Gas-technology **energy conversion**: common ground for the new **fast reactors** and **(V)HTRS**

Whether it be for gas-cooled fast reactors, sodium-cooled fast reactors involving sodium–gas heat exchangers, or high-, or very-high-temperature reactors, gas-technology energy conversion is found to be all-pervasive in such fourth-generation systems.

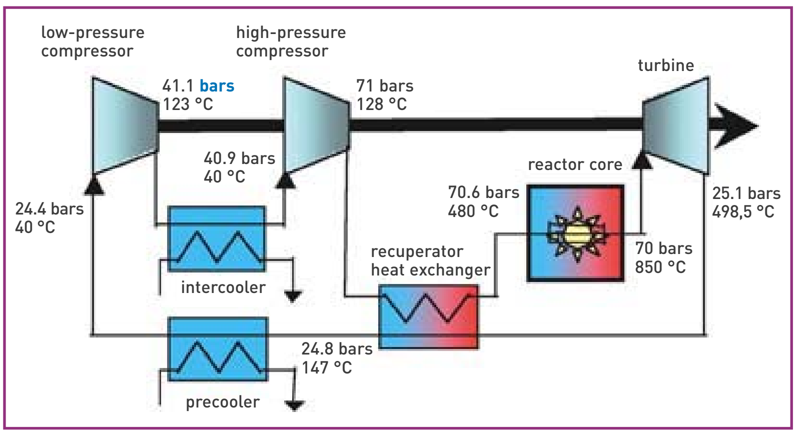
D. Michon-Artechnique/CEA

The CLAIRE loop, at CEA's Grenoble Center, allows the testing of high-temperature reactor components, heat exchangers in particular, in compressed air.

In the 1970s, CEA conducted a quite significant R&D effort to gain mastery of the technologies required to use **helium** as a primary **coolant**, and as an energy conversion fluid (see Focus C, ***Thermodynamic cycles and energy conversion***, p. 23). Some thirty years on, and subsequent to termination of that R&D effort in the 1980s, CEA's prime aim has been to set up new teams, to foster anew the expertise required for the development of **gas-cooled fast reactors** (**GFRs**). The second aim pursued by CEA was to provide support to its partners in industry, chiefly **Areva NP**, with regard to their high-, and very-high-temperature reactor (**HTR**–**VHTR**) projects (see *High-temperature reactors: a recent past, a near future*, p. 51). Such reactors are intended, in particular, for **cogeneration** purposes, i.e. for the combined production of electrical power, and heat for a variety of industrial processes, such as dedicated **hydrogen** production processes.

In 2001, the then most advanced reactor project, standing as reference for the resumed R&D drive, was **General Atomics**' GTMHR (Gas-Turbine Modular Helium Reactor). This thermal-spectrum, helium-cooled reactor project, with a power of 600 MWth, featured a so-called *direct* energy conversion cycle: helium, at 850 °C, from the core outlet being directly channeled to the turbine (see Figure 1). For that reason, initially, the GFR project CEA was bringing forward gave particular weight to the direct cycle. Such a cycle requires,

Figure 1.
An instance of a so-called direct energy conversion cycle, in a gas-cooled reactor.



if high thermodynamic efficiencies are to be achieved (close to 50%), very-high-efficiency components (i.e., components that minimize energy losses), chiefly the recuperator heat exchanger, and the helium **turbomachine**. The recuperator operates at around 500 °C, with a pressure differential of some 5 **MPa** between the two (high-, and low-pressure) heat exchange circuits, and a desired thermal efficiency of 95%. As regards the turbomachinery, the target isentropic efficiencies stood, respectively, at 93% for the turbine, and 89% for the compressors.

In 2003, US **DOE** launched the **NGNP** (New-Generation Nuclear Plant) program, for a 600 MWth very-high-temperature nuclear reactor (target core outlet temperature being 1,000 °C). This program resulted in a reappraisal of so-called indirect energy conversion cycles, in particular with the emergence of Areva NP's ANTARES project. One of its prime goals has been development of the gas-to-gas intermediate heat exchanger (IHX), transferring energy from the **primary circuit** to the **secondary circuit**, and the turbomachinery.

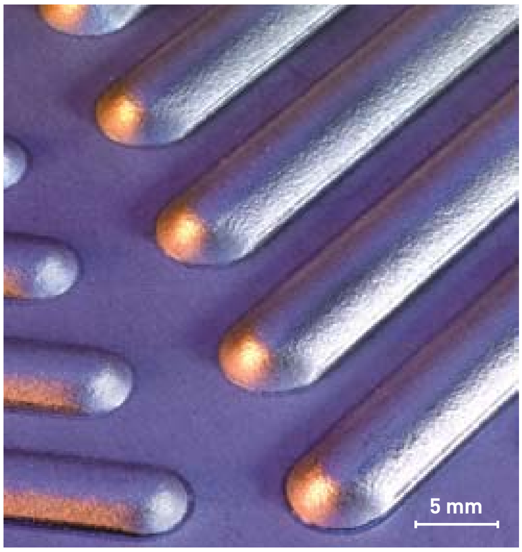
In 2005, CEA resumed R&D on sodium-cooled fast reactors (**SFRs**), looking for significant innovations, with respect to the Superphénix reactor, or the **EFR** (European Fast Reactor) project. In this context, gas-technology energy conversion (using helium, **supercritical** CO_2…) stands as an alternative involving a clean break with use of steam turbines. The sodium–gas intermediate heat exchanger then becomes a major component, requiring investigation.

The development work being carried out at CEA, by teams at the Cadarache, Grenoble, Saclay, and Valrhô centers, is pursuing a number of goals. First of all, these efforts seek to combine high temperatures (850–1,000 °C) with high pressures (7–10 MPa), while complying with the extant regulatory framework for pressure apparatus. They are also aimed at development of a technology of general relevance (sealing devices, thermal barriers, instrumentation), along with specific components (preheaters affording reliability at 950 °C, high-performance, gas-tight circulators, high-performance, compact heat exchangers reliable at temperatures ranging from 550 °C to 900 °C, helium–water heat exchangers [950 °C], fittings [500 °C], valves), while mastering the chemistry, and purification of the gas, or gases involved. The aim, finally, is to gain technical expertise, and operational experience with respect to test loops, and validate **modeling** using **computation codes**, such as the CATHARE operating and **transient** computation code.

