Putting the properties of **nonmetallic materials** to advantage

Alongside metallic materials, nonmetallic materials needs must make their contribution to overcoming the hard points, for reactors of the next generation, particularly for those intended to operate at high, or very high temperature. Ceramics, graphite, and composite materials will find their properties are fully employed, to adjust to the requirements of the new systems, particularly with respect to fuel.

At the present stage of research work on materials for the systems of the future, the investigations launched by CEA and its partners have the prime aim of ascertaining the general data as to the initial thermal–physical, and mechanical properties of the "high-temperature" materials being considered (**ceramics**, and **graphite**, in monolithic or composite forms). Their stability under **irradiation**, for high **fast-neutron fluences**, has to be tested over a wide range of temperatures, and the investigation of the sensitivity of such properties to relevant parameters (fabrication method, purity, **stoichiometry**, **grain** size, porosity) must be undertaken. The experimental irradiation program involves a number of campaigns (FUTURIX, MATRIX: see *Phénix, a unique instrument in the area of fuel*, p. 98), for the investigation of these materials' behavior under irradiation. At the same time, **numerical simulation** experiments are being carried out, to predict the resistance of such materials, mainly with respect to high damage rates, and high temperatures.

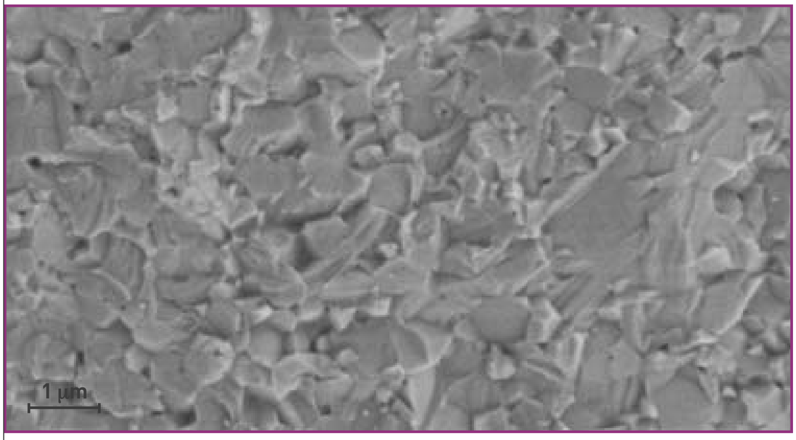
After a survey of the knowledge gained as regards the behavior under irradiation of ceramics as a whole, and the main research areas that have yet to be covered, the following pages provide an overview of graphite, a material being considered for very high temperatures (**VHTR** concept), followed by an outline of the form materials may take, for future high-temperature systems (**GFR** concept), to wit fiber-reinforced **composite** materials.

Ceramics: increasing mastery

Ceramics are materials that go back a long way, being employed from the outset for their resistance at high temperature; nowadays they are grouped into two distinct families: conventional ceramics (the terracotta of our forebears), and "technical" ceramics, of which the Space Shuttle's tiles are a spectacular instance.

It is nowadays agreed such materials are to be defined as nonmetallic, nonorganic materials. They come in a variety of forms: oxides, nitrides, carbides, or combinations of these. All such ceramics are obtained in bulk, through the application of strong heat, combined with external pressure or otherwise, transforming the initial powder mass into a continuous, cohesive solid. They may also be fabricated through use of so-called *chemical vapor deposition* processes.

Figure 1.
Fractograph of a zirconium carbide ceramic, densified with no additions, after grinding, and **sintering** under load.



Aside from their resistance at high temperature, ceramics generally exhibit good mechanical strength, except with regard to **toughness**, and deformation, in which respect they may be improved by turning to the composite path, or nanostructuration (see Figure 1). These two approaches are being investigated, to develop ceramics as **structural** and **cladding materials** for the reactors of the next generation.

How are damage and alteration mechanisms under irradiation to be modeled?

The alteration in operational characteristics exhibited by materials is the direct result of alterations in their chemical, crystallographic, and microstructural integrity. In solid-state materials, irradiation induces three categories of processes, corresponding to three time scales, the mechanisms involved proving far more complex, in ceramics, for insulators in particular, than in metals (see Focus E, ***The main families of nuclear materials***, p. 76). Transmutations alter the chemical identity of **atoms**, yielding new species, which alters

P. Stroppa/CEA

Raman microscopy analysis cell, for the microstructural characterization of irradiated nonmetallic samples, at the Irradiated Materials Investigations and Test Laboratory, at CEA/Saclay.

the material's composition. Of such species, **helium** is of particular importance.
Nuclear collisions induce atomic displacements, modifying the position, and arrangement of atoms, generating **point defects (PDs)** – **vacancies**, and **interstitials** – along with small PD clusters. Under neutron irradiation, such displacements occur in **cascades**, taking a few picoseconds (10^{-12} s). In certain insulators, electronic excitations induce very small atomic displacements, possibly going as far as generating PDs, and major alterations in the electronic structure, strongly affecting defect mobility, as a rule in the sense of very strongly accelerating them. Finally, some ceramics, mainly oxides, undergo **amorphization**, either directly in cascade core regions, or through the accumulation of PDs. Molecular dynamics(1) has recently shown that oxides exhibit the entire gamut of primary damage mechanisms under irradiation, from total amorphization of the cascade cores, to generation of isolated defects, with no amorphization involved, as in uranium oxide, through core amorphization with concomitant generation of isolated PDs at the periphery. Zirconium carbide (see Figure 2), which is electrically and thermally conductive, exhibits damage that is typical of metals, but equally… of **uranium** oxide, which is an insulator! The physical causes for such discrepancies in behavior have yet to be understood.
Finally, the material's microstructure, when subjected to such damage, undergoes, throughout irradiation, a slow evolution, governed by diffusion,(2) which may take a variety of forms: from the aggregation of point defects (**interstitial loops, vacancy clusters**), and solutes (elements in solution), to the precipitation of particles, **cavities** (large vacancy clusters), gas bubbles, intergranular segregation,(3) phase transformations (**radiolysis**, amorphization/crystallization(4)), etc. Such evolution may be either induced, or merely accelerated by irradiation. The ongoing injection of energy prevents the system from achieving thermodynamic equilibrium: its state, and evolution are governed by the competition between ballistic atomic jumps, and thermally activated jumps. The issue is that of understanding, and predicting, the kinetic paths, and possible steady states to which it may tend *(dynamic phase diagrams)*.
Thus, in irradiated silicon carbide (SiC), low-temperature swelling is due to amorphization, possibly reaching some 10%. At higher temperature, this ceases to occur, and swelling is then due to the accumulation of the point defects generated, these in recombining contributing to a reduction in swelling of up to 0.2%. Above 1,000 °C, swelling results from the aggregation of vacancies, rising with fluence and temperature, with no apparent saturation with respect to the former. If the material is subjected to **stress** during irradiation, **irradiation creep** results in deformation which increases with flux, fluence, and temperature. As regards SiC, this remains small, and weakly temperature dependent below 900 °C, though increasing beyond this point.
Likewise, **thermal conductivity** is altered by irradiation. In SiC, this decreases, showing signs of saturation below 25 **dpa**, and 1,000 °C, doubtless owing to the selfsame mechanism, of aggregation of point defects, that underlies medium-temperature swelling; however, the behavior of metal carbides (ZrC, TiC) does not exhibit the same evolution as in insulators such as SiC. As is the case for all properties, thermal conductivity proves highly sensitive to chemical composition, **stoichiometry**, and impurities, as well as to microstructure (grain size, porosity…).

