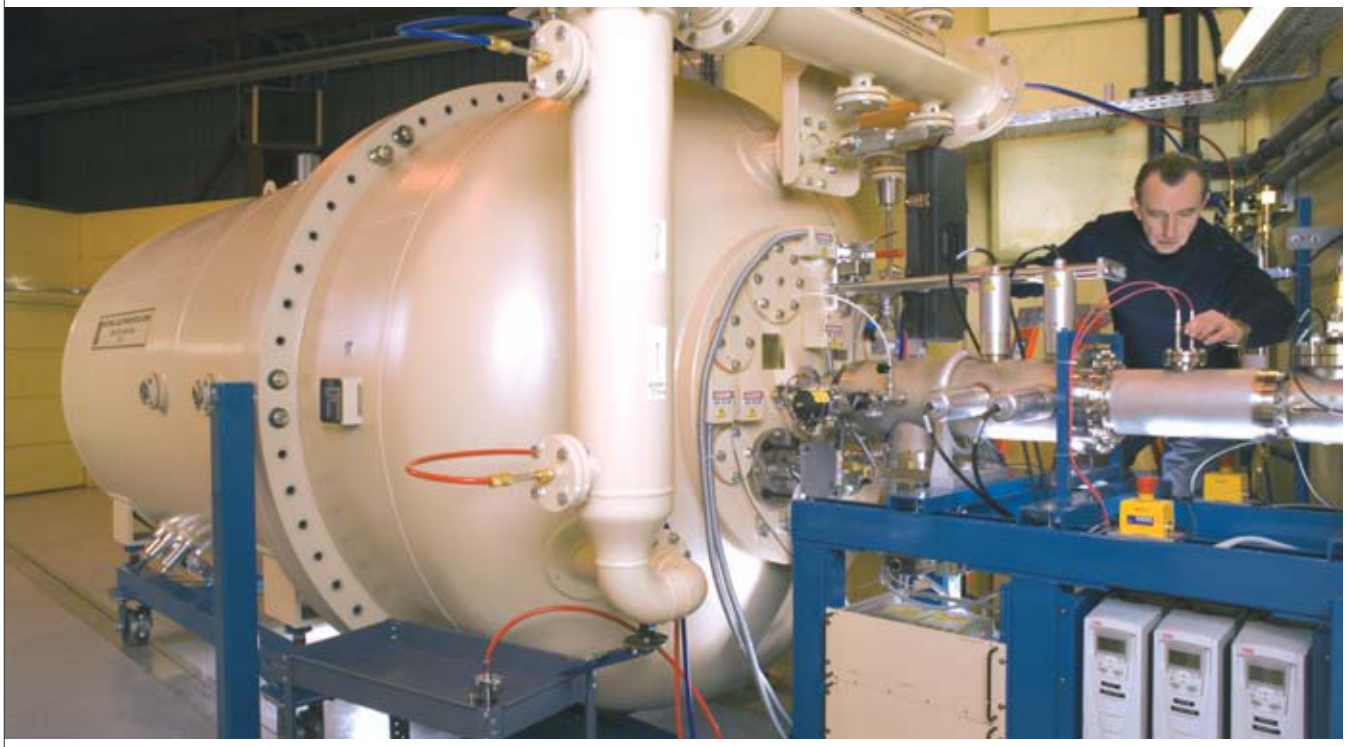
The **JANNUS** platform: ion-beam **simulation** of **materials irradiation**

Achieving the ability to predict, on a short-term basis, the long-term behavior of materials is a major challenge, as regards the viability of nuclear technologies. The aging of materials subjected to neutron irradiation raises issues, for which the JANNUS experimental ion-beam simulation project, set up by CEA and its research partners, intends to bring swift, concrete answers, in close connection with numerical simulation.



A. Gonin/CEA

As a component of the JANNUS multi-irradiation platform, the 3-MV Épiméthée accelerator will yield heavy-ion beams, for the purpose of generating atomic displacement cascades in materials subjected to investigation, to simulate neutron-induced damage. For that purpose, it is fitted with an ECR source, able to produce highly charged ions, such as He^{2+}, Ar^{11+}, Fe^{10+}, or Xe^{17+}, in order to achieve high energies.

What will the behavior of materials exposed inside **nuclear reactors** turn out to be, in fifty years' time? Which materials will prove able to withstand **irradiation**, at very high temperature, in the reactors of the future, or under the extreme conditions prevailing in **fusion** reactors? With respect to **vitrified** waste, which will prove to be the decisive factor, **atomic** collisions, the formation of **electronic** defects, or heating?