The challenges of gas-technology energy conversion

One of the challenges set for the nuclear reactors of the future, be it for SFRs (see *Sodium-cooled fast reactors of the future*, p. 24), GFRs (see *Gas-cooled fast reactors*, p. 38), or VHTRs (see *High-temperature reactors: a recent past, a near future*, p. 51), is that of generating high-temperature heat, to achieve high efficiency (see Focus C, ***Thermodynamic cycles and energy conversion***, p. 23).

For a nuclear reactor, gas-technology energy conversion further allows considering the gas involved as coolant, for the reactor core. This is the so-called direct



D. Michon-Artechnique/CEA

Stamping tests with Inconel 617 sheet alloy, for SPHE-type heat exchanger plates.

cycle, as proposed for GHTMR, or PBMR (see *High-temperature reactors: a recent past, a near future*). If a different fluid is chosen as primary coolant, then the resulting conversion cycle will be of the so-called indirect type, the hot source for the cycle being provided by an intermediate heat exchanger (IHX). This is the case, obviously, for an SFR. It should however be noted that the indirect cycle still stands as the reference, whether it be for Areva NP's ANTARES (VHTR) project, or the 2,400 MWth GFR studied by CEA.

In the direct cycle case, helium is the obvious choice since, aside from its qualities in **neutronic** terms, it stands as one of the best coolant gases, allowing high temperatures to be achieved. For indirect cycles, aside from the qualities it exhibits for cooling purposes, the option selected for the secondary cycle must take into account further parameters, such as compatibility with the primary coolant, and turbomachinery technological maturity.

Gases being considered, in preliminary studies at CEA, are rare gases (helium, argon), nitrogen, carbon dioxide (possibly involving a high cycle point, above the critical point, for the so-called **supercritical** CO_2 cycle), and gas mixtures (e.g. helium–nitrogen).

To cater for the challenges arising with respect to energy conversion, and the innovations calling for investigation, thermodynamic cycle modeling and optimization tools have been developed. Taking the 2,400-MWth GFR cycle, by way of example, the helium primary circuit transfers all of its energy to a helium–nitrogen Brayton cycle, in the secondary circuit. The cold sink for this cycle may allow use of an altogether conventional, steam tertiary cycle (see Figure 1, in *Gas-cooled fast reactors*, p. 40).

For a given set of boundary conditions, optimization of such a cycle is a nonlinear, single- or multi-criterion problem (efficiency, energy loss, installation cost), heavily multivariable (choice of pressures and temperatures, as distributed across the installation), and multiconstraint (physical and technological restrictions), for which a deterministic algorithm soon shows its limitations. Use is made, consequently, of population evolution algorithms, such as *genetic algorithms*.

The findings from cycle studies form the starting point for design studies of key energy conversion components (heat exchangers, rotating machinery), these making it possible to confirm viability, and performance for the cycle considered, while yielding the models required by the codes used for safety studies (computations of incidental, and accidental transients, using CATHARE).

Heat exchangers as key components

Whether it be for gas-cooled reactors, or sodium-cooled reactors involving a gas turbine cycle, heat exchangers stand as key components. Their performance determines overall thermodynamic cycle efficiency. At the same time, catering for severe operating conditions, along with economic, and volume constraints entails the development of innovative solutions, compared to conventional tube technology. The investigations being proposed, in close collaboration with manufacturers, are mainly dictated by heat exchanger operating temperature levels:

– for applications at temperatures lower than 650 °C (gas-to-gas recuperator, sodium–gas heat exchanger), standard stainless **steels** (see Focus E, ***The main families of nuclear materials***, p. 76) prove compatible with the pressure and temperature conditions involved, and development efforts will only be required to adapt extant technologies to the specifications set out;

– for applications in the 650–900°C range (gas-to-gas intermediate heat exchanger), use of nickel-base **alloys** is required. Two candidate materials have been selected, for an extensive qualification program: Inconel 617, and Haynes 230. These alloys are not readily machinable, or weldable. Employing them, for the construction of compact heat exchangers, thus calls for basic investigations as to shaping, etching,(1) and assembly (welding, brazing), in order to validate fabrication processes prior to constructing technological test mock-ups.

In the longer term, and for higher-temperature applications (T_{gas} > 900 °C), novel materials will have to be qualified (involving structures of the **ODS**, **ceramic**, **cermet**… type), along with new assembly techniques. Employment of novel high-temperature materials, combined with the development of specific fabrication techniques (machining, bending, welding) stands as the main technological challenge, the more so since component lifetime specifications are stringent: 20 years for gas-to-gas heat exchangers (T > 800 °C), 60 years for sodium–gas heat exchangers (T< 650 °C).

Owing to the low exchange coefficient exhibited by gases (helium, gas mixture) as a whole, compared to liquids, large surface areas are required, if the desired power levels are to be exchanged. The second goal set for designers is the quest for an economically viable solution, entailing a reduction in capital costs. This is the reason that has led CEA, and its industrial partners to develop compact high-temperature technologies, allowing size reductions for components, and vessels, as shown in Table 1.

(1) In technical parlance, this term refers to the operation of machining (mechanically, or electrochemically) very small channels (about 1 mm in diameter).

exchange power: 600 MWth	technologie compacte	technologie à tubes
volume ⇒ power density	24 m^3 ⇒ 25 MW/m^3	1,500 m^3 ⇒ 0,4 MW/m^3
compactness	540 m^2/m^3	12 m^2/m^3
average exchange coefficient	1,300 W/m^2·K	800 W/m^2·K
number of heat exchanger vessels	1	3-4

Table 1.
Performance comparison, for gas-cooled reactors, of compact heat exchanger technology, and tube technology, for the same level of power exchange.