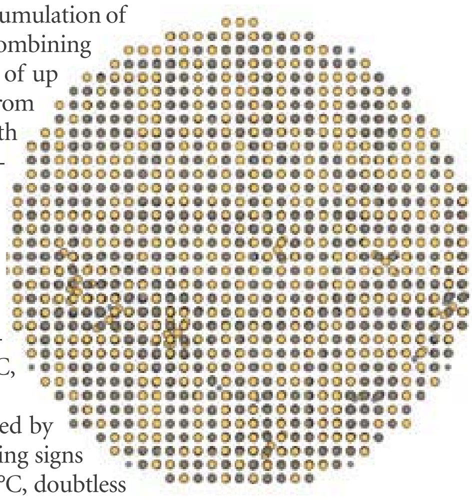


Figure 2.
Molecular dynamics simulation of a displacement cascade in zirconium carbide (Zr atoms shown in yellow, C atoms in blue).

Modeling, and coupling with experiment

Owing to the complexity of the processes involved, predictive **modeling** may not rely solely on a phenomenological approach. To guarantee robustness, models must be grounded in physics, and, as far as feasible, at the scales where the latter is most fully ascertained, this often being – though not invariably so – that of the atom, the more so since irradiation damage generation and evolution mechanisms are precisely induced at this scale.
It is indispensable, first of all, to ascertain the precise diffusion mechanisms involved, and those for its alteration by irradiation: ***ab-initio*** computation of electronic structures allows the basic properties (structure, formation, migration) of point defects to be arrived at. This has made possible the full modeling of *self-diffusion*, as of the crucial effects of impurities. It is also

(1) Molecular dynamics: a simulation technique allowing the step-by-step description of the dynamics of the ensemble of atoms in a molecular system, once the forces exerted on each atom are known. When starting from first principles, this is known as *ab-initio* molecular dynamics.

(2) Diffusion: the migration of atoms inside a solid, liquid, or gaseous medium, under the effects of thermal agitation.

(3) Intergranular segregation: the adsorption of atoms at grain boundaries.

(4) Crystallization: the process bringing to the crystalline state, characterized by a regularly ordered arrangement of particles of matter, as opposed to the amorphous state.

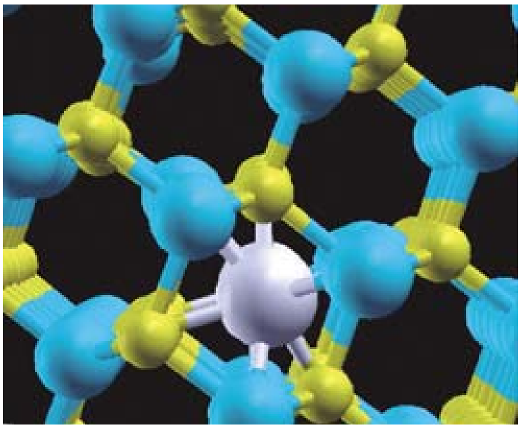


Figure 3.
Diffusion of palladium in SiC: a palladium atom is seen at the saddle position, ready to migrate along channel 110 of the silicon carbide's cubic lattice; results from *ab-initio* computation (Pd shown in gray, C in yellow, Si in blue).

used to understand **fission product** diffusion in SiC (see Figure 3).
Molecular dynamics is the basic tool for the investigation of ballistic damage processes, however its effectiveness is dependent on the quality of the interatomic potentials used. As regards insulators, moreover, the *ab-initio* approach, i.e. an approach based on rigorously taking into account the quantum character exhibited by physics at the atomic scale, is indispensable if electronic effects are to be taken on board, in particular the damage due to particles other than high-energy neutrons, **electrons**, and **photons**.
Predictions of long-term microstructural evolution kinetics, as outlined above, rely on already highly developed models, that have shown good performance with respect to metals, application of which to ceramics, however, is only just beginning. The understanding, and modeling of mechanical behavior are likewise far more advanced for metals than for ceramics, however the **multiscale** approach, starting from the atomic scale, is still barely at an initial stage.
At the same time, modeling must be closely coupled with experiment. Aside from gaining the relevant data as to behavior subsequent to neutron irradiation for the materials selected, it is indispensable to conduct a targeted experiment drive, aimed at ascertaining basic physical properties, and behaviors, and at the parametrization, and validation of the models
Thus, charged-particle irradiation – involving **ions**, and electrons – affords the possibility of mimicking, and analyzing, in detailed fashion, damage mechanisms in small, in**activated** samples, which are thus amenable to a whole range of measurements, and observation, from the atomic scale up, both *in situ*, and *ex situ*.
Only through such coupling of experiment with modeling may the representative character of experimental irradiations be founded, along with the validity of extrapolations to operational conditions. Further, this provides an invaluable tool for the design, and optimization of in-reactor irradiation experiments, in terms of relevance, and economics, and for the interpretation of findings.

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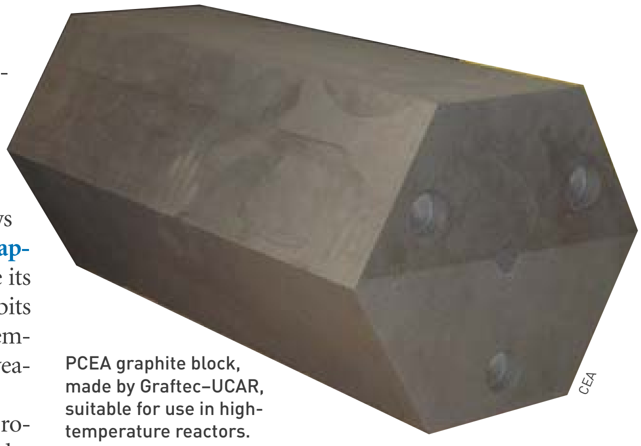
Graphite revisited

Graphite blocks for the stack in the Japanese HTTR high-temperature reactor (top), and reflector blocks for the Chinese HTR-10 high-temperature reactor (bottom).