Experiments in actual operating conditions require considerable leadtimes, conceivably unattainable with respect to nuclear waste, along with conditions that may only be found in prototypes still being built, or planned. The considerable advances achieved by **numerical simulation** allow irradiation mechanisms to be explored at increasingly diverse scales, however this cannot really take on board the complexity of actual materials, or of combinations of stresses. Experimental simulation, subjecting actual materials to exposure to conditions more readily accessible than actual operating conditions, over shortened intervals, provides a medium way.

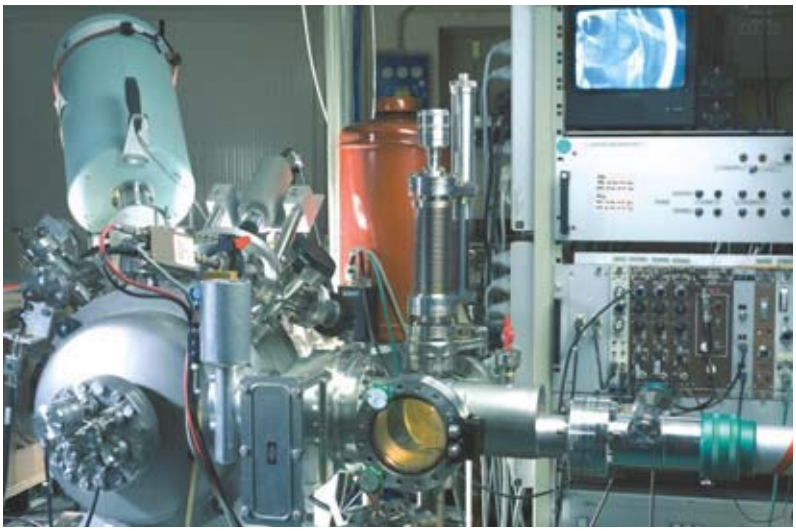
Ions instead of neutrons

Neutrons interact with matter through collisions with **nuclei** (ballistic damage), and nuclear reactions. The initial atoms struck, once set in motion by neutrons, are nothing else than **ions** of the material. The **cascades** of atomic collisions they set up, and the electronic excitations they induce on their path, along with atoms displaced by them, overwhelmingly account for the ballistic damage found. The latter thus comes down, in practice, to an internal ion bombardment. It may thus be seen as legitimate to substitute an ion beam, for the neutrons involved. The nuclear reactions yield, within the material, reaction products, most commonly **light nuclei** – **hydrogen**, or **helium** – but also heavy **transmutation** products, through neutron absorption. Injecting such products

by ion implantation[1] makes it possible to simulate the emergence of these extraneous nuclei inside the material.
Simulating these two effects by ion beams, generated by electrostatic accelerators,[2] affords many benefits. Far higher damage and implantation rates than with neutrons may readily be achieved: over a few hours, it is feasible to simulate the entire in-reactor lifetime of the material. The material is not **activated**, which makes it possible to observe it during irradiation, and immediately afterwards, with no particular precautionary measures required. Operating a small accelerator proves far less expensive, and far more flexible, than operating a nuclear reactor, insofar as its functioning parameters may be readily modified. On the other hand, since a single-energy ion beam does not mimic the diverse energies (and, in some cases, masses) exhibited by the atoms initially struck by neutrons, several experiments, involving a variety of beams, are required to predict the effects of neutrons.
When the aim is to effect a deliberate alteration of the materials (in terms of properties, microstructure, composition…), ion beams, owing to the flexibility with which irradiation conditions may be varied, the short experiment time required, and ease of *in situ* observation, prove particularly well suited to researching the conditions for the desired thermodynamic (and possibly chemical) disequilibrium. Subsequently, they prove highly suitable for the production, in large batches and at low cost, of components, be they electronic, magnetic, optical…

Double beam, triple beam

Atomic **defects**, generated by collisions, and reaction products are mobile. They interact with each other, cancel each other out, or come together. Further, the presence of excess atomic defects alters atom mobility. The kinetics of these interactions is thus dependent on the irradiation flux, and not only on temperature, as would be the case out of irradiation conditions. Thus, it comes as no surprise that the existence of combined effects – the mutual reinforcement of effects, or the partial healing of one effect by the other – was noted very early on. Typically, helium, in the event of prior implantation in **steels**, will not exhibit the same behavior at all as helium gradually generated by nuclear reactions under irradiation. Indeed, having had time to form clusters, it does not exhibit the same mobility. It is also highly likely that, in vitrified waste, the opposing effects of atomic motions, and electronic excitations yield a resultant that may not be simulated, when these are separated in time. The advantage of exposing the material, at one and the same time, to two or three ion beams, to mimic simultaneously damage effects, and reaction product generation effects, or the effects of **alpha emission**, with concomitant **recoil nuclei**, will thus be apparent. For the controlled modification of materials, combining implantation (the chemical factor), with injection of ballistic and electronic defects (the kinetic factor), likewise involves use of a double beam.



