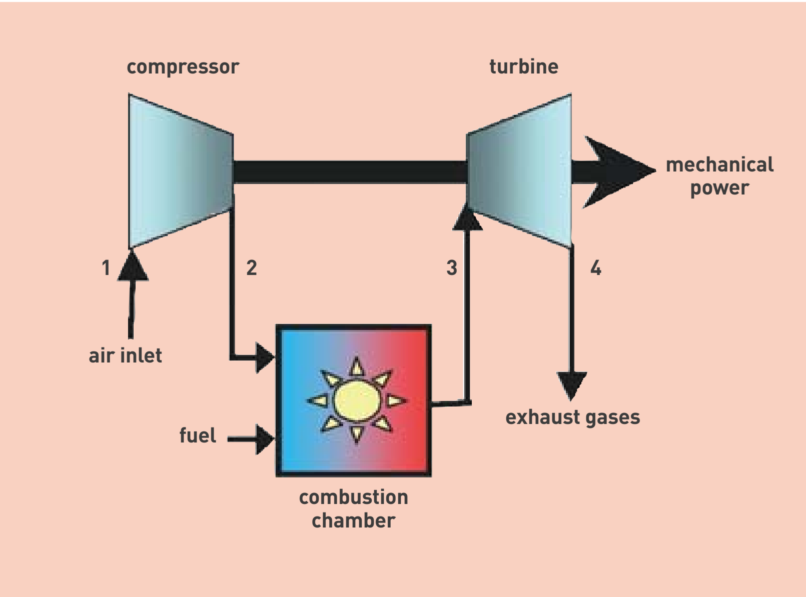


Figure.
Brayton cycle, as implemented in an open-cycle gas turbine.

FOCUS D

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.

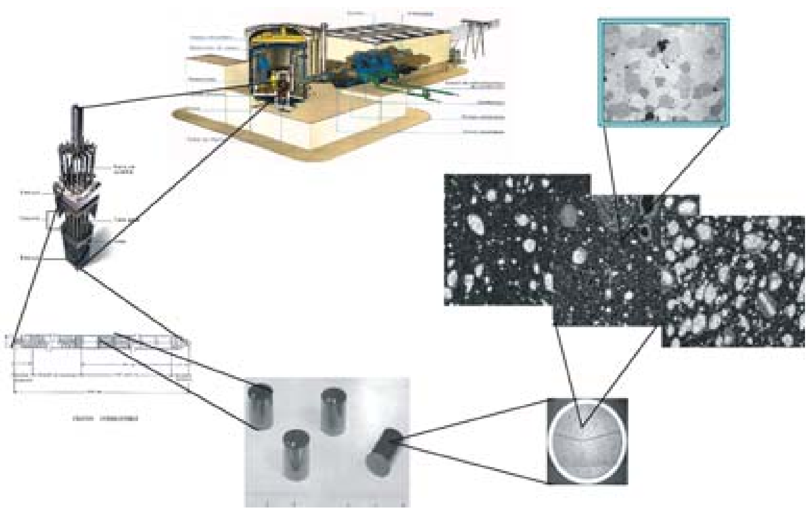


Figure.
Improving nuclear fuel reliability, and cost-effectiveness calls for finescale modeling of that fuel, through a multiscale approach, from reactor to fuel microstructure (in this instance, MOX fuel). Microstructural characteristics (porosity, cluster size and distribution, grain size…) have a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

FOCUS (Cond't) E

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.

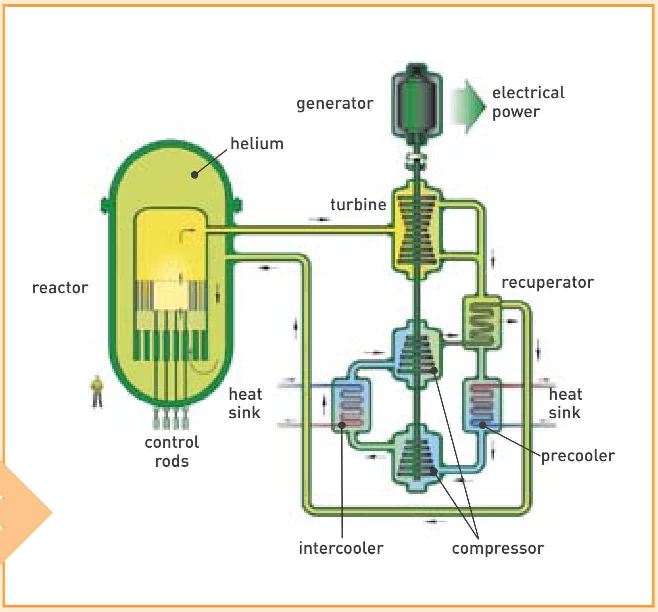
The six concepts selected by the Gen IV Forum

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

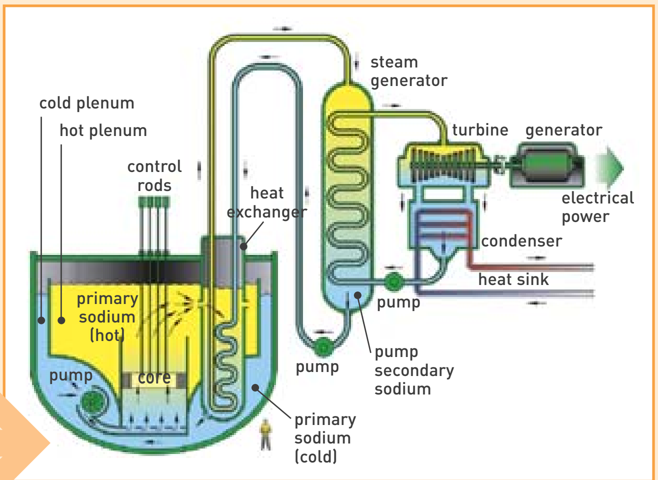
The gas-cooled fast reactor system (GFR) is a high-temperature, gas-cooled (helium-cooled as a rule), fast-neutron reactor allowing actinide recycle (homogeneous, or heterogeneous), while sustaining a breeding capability greater than unity. The reference concept is a helium-cooled, direct- or indirect-cycle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