Carbon comes in many forms, but just two allotropic(1) forms: diamond, and graphite, which **crystallizes** according to the **hexagonal** system (see Focus E, ***The main families of nuclear materials***, p. 76). Provided it is sufficiently pure, graphite acts as a good **neutron moderator**, as it slows neutrons down, while not absorbing them (the **capture cross-section** exhibited by ^{12}C is small, while its **elastic scattering**(2) **cross-section** is large). It exhibits advantageous mechanical characteristics at high temperature, is relatively easily machined, and is but weakly activated under irradiation.
A nuclear graphite must exhibit good mechanical properties, and satisfactory dimensional stability under irradiation. It should capture as few neutrons as possible, and, finally, yield acceptable waste subsequent to irradiation, hence bear very low amounts of absorbent impurities, or such as are liable to activation.
Nuclear graphites are produced (see figure 4) from petroleum coke, or from coal-tar pitch mixed with a binder. The calcinated coke is ground, passed through a sieve, then the grains obtained are mixed in suitable proportions, to achieve adequate density, and favor the escape of volatile materials from the binder. The coke mix is as a rule blended at 165 °C with a coal-tar pitch, shaped by extrusion forming, or one-directional pressing, then baked at 800–1,200 °C to coke the binder. Subsequently, the product undergoes one or more impregnations, to increase its density, and enhance its mechanical characteristics. Finally, it is graphitized at 2,500–3,000 °C, to obtain a hexagonal crystal structure (see Figure 5). This graphitization is carried out in the presence of purifying agents (NaF, MgF_2, Cl_2…), allowing a "nuclear-grade," low-impurity-content graphite to be obtained.



PCEA graphite block, made by Graftec–UCAR, suitable for use in high-temperature reactors.

(1) Allotropic: having the property, for a substance, of keeping its chemical properties, despite a change in its crystal structure.

(2) Elastic scattering: an interaction between particles, involving the transfer of momentum from one particle to another, while not altering the latter's nature (the classical equivalent being the collision of two billiard balls).
The incident particle rebounds off the target particle with a kinetic energy lower than its initial energy. The transferred momentum imparted to the target causes it to recoil. If its recoil energy is sufficient, the target particle thus emitted may be detected (e.g. a recoil electron, when scattering take place on an electron target).

The nature of the coke used, and the shaping technique selected are of great importance, since they determine the isotropy(3) exhibited by the graphite obtained, and hence the evolution of its macroscopic properties under irradiation.
At the scale of crystallites(4) (sizes of which, along the *c* crystallographic axis [L*c*], range from 20 **nm** to 140 nm), which may be seen as single crystals,(5) irradiation by fast neutrons causes a displacement of carbon atoms, to interstitial locations, between graphene planes, and vacancies within these plane (see Figure 6). The accumulation of vacancies results in a contraction of crystal cells along the *a* axis, while accumulation of interstitials causes expansion along the *c* axis. Under irradiation, crystallite size along the *a* axis (L*a*) will thus decrease, whereas L*c* will increase. Such an evolution of crystallites under irradiation will obviously induce an evolution in the macroscopic properties exhibited by the irradiated graphite (dimensional changes, mechanical and thermal properties). Now, in high-temperature reactors, irradiation temperature ranges from 500 °C to 1,200 °C.
Dimensional changes in polycrystalline(6) graphites are dependent on three parameters. The first of these is irradiation temperature: between 300 °C and 700 °C, a contraction occurs, along the polycrystalline graphite's two preferential directions, with larger deformations found along the direction parallel to the grains (see Figure 7). Deformation rates, as a function of fluence, decrease as irradiation temperature rises. Above 700 °C, contraction likewise occurs, along both directions, however deformation rates now rise with irradiation temperature. Crystallite size is the second parameter: crystallite size, and perfection rise with graphitization temperature. Now the larger the crystallite size, the greater the dimensional stability of graphite under irradiation is found to be. The third parameter is graphite isotropy: dimensional variations are all the smaller, the more the graphite is isotropic.
For a given temperature, thermal conductivity decreases monotonically as fluence rises. It reaches a saturation value, for a fluence that is irradiation temperature-dependent. For a given fluence, the degradation of thermal conductivity under irradiation is all the smaller, the higher the irradiation temperature (see Figure 8).
Under irradiation, the **Young's modulus** (E) of polycrystalline graphite rises very significantly, owing to the blocking of shear deformations by interstitial defects, which may result in embrittlement of the material. This increase already occurs for low fluences, and is found to be all the larger, the lower the irradiation temperature. A similar behavior is found for fracture strength (tensile, bending, compressive), and the *stress intensity factor* (K_{IC}).
Whereas thermal creep in graphite only becomes notable above 2,000 °C, irradiation creep already sets in at 100 °C, resulting in deformations that may be ten times larger than those obtained out of irradiation.(7) Primary and secondary creep deformations correspond, respectively, to the two terms in the equation, being proportional to applied stress σ. The primary creep constant *b* rises with irradiation temperature; the secondary creep constant *k* only rises with irradiation temperature over the 500–1,400 °C range.
While it exhibits satisfactory mechanical behavior at high temperature, graphite does prove highly sensitive to the presence of oxidizing species in the **coolant** helium. At the same time, accident scenarios for high-

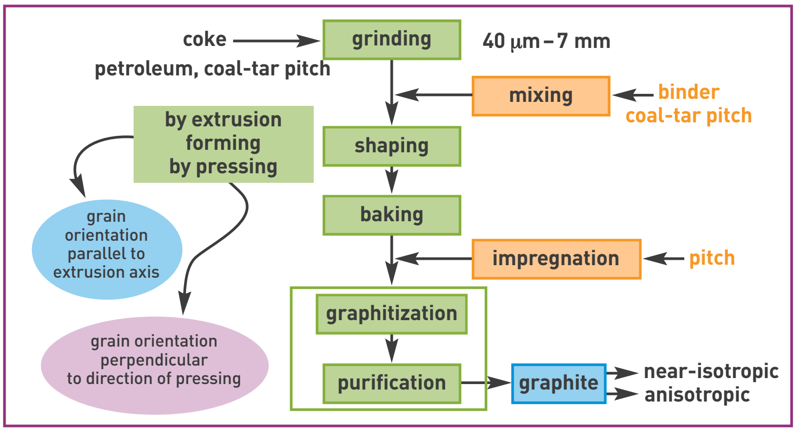


Figure 4.
Production of graphite.