A. Gonin/CEA

The ion-beam analysis chamber for the Yvette accelerator, fitted with various charged particle detectors and **X**- and **gamma photon** detectors. In the JANNUS project, the Yvette accelerator, as part of the triple beam, will be devoted to helium implantation, to simulate its formation through nuclear reactions, and *in situ* characterization of the materials investigated, by means of ion-beam analysis techniques. It will also be used for *ex situ* characterization and teaching.

Experimental and numerical simulation: a mutual enrichment situation

The reactors of the future, whether **fourth-generation** reactors or fusion reactors, will have to turn to refractory materials, further exhibiting high irradiation resistance: metallic materials, but equally **ceramics**, and **composites**, for which behavior under irradiation is far less well known than is the case for metals, or **alloys** (see *Putting the properties of non-metallic materials to advantage*, p. 78; and *Metallic materials, one of the keys for the fourth generation*, p. 71). A huge area is thus being opened up for investigation, with respect to these novel materials.
Theory, and **modeling** often make it possible to ascertain which configurations, or which kinetics are the most likely; to predict the possible emergence of unexpected processes; and suggest experiments for the purposes of corroborating such predictions. Such corroboration, and the quantitative determination of the parameters they involve are requisites, if the former are to make further headway – and this relies on experiment. The latter experiment sometimes results in further processes being uncovered, that will have to be included in the models. Conversely, experiment design will often turn to the model, to seek pointers as to foreseeable trends, and an estimate of the conditions for which the process subject to investigation will be found; and subsequently for corroboration of the interpretation of its findings, or keys for a better understanding.
With respect to this process of mutual enrichment, JANNUS (Jumelage d'Accélérateurs pour les Nanosciences, le NUcléaire et la Simulation: Joint Accelerators for Nanosciences and NUclear Simulation) will bring the swift response of ion beam experi-

(1) Ion implantation: a process whereby ions may be implanted into a solid, altering its properties, this yielding both a chemical alteration of the target, and possibly a modification of its **crystal** structure. The implantation apparatus comprises an ion source, an electrostatic accelerator, and a chamber, where the target is positioned.

(2) Particle accelerator: a machine devised to generate, shape, and accelerate a beam of charged particles directed onto targets, or other beams of accelerated particles. Accelerators may use electrostatic accelerating forces, i.e. forces exerted by static electric charges (Van de Graaff accelerators, tandem accelerators), or electromagnetic forces (cyclotrons, linear accelerators, synchrotrons).

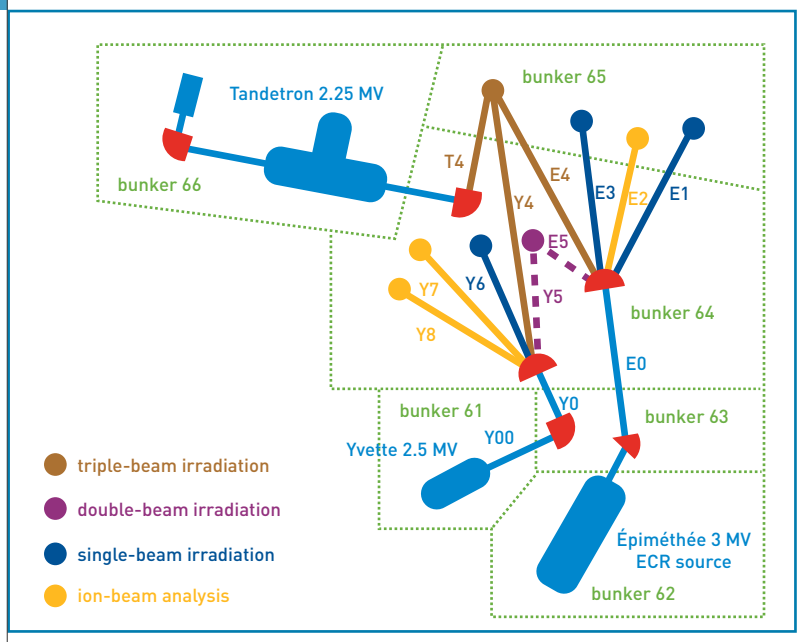


Figure 1.
Schematic of the triple-beam installation layout, at the Saclay site.