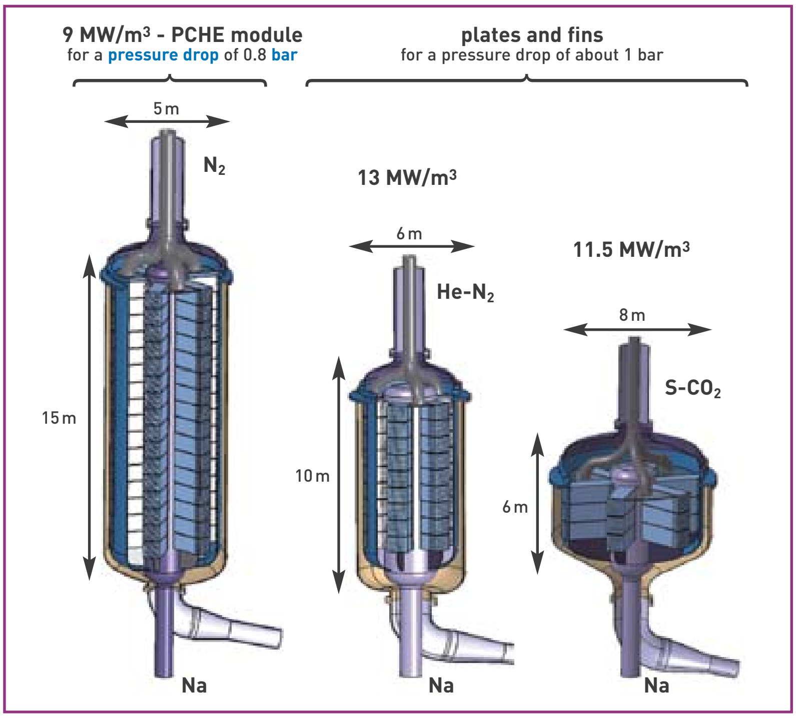
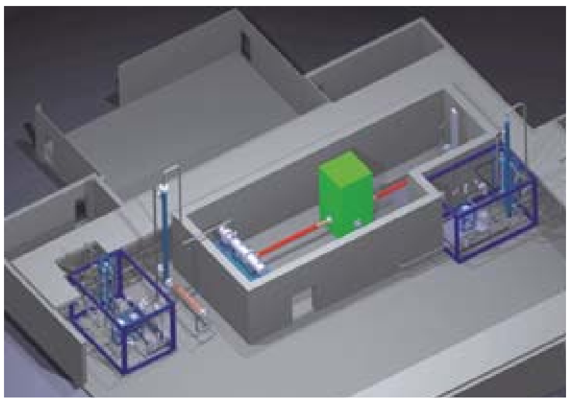


Figure 2.
Evidencing the impact sodium–gas heat exchanger technology, along with the nature of the gas used (nitrogen, helium–nitrogen mixture, or supercritical CO_2) may have on the size of a 400-MWth component, for a concept of the "pod-contained modules" type.


A. Gonin/CEA


CEA

Reception building for the HPC helium purification loop, and the future HELITE loop (HElium Loop for Innovative TEchnology, shown right), due to come on stream in 2008, at CEA's Cadarache Center.

Economic competitiveness involves further aspects. First of all, component thermal efficiency, which has a direct impact on overall thermodynamic cycle efficiency. The best tradeoff has to be found, between maximum efficiency, calling for extensive exchange surface areas, and thus a large size, and reasonable component size, involving a loss in efficiency assessed as acceptable (see Figure 2). Designers are thus aiming for efficiencies close to 90–95%. The second aspect to be taken on board is reactor availability, which may, obviously, be affected by component reliability (design and fabrication quality), but equally by the degree of modularity selected for the heat exchange component, this in turn bringing in the notion of inspectability, and repairability. Should designs go for smaller heat exchangers, more readily repairable, though involving higher overall capital costs, or larger components, proving more expensive to replace? For sodium–gas heat exchangers, one criterion for selection of the gas involved is its chemical reactivity with sodium, and steels used for the structures.

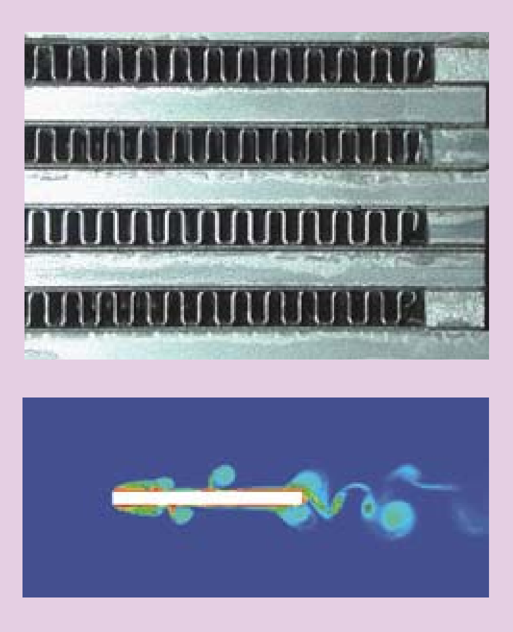


Figure 3.
An example of Nordon-technology PFHE heat exchanger; and simulation of the flow around a heat-exchanger fin.

Several types of plates being investigated

Compact heat exchanger technology is characterized by plate heat exchangers, featuring small fluid flow passage cross-sections (hydraulic diameter of around 1–2 mm, for gases). Two main technology families are being targeted by development work: welded etched plates; and brazed, or welded plates and fins (see Table 2). The most promising technologies, for gas-to-gas heat exchangers, taken in decreasing order of potential, are: stamped plates (SPHEs: stamped-plate heat exchangers), plates and fins (PFHEs: plate–fin heat exchangers), and etched plates (MPHEs: machined-plate heat exchangers; and PCHEs: printed-circuit heat exchangers) (see Box, and Figures 4 and 5).