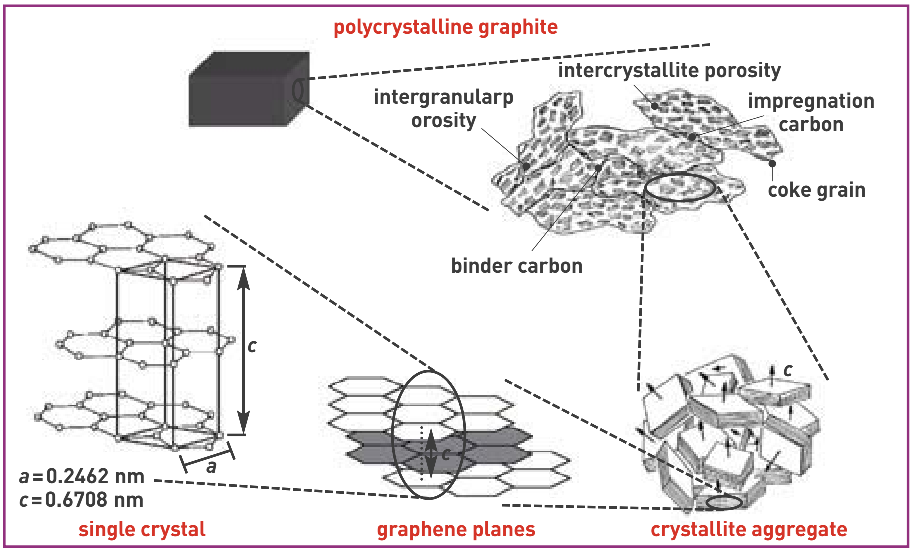


Figure 5.
Structure of graphite.

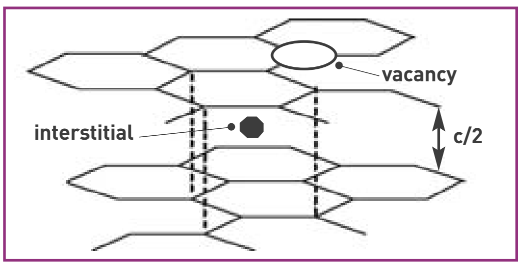


Figure 6.
Neutron irradiation damage in a crystallite.

(3) Isotropy: the characteristic of exhibiting identical physical properties in all directions.

(4) Crystallite: a region of material (a "grain," in metallurgical parlance) exhibiting the same structure as a single crystal, of a size ranging from a few nanometers to several millimeters, the boundary between two crystallites being a "grain boundary."

(5) Single crystal: a macroscopic object, forming a single piece, obtained by translation of the elementary unit cell of crystal structure across the three spatial dimensions, within which the crystal's properties consequently reflect those exhibited by that structure at the atomic scale.

(6) Polycrystalline: consisting of several crystals.

(7) Creep deformations under irradiation ε_c may be expressed as follows:

$$\varepsilon_c = \frac{\sigma}{E_0}(1 - e^{-b\gamma}) + k\sigma\gamma$$

where σ is the stress, E_0 the graphite's Young's modulus before irradiation, and γ is the neutron fluence.

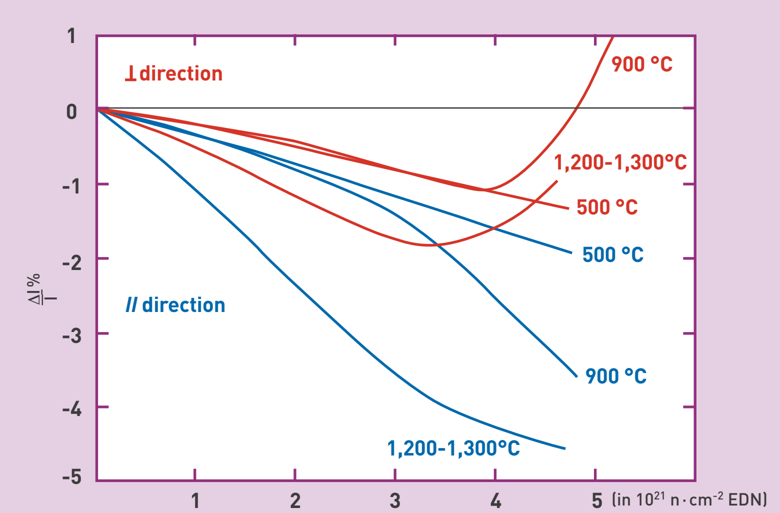


Figure 7.
Dimensional changes in a near-isotropic polycrystalline graphite, at various irradiation temperatures, along the directions parallel, and perpendicular to extrusion. EDN: equivalent DIDO nickel. A fluence of 10^{21} n · cm^{-2} (EDN) is equivalent to a fluence of 1.76 · 10^{21} n · cm^{-2} (E > 0.1 MeV).

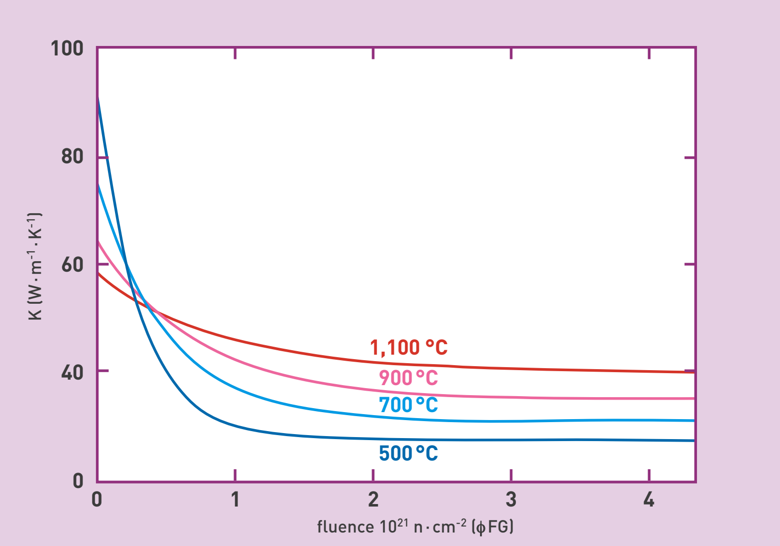


Figure 8.
Thermal conductivity K of a near-isotropic polycrystalline graphite, at various irradiation temperatures (thermal conductivity is measured at irradiation temperature).
ϕFG: equivalent fission fluence for graphite. A fluence of 10^{21} n · cm^{-2} (ϕFG) is equivalent to a fluence of 0.96 · 10^{21} n · cm^{-2} (E > 0.1 MeV).

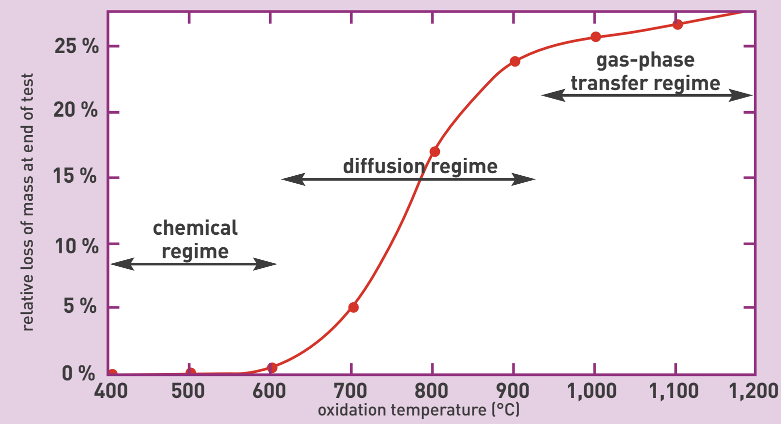


Figure 9.
Oxidation kinetics of a graphite, using thermogravimetry (relative mass loss after 4 hours' oxidation in dry air), as a function of oxidation temperature.