ments, while bringing together the diverse expertise of simulators, and experimenters, at the moment where the space and time scales involved in their activities are converging. As a joint project of CEA (Nuclear Energy Division/Nuclear Materials Department [DEN/DMN], and Institut national des sciences et techniques nucléaires [**INSTN**]), **CNRS** (Institut national de physique nucléaire et de physique des particules–Centre de spectrométrie nucléaire et de spectrométrie de masse [**IN2P3–CSNSM**]), and **Paris-XI–Orsay University**, with support from the Île-de-France Region (Paris area), and the Essonne *département*, JANNUS will make available to the international scientific community five accelerators, and one in-line electron microscope, for investigation and teaching purposes, with regard to irradiation effects, as applied both to nuclear energy materials aging, and the deliberate, controlled modification of industrial materials.

The equipment, and performance

The performance required, for the investigation of the aging of nuclear energy materials and the controlled modification of materials, broadly coincide. The taking over of extant equipment and buildings has led to work being spread over two sites, close to each other, with a triple-beam facility at Saclay, and a double beam, coupled to a **transmission electron microscope** (**TEM**) at Orsay.

At the Saclay site, three accelerators will be deployed around an experiment chamber, placed under ultra-vacuum (pressure in the 10^{-7}–10^{-9} **mbar** range), where their beams will converge onto a single target (see Figure 1), which may be heated up to 800 °C, or even 1,200 °C subsequently, or cooled down to the temperature of liquid nitrogen (– 196 °C). High damage rates – up to 100 **dpa** in one day – will be achievable, by means of **heavy ions**, from the 3-megavolt (MV) Épiméthée linear accelerator. Its electron cyclotron resonance[3] source, indeed, has the ability to generate high-intensity, multicharged ion beams. Multiplied as it is by the ion charge, the accelerating voltage endows these beams with such kinetic energy that penetration stands at several **microns** or so, in solids. Épiméthée will also feature a single-beam irradiation line, and will have the ability to accommodate, at a later date, three further lines. INSTN's 2.5-MV Van de Graaff accelerator, Yvette, will carry out helium implantation, also providing beams of hydrogen and helium **isotopes**, for ion-beam analysis purposes, on two dedicated lines, and in the triple-beam chamber. It will have the ability to be fitted with two further lines at a later date. Hydrogen implantation will be carried out by means of a tandem accelerator, the 2.25-MV Tandetron,[4] this further having the ability to generate ions that are not readily obtained with the ECR source, halogens[5]

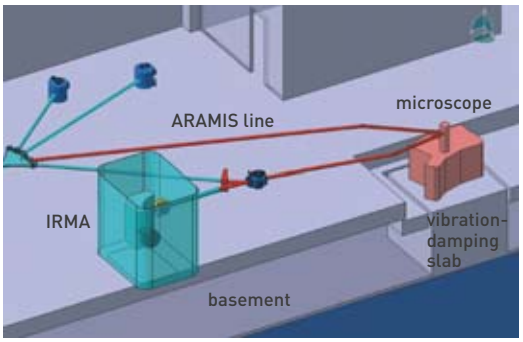


Figure 2.
Layout schematic of the double beam, linked to an in-line transmission electron microscope, at the Orsay site.

The Tandetron, a 2.25-MV tandem accelerator, is to have for its main task, in JANNUS, the simulation, through implantation, of the generation of hydrogen (in the form of H^+ ions) by nuclear reactions in irradiated materials. It will also be used to generate negatively-charged ion beams, such as Cl^-, I^-...