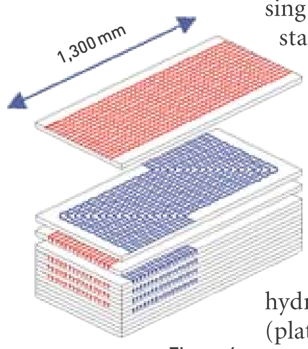


Figure 4.
Representation (CAD imagery) of the stacking of primary and secondary circuits in an MPHE heat exchanger, and of the exchange channel geometry.

R&D on gas-to-gas intermediate heat exchangers (850 °C), conducted in the context of Areva NP's ANTARES (600-MWth) project, has focused on the materials, dimensioning (gas-hydraulic, thermal, mechanical), and fabrication (plate etching, and welding) aspects.

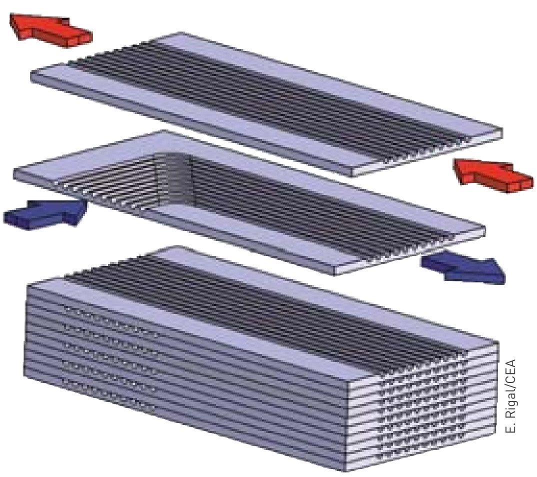
As regards selection of a metal, two candidate nickel-base alloys, Haynes 230 and Inconel 617, are the subject of an extensive qualification program, launched as a tripartite (**EDF**–CEA–Areva NP) endeavor. With respect to plate etching, two techniques have been subjected to a validation program, with both materials: high-speed machining (HSM), and electrochemical machining. With respect to assembly (welding), tests have shown these alloys prove difficult to join by diffusion welding, since they readily give rise to surface oxide formation, which is detrimental to joint mechanical strength. Specific surface preparation and degassing techniques have therefore been developed, to obtain a joint exhibiting adequate metallic, and mechanical properties. The welded plates form elementary blocks, which, when assembled, yield "exchanger modules." Altogether, 8 modules are held inside the heat exchanger vessel.

With respect to sodium–gas heat exchangers, the most promising technology is plate and fin technology. A number of evaluations have been carried out with both types of compact heat exchangers, involving various fluids, such as helium, or sodium, on the primary side of the exchanger, and nitrogen, supercritical CO_2, and the nitrogen–helium mixture (80%–20% by mass), on the secondary side. Figure 2 gives an idea – to some extent – of the impact heat exchanger technology, along with the nature of the fluid used, may have on the size of a given heat exchanger component, for a concept of the "pod-contained modules" type. This highlights the role played by

heat exchangers (HEs)		assembly	manufacturer - developer
compact etched-plate type	chemically etched PCHE : printed-circuit HE	diffusion welding	Heatric (UK)
	mechanically etched MPHE : machined-plate HE	diffusion welding	Areva-CEA
	stamped plates SPHE : stamped-plate HE	welding	Alfa-Laval Vicarb-Packinox CEA-Areva
compact plate-and-fintype	PFHE : plate-fin HE	brazing	Nordon (F)-Areva Brayton Energy (USA)
	FPHE : fin-plate HE	diffusion welding	Heatric (UK)

Table 2.
The two main families of compact heat exchanger technologies.

helium, in the mixture of 20% (by mass) He and 80% N_2: such a mixture allows a significant improvement, compared with N_2 used on its own, in thermal properties, resulting in smaller heat exchanger component size.
For the purposes of selecting a technology option, an initial stage involves constructing small compact heat exchanger mockups, and testing them in air, at low pressure (1–10 bar), on the 100 kW CLAIRE loop operated by GRETHE (Groupement pour la recherche sur les échangeurs thermiques: Heat Exchanger Research Group), at CEA's Grenoble Center, or directly with a representative fluid (N_2, He–N_2, He…), at high pressure (up to 90 bars), on the DIADEMO or HEDYT loops, at CEA's Cadarache Center. This makes it possible to identify possible manufacturers, to control their know-how, and characterize the technology involved, in steady regime, and during thermic **cyclings**. These tests further allow the validation of the exchange correlations used in the thermal–hydraulic dimensioning of such heat exchangers.
Recuperator qualification trials (type 316 steel, at 550 °C) have been carried out under European Union auspices (HTR–E project), covering two technologies, from Nordon, and Heatric, in air. They made it possible to confirm a compact heat exchanger technology could be put forward for the recuperator of a direct-cycle, 600-MWth gas-cooled reactor.
This stage, however, is not enough, since it is unrepresentative of in-reactor conditions. Tests with actual operating gases (helium), at high pressure, and high temperature, are indispensable, if fuller expertise is to be achieved. Subsequent to small-scale qualification work, allowing a selection of concepts, validation at a more representative scale is essential. This final stage may only be carried out on a higher-power helium loop (HELITE loop: 1 MW, 850 °C, 6–7 MPa, being readied at Cadarache). It is only at the outcome of this two-stage process that choice of a technology may be made.
There then remains the task of constructing a complete 10–20 MW elementary module, to obtain a full-scale subassembly (full-scale channel geometry, length, width, and height).



CAD view of a stack of corrugated plates, for an intermediate heat exchanger (IHX).

Three plate technologies for compact heat exchangers

SPHE technology
3D structuration of surfaces feasible.
Satisfactory level of stamped plate metallurgy.
Greatly improved heat exchanges, through **turbulence** involving a small pressure drop.
Strong innovation potential, in an industrial partnership context.
Low stamping costs.
Volume (600 MWth): 17 m^3, i.e. 35 MW/m^3.