EDF Médiathèque/Michel Brigaud

Assembling the graphite stacks in the EDF3 reactor, at Chinon (France), in the mid-1960s.

temperature reactors do take on board a massive air inflow into the primary circuit. It is thus apparent that it is highly important to gain knowledge of graphite oxidation by air, which in practice is governed by a variety of processes, depending on temperature (see Figure 9).

Up to 500–600 °C, graphite oxidation is controlled by the chemical reaction with oxygen. This is slow, but uniform across the entire graphite volume, which may result in major degradation of mechanical properties. In such conditions, the presence of certain impurities (iron, vanadium, lead) will act as catalyst(8) for the oxidation reaction. Between 600 °C and 950 °C, the limiting step becomes the diffusion of gaseous species across the graphite porosity. Oxidation rate increases, however oxidation takes place at a lower depth inside the solid. The important parameter, in this case, is graphite porosity. Above 950 °C, graphite oxidation is controlled by the arrival of gas at the surface of the sample. It is faster, and advances from the outside of the graphite. At the end of its in-reactor life, graphite forms a nuclear waste material, for which knowledge of the radiological inventory is essential. Operational feedback on graphite from **UNGG** reactors shows their specific **activity** stands at around 5 MBq/g, after 5 years' cooling. The most problematical **radioelements** are ^{3}H, ^{14}C (90% of the activity, after 120 years' cooling), and ^{36}Cl, which, while it exhibits low specific activity, has a **half-life** of 300,000 years, and is poorly retained in geological environments. In any event, such irradiated graphites are not intended to have, for an outlet, the Aube Disposal Center run by **ANDRA**, the option selected being that of a dedicated disposal facility, in a subsurface site. Graphite irradiated in future high-temperature reactors will run to considerable amounts (6,100 tonnes per 600-**MWth** reactor, operating over 60 years). It will thus become necessary to deploy a waste reduction strategy, as regards the graphite thus yielded, through implementation of such treatment processes as decontamination, incineration, recycling, or reuse of irradiated graphite.

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(8) Catalyst: that which accelerates a chemical reaction, without the accelerating substance itself being altered, or only temporarily so

Composites make their mark for high temperatures

The advantage afforded by fiber-reinforced **composite** materials lies in their outstanding mechanical properties (specific strength, or specific Young's modulus), which, at ambient room temperature, may prove 5 or 6 times higher than those found for steel, which stands as a reference. This advantage is further sustained at high temperatures (higher than 1,000 °C), since composite ceramics exhibit, among other characteristics, reproducible behaviors, and high breaking strength, by contrast with monolithic ceramics, conventionally employed in this temperature domain.
With respect to the high- or very-high-temperature reactors of the future, use of carbon-based, or silicon-carbide-based composites is being considered for certain components requiring very high performance, such as **fuel claddings**, **control rods** and guide tubes,(1) **hot gas duct** insulators, or core support structures. Development work on such materials is being conducted under the aegis of in-house programs at CEA, partnerships with the Thermostructural Composites Laboratory (**Laboratoire des Composites Thermostructuraux**), and manufacturers such as **Areva**, and **SNECMA Propulsion Solide** (SPS, part of the **Safran** Group), as also of international collaborations (**INERI** project).

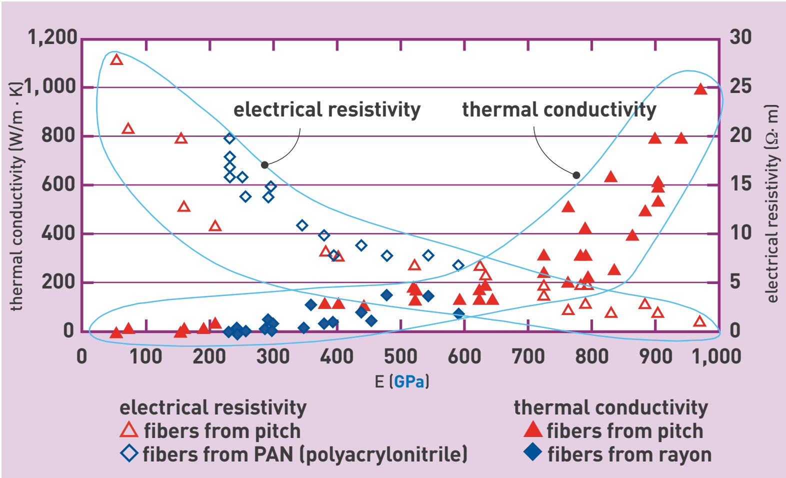


Figure 10.
Thermal conductivity and electrical resistivity (ohm · m) of pitch and PAN fibers at 300 K (data from suppliers; CEA/DTEC report).

Carbon–carbon composites for control rods

Carbon–carbon (C–C) composites feature the common characteristic of consisting of carbon fibers, and a carbon matrix, however this may result in materials exhibiting highly diverse properties. Indeed, carbon may synthesized under a variety of organizational forms, exhibiting characteristics – thermal, mechanical, electrical – that may range over several orders of magnitude, whether it be with regard to matrices, or fibers (see Figure 10). Designers may thus draw on a very wide variety of basic constituents, but equally of textile architectures, to specify materials best meeting the desired functions. At CEA, the two main applications, leading to advanced investigations, are for missile nosecones, and for components of the inner vessel in a thermonuclear **fusion** reactor, facing the plasma. In the case of control rods (assemblies of tubes, some 50 cm in length, 10 cm in diameter, and 5 mm thick), materials must be able to withstand, under normal conditions (at a temperature of 1,000 °C, over 10 years), thermal–mechanical stresses under irradiation (about 0.1 dpa/year). Catering for accident conditions requires resistance to heavy drop impacts, and mechanical loadings, or, further, to oxidation at high temperature (several hours at 800–1,600 °C). The types of composites which, at first blush, should best meet requirements are three-directional (3D) composites – a configuration that endows them with better dimensional stability – featuring a fiber base exhibiting a highly organized texture (close to that of graphite), allowing them to sustain better mechanical properties under irradiation. The main research effort is currently addressing the development of protective coatings against oxidation (see Figure 11), and, in the short term, irradiation and characterization campaigns covering fibers and mini-composites, to augment knowledge in this area.

(1) Guide tubes: in a PWR-type fuel assembly, these are tubes of dimensions comparable to those of fuel rods, allowing clusters of absorbent rods to slide, and fixed burnable poison clusters, or sealing clusters to be positioned.