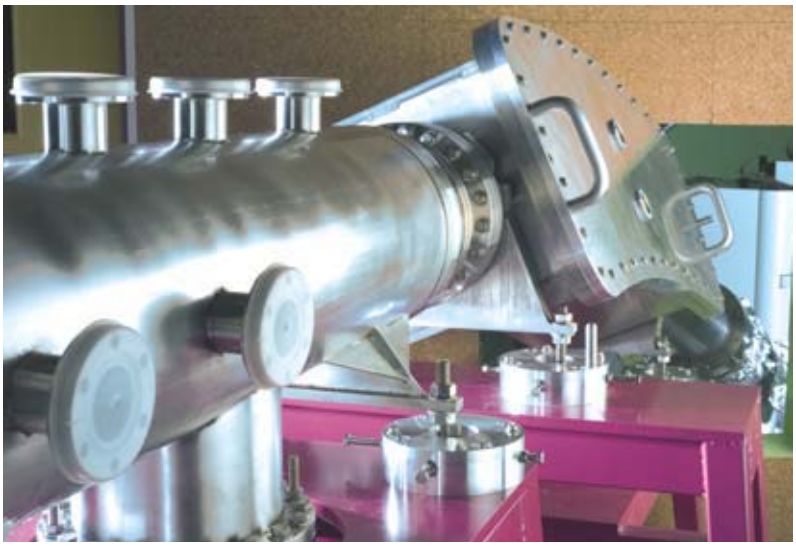
A. Gonin/CEA

(3) An electron cyclotron resonance (ECR) source is fitted to most heavy ion accelerators, in order in particular to generate multicharged ions (i.e. atoms from which several electrons have been stripped). A **plasma** is generated inside a magnetized vacuum chamber, by injecting an electromagnetic wave having a frequency equal to the *Larmor precession frequency* of electrons. This wave produces high-energy electrons, having the ability to ionize atoms, down to the inner shells.

(4) Tandetron: a tandem-type accelerator, in which ions are accelerated twice. Suitable in particular for mass spectrometry and low **radioactivity** measurements, this machine has the ability to carry out radioactive carbon-14 dating on amounts of organic material smaller than one milligram. The apparatus makes a direct measurement of the number of atoms (radioactive or otherwise) in the sample, by deflecting particles as a function of their mass, charge, and energy.

(5) Halogens: a group of chemical **elements** (fluorine, chlorine, bromine, iodine, astatine), some of which occur in gaseous form inside nuclear reactors, as part of the **fission products**, tending to slow down reactions.

in particular. In the latter configuration, hydrogen and helium implantation may be carried out by Épiméthée and Yvette. Irradiation and implantation homogeneity, across a 20-mm diameter, will be ensured, in depth at the target, by energy degraders, and, laterally, by electrostatic scanning on each line of the triple beam.
At Orsay, CSNSM can avail itself of one 2-MV, tandem-type accelerator, ARAMIS, and one 190-kV implanter, IRMA. Coupling a new, 200-kV transmission electron microscope to these two machines will make it possible to investigate damage kinetics, through the continuous observation of a region exposed to two irradiation beams, or to one irradiation and one implantation simultaneously (see Figure 2). This microscope will bring **resolution** better than 0.3 **nm**, and operate under a vacuum at $5 \cdot 10^{-8}$ mbar. An ultrathin sample-holder, heating to 800 °C, has been designed to minimize masking effects, whatever the tilt of the object. Measurement of the flux, on a continuous basis, will be available for both ion beams.


A. Gonin/CEA

The ion beam line from the ARAMIS accelerator, a 2-MV tandem accelerator sited at Orsay, arches down towards the column of the transmission electron microscope.
The ions will strike the target surface at an angle of 22° relative to the horizontal plane.

A multiplicity of applications to materials for the reactors of the future

Bearing in mind the vast range of materials being investigated for use in fourth-generation nuclear reactors, applications for the JANNUS multi-irradiation platform are many. Two research thrusts, in this respect, will be mentioned here.
The first example concerns the behavior under irradiation (around 100 dpa), and at high temperature (400–550 °C) of chromium **steels** exhibiting a **martensitic structure**, representative of the **vessel** material in future reactors. The aim is to ascertain precisely the **swelling** mechanisms occurring in such alloys, under the joint effects of damage, and helium accumulation. The major experimental parameters being monitored will be temperature, flux, total damage dose, and alloy chemical composition.
The second example relates to **TRISO**-type particle **fuels**, intended for high-temperature reactors (see *High-temperature reactors: a recent past, a near future,* p. 51), and more particularly to the evolution of ceramic coatings (silicon carbide, or zirconium carbide). Answers must be provided, on the one hand, as to the resistance of the coating layer to an accumulation of **irradiation defects**, and, on the other hand, as to the effects of implantation, and atomic transport of **fission products**, emanating from the fuel core, within the carbide lattice. It should be noted, as regards the latter point, that numerical simulations at the atomic scale, relying on ***ab-initio* methods**, are already being undertaken.

A. Gonin/CEA

CSNSM's new transmission electron microscope will make it possible to observe the microstructural evolution of a sample during its simultaneous irradiation by two ion beams. The microscope column will receive two lines of energetic ion beams, directed symmetrically with respect to each other. The line from the ARAMIS accelerator may be seen. Construction of the line from the IRMA implanter is under way.