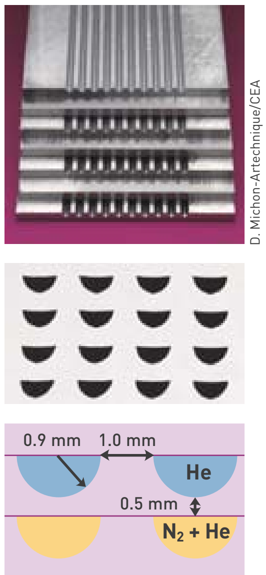
PFHE technology
Augmented surface areas, owing to fins → 1,500–4,000 m^2/m^3.
Greatly improved heat exchanges through turbulence, involving a higher pressure drop however (due to flow resistance encountered by the fluid).
Strong innovation potential with respect to fins.
Volume (600 MWth): 25–30 m^3, i.e. 20–24 MW/m^3.

MPHE technology
Size reduction (microstructures) → 500–1,000 m^2/m^3.
Slightly improved heat exchanges.
Strong issues arising as to fluid distribution.
Volume (600 MWth): 23 m^3, i.e. 26 MW/m^3.

Turbomachinery studies (compressor, turbine)

Fitting the whole picture together again, for fast reactor concepts, entails collecting the data required for the evaluation of incidental, and accidental transients, and obtaining estimates of capital costs.
The initial step involves the preliminary dimensioning of components. Turbine and compressor preliminary dimensioning modules have thus been developed, allowing confirmation of the feasibility, and size of such components, along with thermodynamic cycle performance.
With respect to reactor incidental/accidental transient analyses, the strong coupling must be taken on board, that prevails between energy conversion system, and reactor core (this being highly significant for direct cycles). Thus, the computation codes used to simulate the reactor must take into account all, or part of the energy conversion cycle, which is a significant difference, compared with design studies for water reactors. To meet this issue, of a mainly thermal–hydraulic character, a **multiscale approach** (from point model to 3D CFD(2)) has been implemented.
A turbomachinery initial, point model has thus been set up in the CATHARE system code, modeling the thermal–hydraulic functioning of a nuclear reactor. This allows a reasonable simulation of reactor behavior to be achieved, in steady as in transient regimes. The characteristic curves for each turbine, and each compressor are provided as input data, as obtained through preliminary dimensioning. A 1D axisymmetric model, developed for steady and unsteady regimes, allows a number of compressor and turbine characteristics (number of blades, size, etc.) to be taken on board. It further allows a slightly more local des-

(2) 3D CFD (3-dimensional computational fluid dynamics): a type of general-purpose fluid mechanics code (a so-called CFD code), used to solve coupled transport equations through discretization of the variables field, across as fine a mesh as required.



Stack of MPHE-type grooved plates, before and after assembly by diffusion welding.

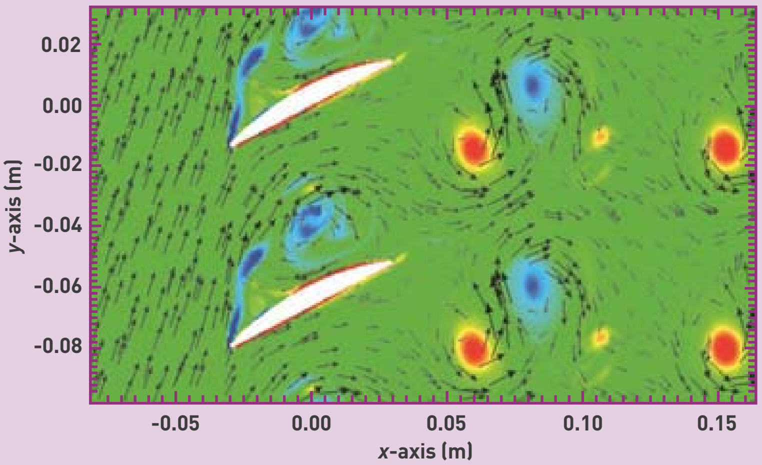


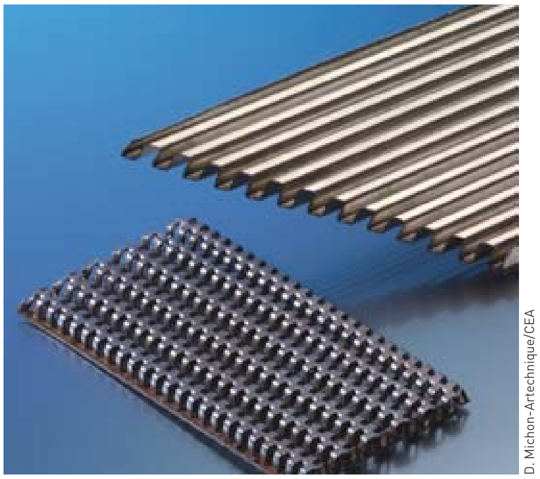
Figure 5.
3D computation of a compressor grid at high angle of incidence (vorticity field). The evolution, over space and time, of vorticity (component perpendicular to the circumferential section plane) is shown, superposed with the instantaneous velocity field, as obtained in large eddy simulation (LES). The reference frame is static, and flow is periodic, along the "vertical" direction.

cription of the turbomachinery component (actual fluid path, heat exchanges…) to be provided.
The 3D CFD approach makes it possible to determine the correlations to be used in the more global models, and, ultimately, to locate hot points on turbine blades, and discs, and to achieve a better estimate of engine efficiency, through the local information thus obtained (see Figure 5). Such physical and numerical investigations have to be validated by experimental data. An analytical program would thus need to be undertaken, in the event of the direct cycle ultimately being the chosen option.
So-called *system* tests (tests involving an assembly comprising several components) are also required, in particular with regard to the REDT (Réacteur expérimental de développement technologique) reactor, which should be the first gas-cooled, **fast-neutron** installation. Such tests would allow the various strategies for **decay heat** removal from the core to be evaluated (incidental/accidental transients). This is the purpose assigned to the SALSA facility, the preliminary design study for which was completed in 2006. Dedicated validation loops are also being developed,