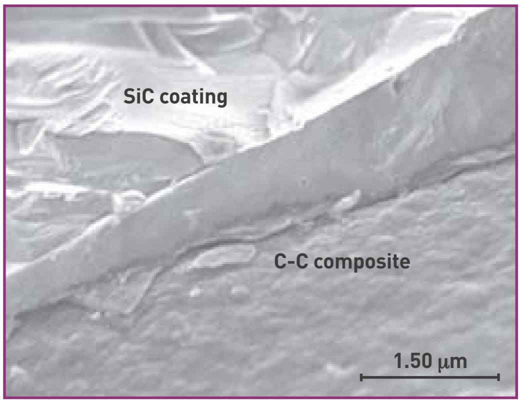


Figure 11.
C–C composite fabricated using the CEA calefaction process, coated with a silicon carbide layer.

SiC–SiC composite honeycomb for fuel claddings

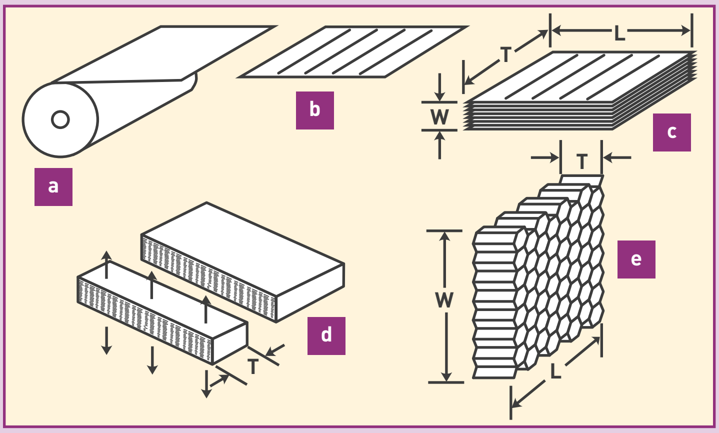
The two concepts being considered for fuel claddings, in **fast-neutron-spectrum**, **gas-cooled** reactors, were outlined in the preceding chapter. The first of these, the so-called *plate fuel* concept, is the most advantageous, allowing as it does higher **burnup**, and the individual confinement of each pellet. Claddings must ensure, at high temperature (~ 900 °C), under high dose conditions (15 dpa), and over extended periods (around 10 years), the functions of mechanical structuration (resistance to pressures higher than several hundred **MPa**), of solid and gaseous **fission product** containment (internal pressure possibly reaching some 100 bars), and heat removal (> 10 W/m · K).
Owing to the very low cladding thicknesses being sought, monolithic ceramics are virtually ruled out, and the

Carbon–carbon (C–C) composite used in the Tore Supra fusion machine, for the first (inner) wall, facing the plasma (the component is 1 m long).

Honeycomb fabrication through the expansion technique

For honeycomb fabrication, a roll of the cell wall base material (a), such as a fiber fabric, is cut into sheets, onto which lines of adhesive, or resin are laid down (b). The width of the resin lines matches that of a honeycomb wall; the interval between lines is three times wall thickness. These sheets are stacked (c), and the stack is heat treated, possibly under pressure, resulting in polymerization,[1] and bonding between sheets. The stack obtained is then sliced, to the thickness required (d). The slice is then stretched, until hexagonal cells are obtained (e). The fabric-based honeycomb may then be impregnated with resin, and polymerized. An organic-matrix composite honeycomb is thus obtained, which, with certain resins (phenolic resins), may be turned into carbon, through pyrolysis.

(1) Polymerization: the gradual adding of monomer molecules one to another, by means of covalent bonds (involving the sharing of electrons by two atoms), to form a polymer, i.e. a macromolecule of high molecular weight, featuring the repetition of the same pattern.



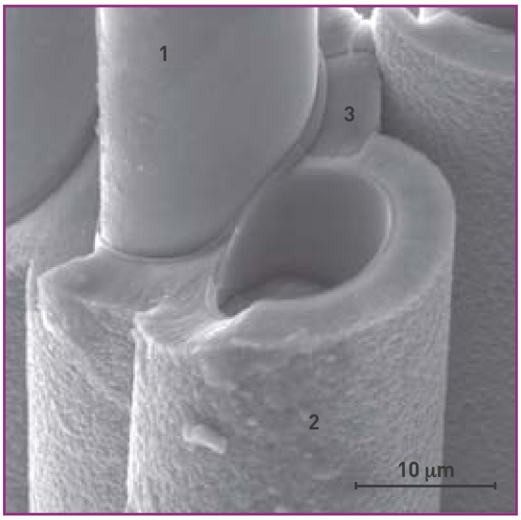
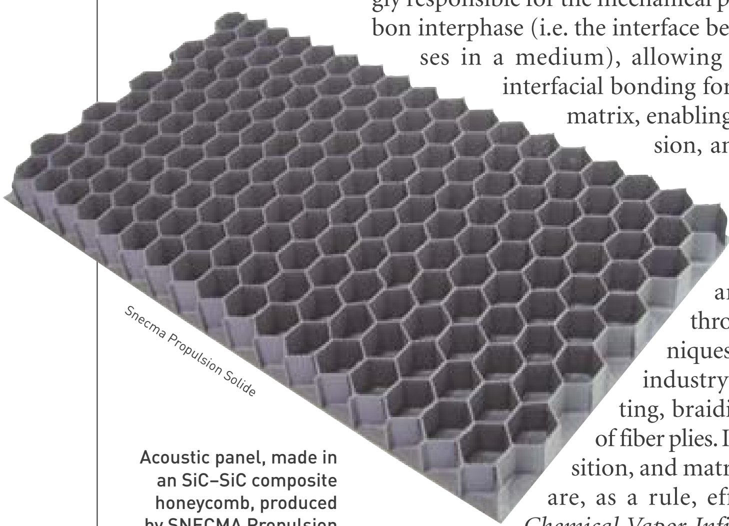


Figure 12.
SiC–SiC composites, comprising SiC fibers (1), a C interphase (2), and a SiC matrix (3) (Claude-Bernard-Lyon University/Laboratoire matériaux et interfaces).