An international users facility

The JANNUS multi-irradiation platform will deploy, for the first time in Europe, a triple beam, and an in-line TEM, on a double beam. There are, around the world, just two other extant triple-beam facilities, at Kyoto and Takasaki, in Japan; four double-beam facilities, at Rossendorf and Jena (Germany), Salford (United Kingdom), and Tokyo (Japan); and one single-beam setup, coupled to an in-line microscope, at Argonne (United States). JANNUS will be opened up to the international community as early as 2008. It has been recognized by **Euratom** as an instrument of major importance for the development of the reactors of the future. Its unifying role, at the meeting point of simulation and experiment, will be combined with a teaching and training remit, in the area of radiation–matter interactions, and ion beam applications.

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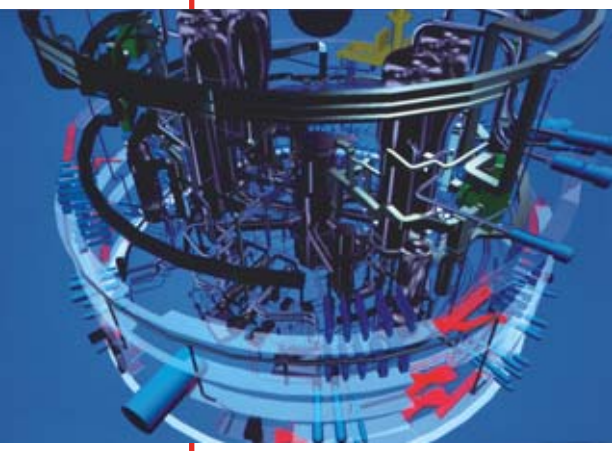
FOCUS A

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

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

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.

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

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

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

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

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

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

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

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

FOCUS C

Thermodynamic cycles and energy conversion

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

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

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

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

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

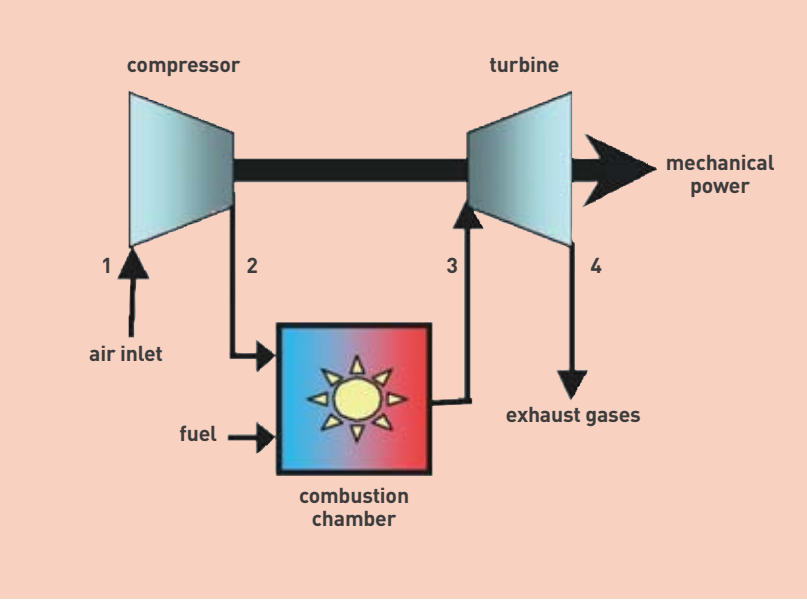


Figure.
Brayton cycle, as implemented in an open-cycle gas turbine.

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

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

What is multiphysics, multiscale modeling?