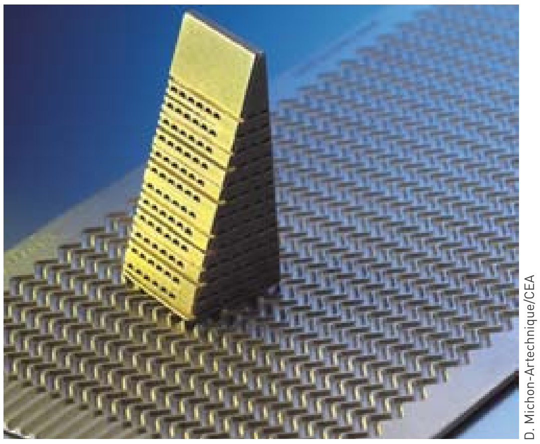
parameters	test loop	reactor
flow rate	$e \cdot Qm$	Qm
rotation speed	$\omega \cdot 1/\sqrt{e}$	ω
size	$e^{1/2} \cdot L$	L

parameters	test loop (20 MW)	reactor 600 MW
number of stages	12	12
rotation speed	15,000	3,000
helium flow rate (kg/s)	12.7	317.5
turbine power (MW)	22.0	558.0
inlet pressure (bar)	70.7	70.7
outlet pressure (bar)	26.4	26.4
inlet temperature (°C)	848	848
outlet temperature (°C)	510	510
turbine lenght (m)	0.4	2
average disc radius (m)	0.17	0.85
average blade height (m)	0.028	0.14
average Reynolds number	~ 210,000	~ 110,000

Tables 3 and 4.
Turbine similarity parameters and main characteristics, as compared between test loop, and reactor.



Straight-finned and Nordon-technology staggered-fin plates.



Chemically etched, diffusion-welded plates (Heatric technology). The component is about 5 cm tall.

or operated in other countries (Japan, South Africa, United States). CEA has taken part in an international model validation exercise, carried out in a South African facility, under **IAEA** auspices. The effort conducted by German teams to recover, and reassess findings and investigation work from previous decades should also be mentioned.
As regards the direct cycle case, CEA has looked into the tests required to achieve technological qualification of the helium turbomachine (materials, design, fabrication). Such qualification entails preserving, without undue distortion, certain gas-hydraulic characteristics of helium machinery, such as high **Reynolds number**(3) (Re < 200,000), and fairly low **Mach number**(4) (M < 0.4). Investigations carried out at CEA, with assistance from **Institut Von Karman** (Belgium), have found that the best "efficiency cost/effectiveness" ratio arises for turbines of around 20 MWth power (Q_{He} = 10 kg/s), i.e. at a 1/5 scale for length, 1/25 in terms of power, taking as reference the turbine for a 600-MWth reactor (see Tables 3 and 4).

(3) Reynolds number: the ratio of aerodynamic forces, over viscosity forces, for a given flow. This allows the computation of boundary layer characteristics, and the characteristics of boundary layer resistance to separation.
(4) Mach number: a dimensionless number, indicating the ratio of the local velocity of a fluid, over the speed of sound in that same fluid.

Mockup of a Nordon-technology heat exchanger for a very-high-temperature reactor, as tested on the CLAIRE loop, at Cadarache. The thermocouples may be seen (in green).

Circulator development

In REDT, or in GFRs, primary helium must circulate in normal operating conditions, but equally in accident conditions. It should be noted that gas-technology pumping is likewise a crucial component in currently operated, or planned test facilities. The compressors used in such conditions are commonly known as "circulators." It is the impeller of these turbomachines that imparts to the fluid the mechanical energy required for its motion.
The nature of the gas involved, and circuit characteristics (**pressure drop**, flow rate) dictate selection of the most appropriate compression stage to ensure circulation. Stage efficiency has a direct impact on the energy cost of pumping operation, and on the heating experienced by the gas as it passes through the compression stages. This heat input must be taken into account when designing components lying downstream from the compressor.
As with energy conversion turbomachinery, preliminary dimensioning models have been developed. This was the case for the main circulator, and emergency system circulators, for the initial general review of the 2,400-MWth GFR concept. Characteristic curves have thus been determined for operating conditions, to be included in the CATHARE code.
Tests were found to be required, to resolve the technical uncertainties involved in the technologies selected for the design of test loop circulators. Along with performance validation for helium operation, the tests made it possible to reach favorable conclusions, regarding the behavior of the bearings and motor kept in gas, inside the vessel.
It must be emphasized that all the development work being conducted in the context of gas-cooled reactor programs is being very closely watched by thermics specialists active in the **fusion** field, since helium is likewise being considered as a coolant for fusion reactors.

A compressor, featuring peripheral stages and a rotating assembly operating in gas, being readied for tests.

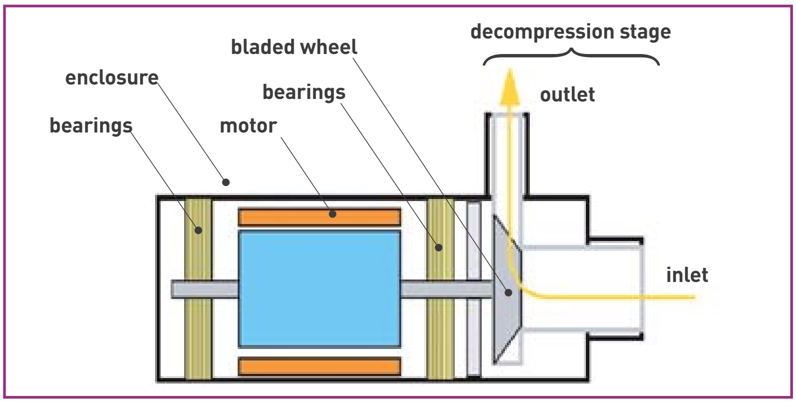


Figure 6.
Typical structure of a compressor featuring a centrifugal stage, and a rotating assembly operating in gas.