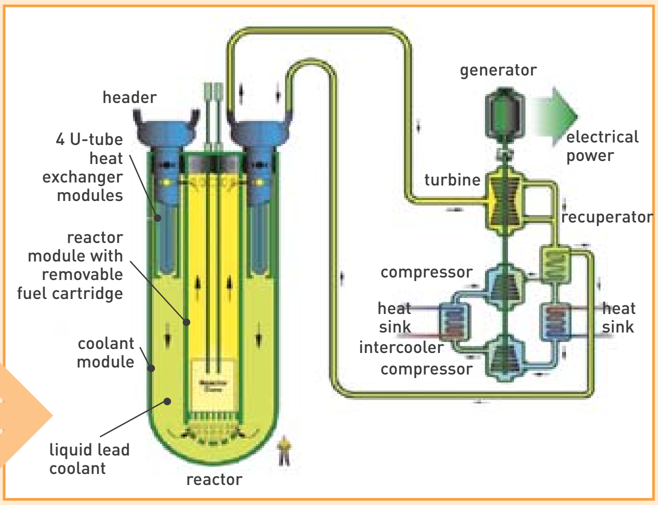
Le SFR

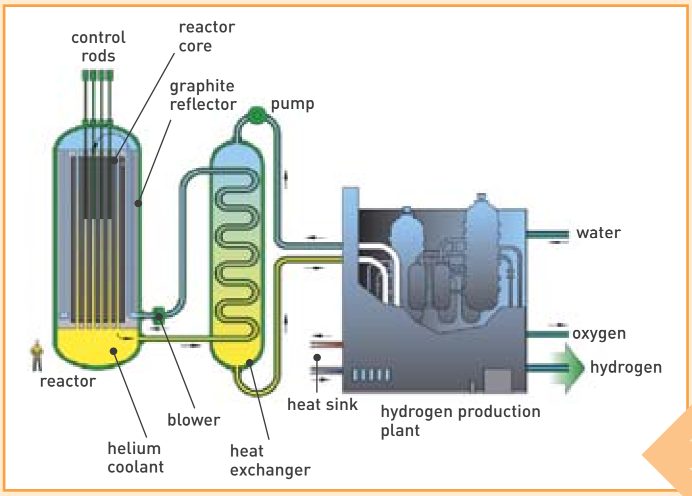
The sodium-cooled fast reactor system (SFR) is a liquid-sodium-cooled, fast-neutron reactor, associated to a closed cycle, allowing full actinide recycle, and plutonium breeding. Owing to its breeding of fissile material, this type of reactor may operate for highly extended periods without requiring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

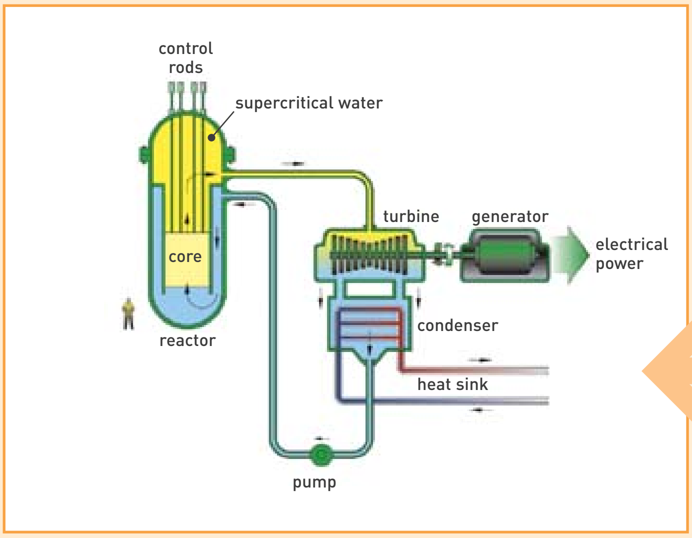
The lead-cooled fast reactor system (LFR) is a lead- (or lead–bismuth alloy-) cooled, fast-neutron reactor, associated to a closed fuel cycle, allowing optimum uranium utilization. A number of reference systems have been selected. Unit power ranges from the 50–100 MWe bracket, for so-called battery concepts,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





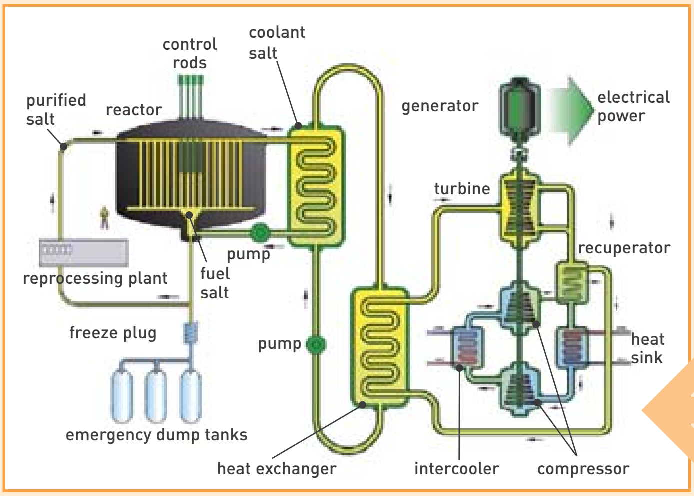
VHTR

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.

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.