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

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

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

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

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

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

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

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

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

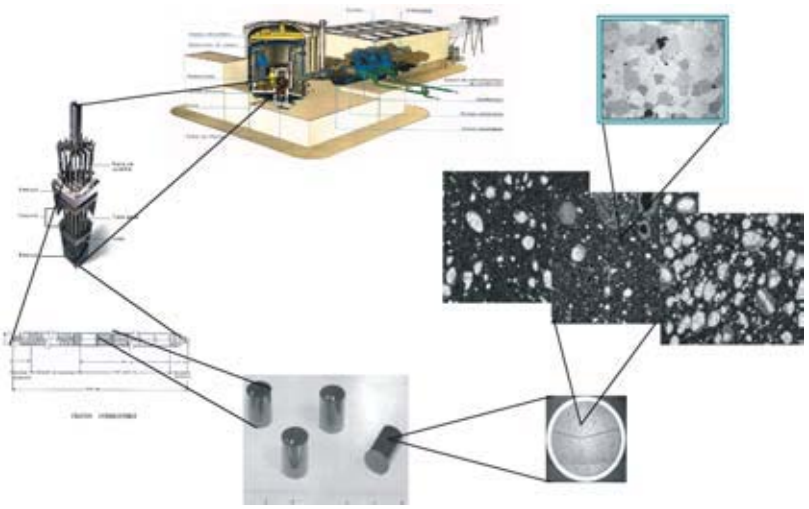


Figure.
Improving nuclear fuel reliability, and cost-effectiveness calls for finescale modeling of that fuel, through a multiscale approach, from reactor to fuel microstructure (in this instance, MOX fuel). Microstructural characteristics (porosity, cluster size and distribution, grain size...) have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

The main families of nuclear materials

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

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

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

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

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

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

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

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

Crystal structures

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

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

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

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

Steel families

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

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

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

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

Areva NP

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

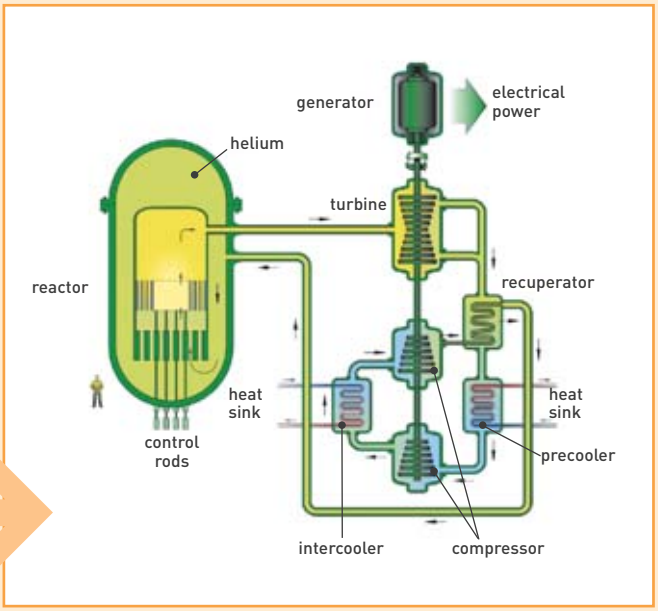
The six concepts selected by the Gen IV Forum

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

GFR

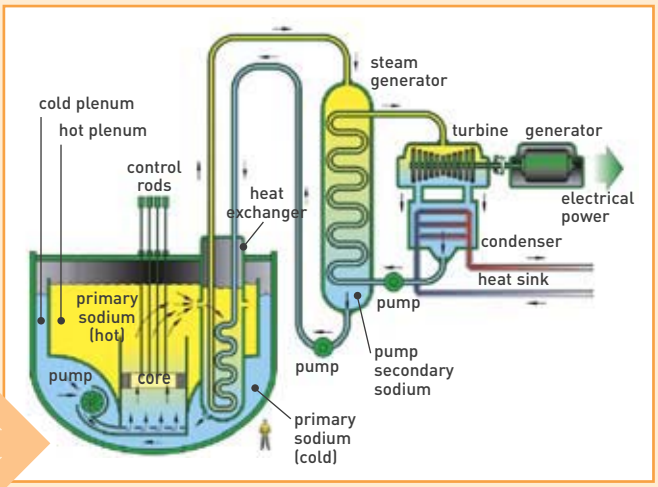
The gas-cooled fast reactor system (GFR) is a high-temperature, gas-cooled (helium-cooled as a rule), fast-neutron reactor allowing actinide recycle (homogeneous, or heterogeneous), while sustaining a breeding capability greater than unity. The reference concept is a helium-cooled, direct- or indirect-cycle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