A sound construction expertise

CEA has embarked on an extensive R&D program, intended to provide support for the design of fourth-generation nuclear reactors. Helium technology may be seen as common ground for the development of such innovative systems, be they gas-, or sodium-cooled. Since 2001, revamped teams have been at work on these issues, and a sound expertise is being built up, over the years. For every technological innovation, the keys for success entail ascertaining safer operation is ensured, taking on board the concept of sustainable development, of more industrial-type construction – for competition, as regards such reactors for the latter half of the 21st century, is on a global scale – and, finally, greater ease of operation, this covering greater component reliability, and the taking on board, from the design stage, of inspectability, repairability, and constraints relating to the cycle back end (waste reprocessing pathway, storage, transportation).
One point should be particularly noted, and that is the lifetime specification set out for these systems, to wit 60 years. Such a requirement implies a different outlook for designers, since solutions no longer entail systematically going for performance in terms of temperature, or efficiency. In such a context, the key lies in the quest for higher-performance, long-lasting materials.

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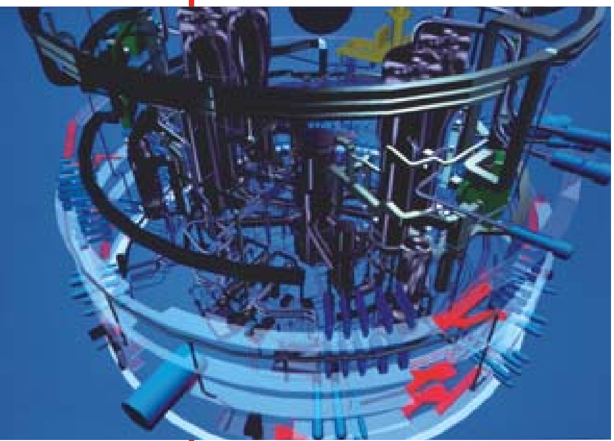
FOCUS A

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.

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.

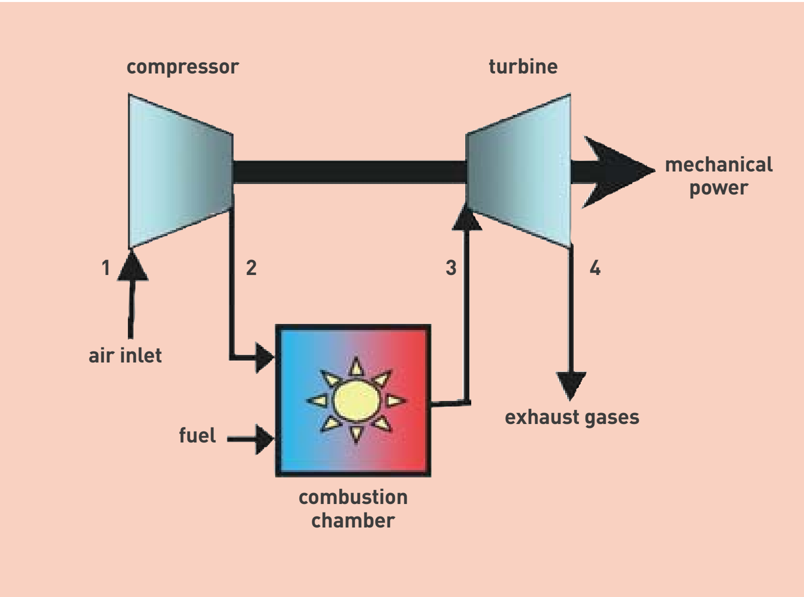


Figure.
Brayton cycle, as implemented in an open-cycle gas turbine.

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.

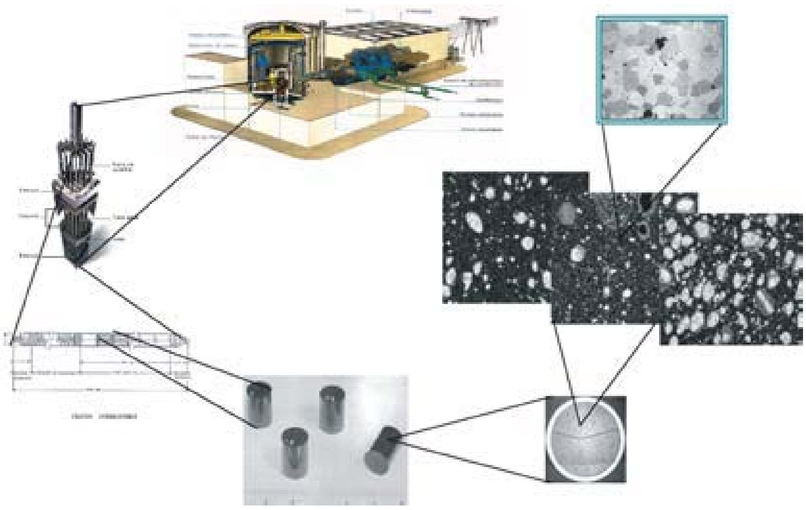


Figure.
Improving nuclear fuel reliability, and cost-effectiveness calls for finescale modeling of that fuel, through a multiscale approach, from reactor to fuel microstructure (in this instance, MOX fuel). Microstructural characteristics (porosity, cluster size and distribution, grain size...) have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

FOCUS E

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.

(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.

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,

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.