Acoustic panel, made in an SiC–SiC composite honeycomb, produced by SNECMA Propulsion solide (cell size ~ 1 cm)

option naturally turns to composite ceramics, in particular of the SiC–SiC type, exhibiting as these do mechanical behavior, under high irradiation and at high temperature, that would seem to be satisfactory. These materials comprise SiC fibers, overwhelmingly responsible for the mechanical properties; a carbon interphase (i.e. the interface between two phases in a medium), allowing adjustment of interfacial bonding forces; and an SiC matrix, enabling material cohesion, and load transfer to the fibers to be ensured. The fiber reinforcements are fabricated through use of techniques from the textile industry: weaving, knitting, braiding, or needling of fiber plies. Interphase deposition, and matrix densification are, as a rule, effected through *Chemical Vapor Infiltration* of precursor compounds, inside furnaces at temperatures of around 1,000 °C. Such materials exhibit high performance, however they are restricted to high-tech applications, for aerospace, owing to their extremely high fabrication costs. Manufacturing processes are well mastered by a few manufacturers, and allow fabrication of large shaped components, however fabrication of claddings in the form of honeycomb plates, or of pins involves quite considerable difficulties (with very high geometric accuracies required, identical fiber contents and wall thicknesses imperative, low material thicknesses, extreme fiber stiffness – making for delicate handling – and the need to develop an interphase other than carbon, since the latter does "bear up" adequately under irradiation). The only composite ceramic honeycomb fabrication techniques developed, at present (by **EADS Astrium**, and SPS, in particular), involve fabrication, as a first step, of fiber structures, using either one of the two conventional techniques, of fabric expansion (see Box), or bonding (adhesive bonding, or other). Subsequent to this, impregnation with, and pyrolysis[2] of, an SiC precursor, and/or CVI[3] densification are carried out. Such techniques, however, are not compatible with the imperative specifications, since they do not allow a constant wall thickness to be obtained, or high geometric accuracy to be achieved. Investigations initiated at CEA's Le Ripault Center have allowed the development – through use of a novel process, subject to a registered patent – of materials that should meet the required dimensional characteristics. Mechanical trials, and irradiation tests (FUTURIX program) will make it possible to evaluate this solution.

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(2) Pyrolysis: chemical decomposition through the effect of heat.

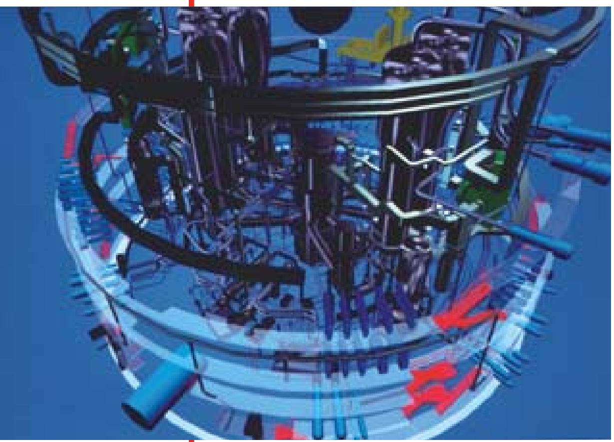
(3) CVI (Chemical Vapor Infiltration): a densification process, involving vapor infiltration into a fiber fabric, used for thermostructural composites, in particular carbon–carbon (C–C) composites.

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,



Areva NP

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

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).

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.

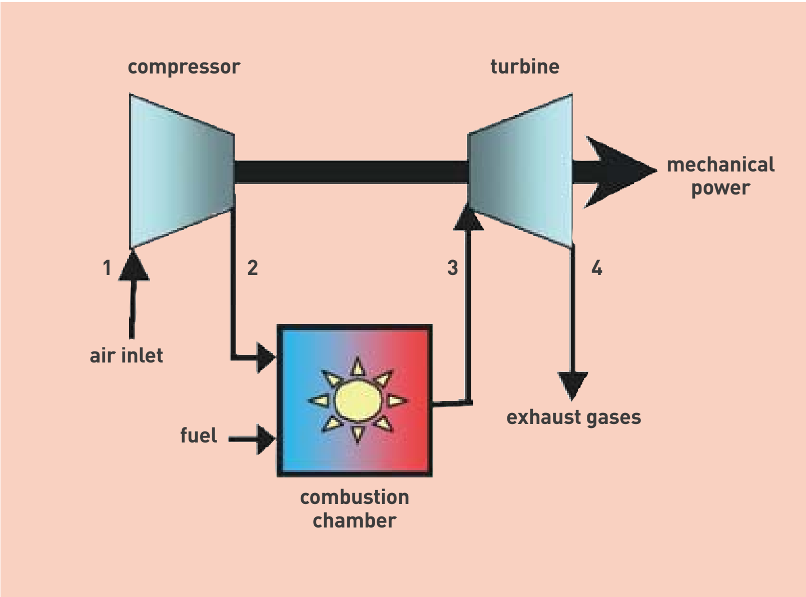


Figure.
Brayton cycle, as implemented in an open-cycle gas turbine.

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.

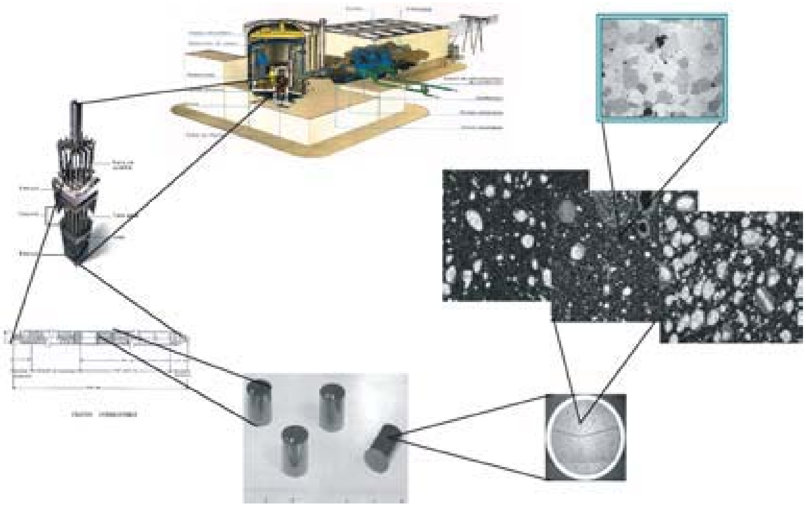


Figure.
Improving nuclear fuel reliability, and cost-effectiveness calls for finescale modeling of that fuel, through a multiscale approach, from reactor to fuel microstructure (in this instance, MOX fuel). Microstructural characteristics (porosity, cluster size and distribution, grain size...) have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

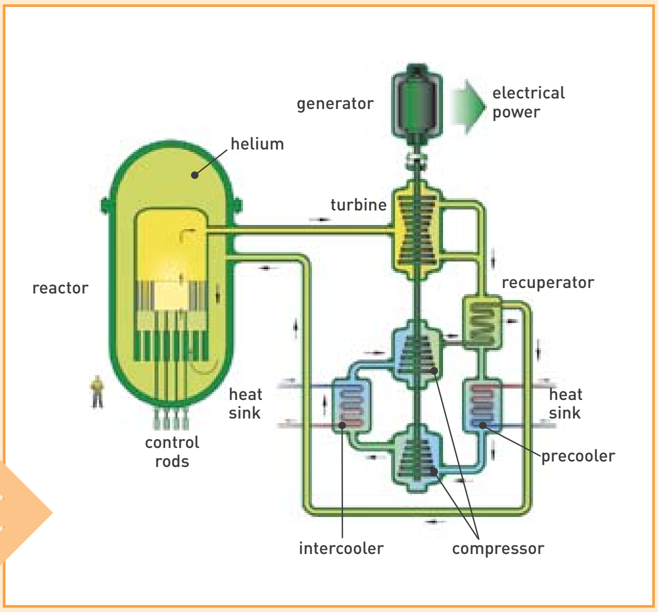
The six concepts selected by the Gen IV Forum