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



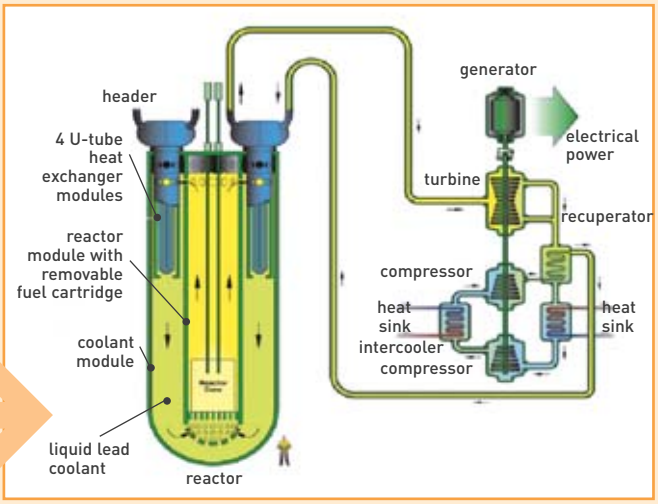
Le SFR

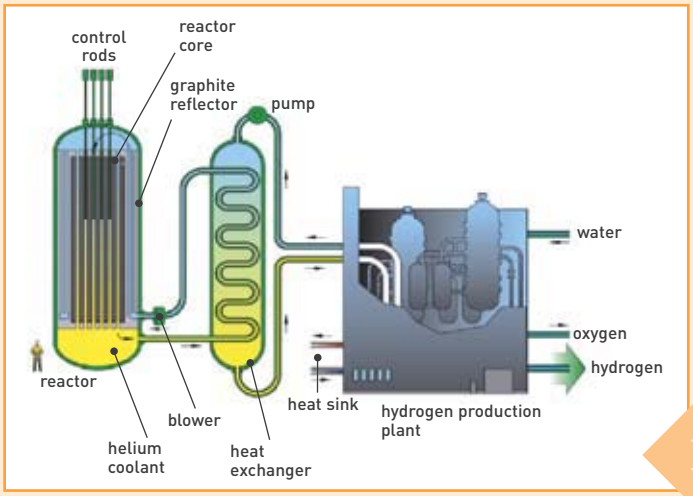
The sodium-cooled fast reactor system (SFR) is a liquid-sodium-cooled, fast-neutron reactor, associated to a closed cycle, allowing full actinide recycle, and plutonium breeding. Owing to its breeding of fissile material, this type of reactor may operate for highly extended periods without requiring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



LFR

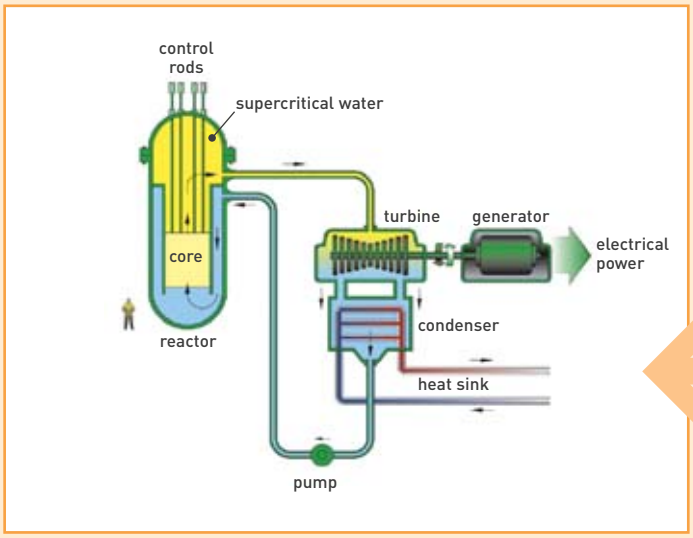
The lead-cooled fast reactor system (LFR) is a lead- (or lead–bismuth alloy-) cooled, fast-neutron reactor, associated to a closed fuel cycle, allowing optimum uranium utilization. A number of reference systems have been selected. Unit power ranges from the 50–100 MWe bracket, for so-called battery concepts,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





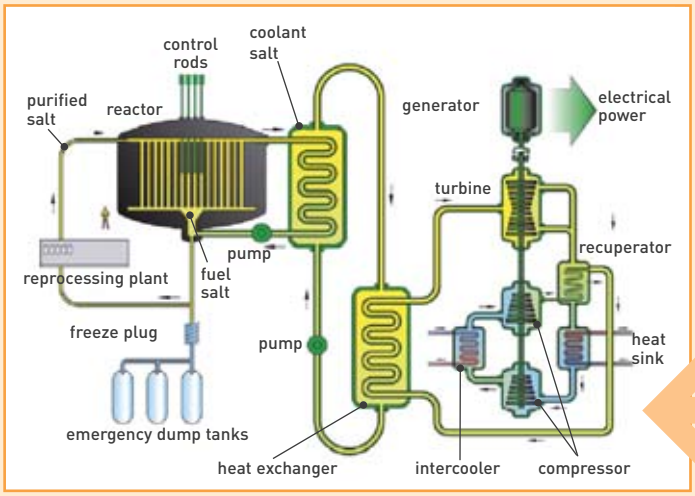
VHTR

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



SCWR

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



MSR

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