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

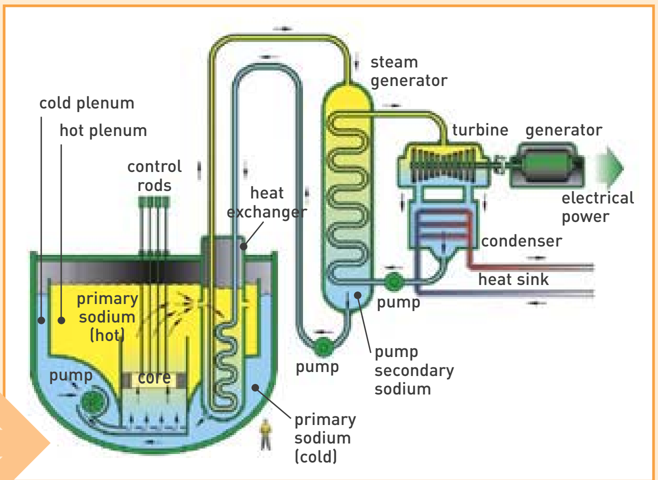
The gas-cooled fast reactor system (GFR) is a high-temperature, gas-cooled (helium-cooled as a rule), fast-neutron reactor allowing actinide recycle (homogeneous, or heterogeneous), while sustaining a breeding capability greater than unity. The reference concept is a helium-cooled, direct- or indirect-cycle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



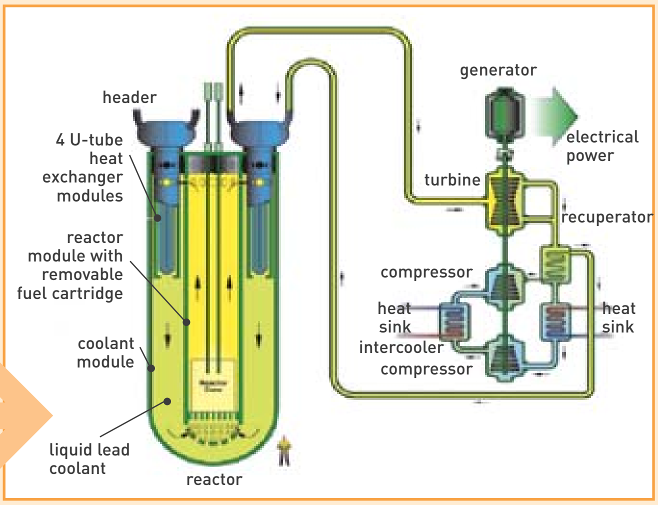
Le SFR

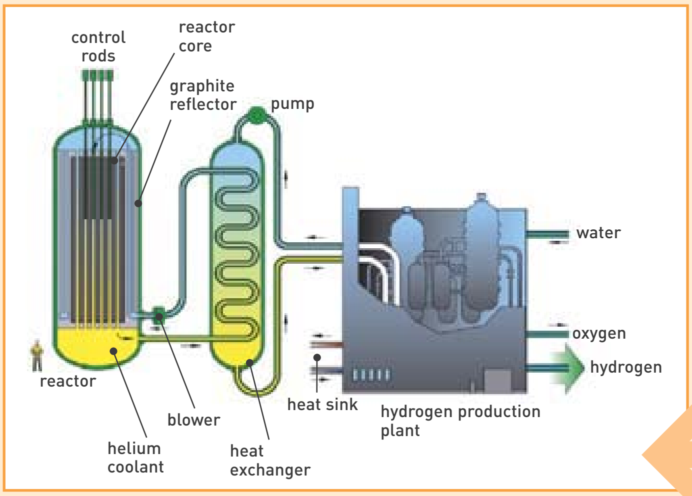
The sodium-cooled fast reactor system (SFR) is a liquid-sodium-cooled, fast-neutron reactor, associated to a closed cycle, allowing full actinide recycle, and plutonium breeding. Owing to its breeding of fissile material, this type of reactor may operate for highly extended periods without requiring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

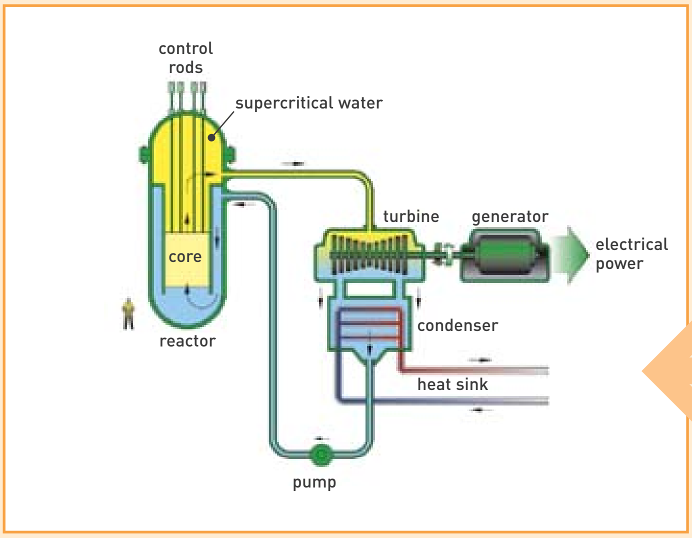
The lead-cooled fast reactor system (LFR) is a lead- (or lead-bismuth alloy-) cooled, fast-neutron reactor, associated to a closed fuel cycle, allowing optimum uranium utilization. A number of reference systems have been selected. Unit power ranges from the 50–100 MWe bracket, for so-called battery concepts,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





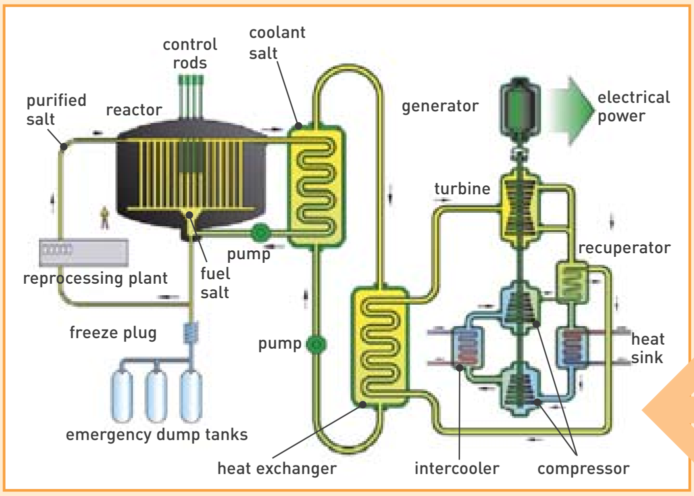
VHTR

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.