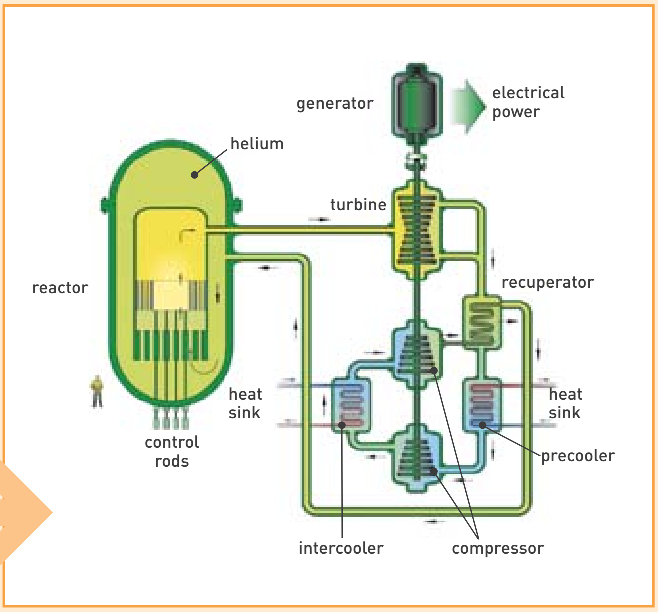
The six concepts selected by the Gen IV Forum

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

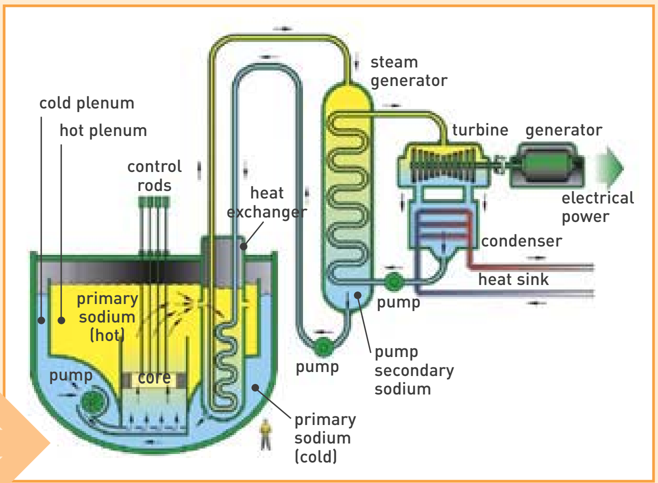
The gas-cooled fast reactor system (GFR) is a high-temperature, gas-cooled (helium-cooled as a rule), fast-neutron reactor allowing actinide recycle (homogeneous, or heterogeneous), while sustaining a breeding capability greater than unity. The reference concept is a helium-cooled, direct- or indirect-cycle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



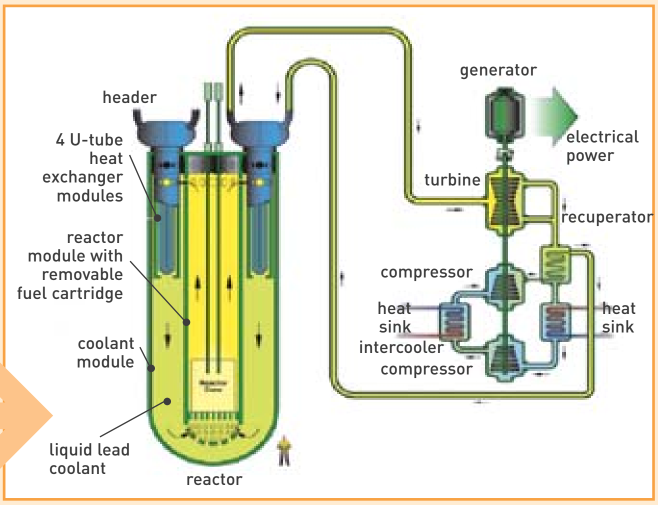
Le SFR

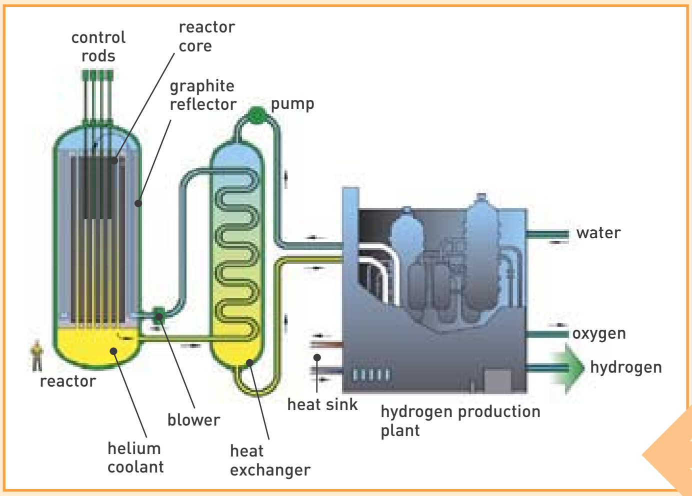
The sodium-cooled fast reactor system (SFR) is a liquid-sodium-cooled, fast-neutron reactor, associated to a closed cycle, allowing full actinide recycle, and plutonium breeding. Owing to its breeding of fissile material, this type of reactor may operate for highly extended periods without requiring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

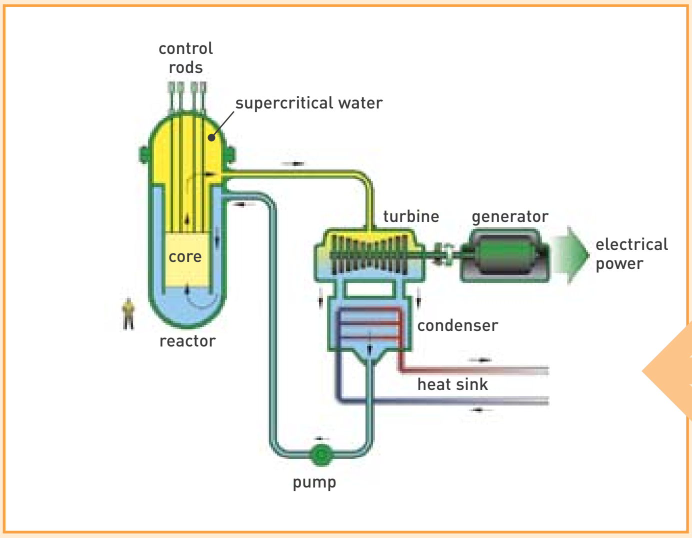
The lead-cooled fast reactor system (LFR) is a lead- (or lead-bismuth alloy-) cooled, fast-neutron reactor, associated to a closed fuel cycle, allowing optimum uranium utilization. A number of reference systems have been selected. Unit power ranges from the 50–100 MWe bracket, for so-called battery concepts,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





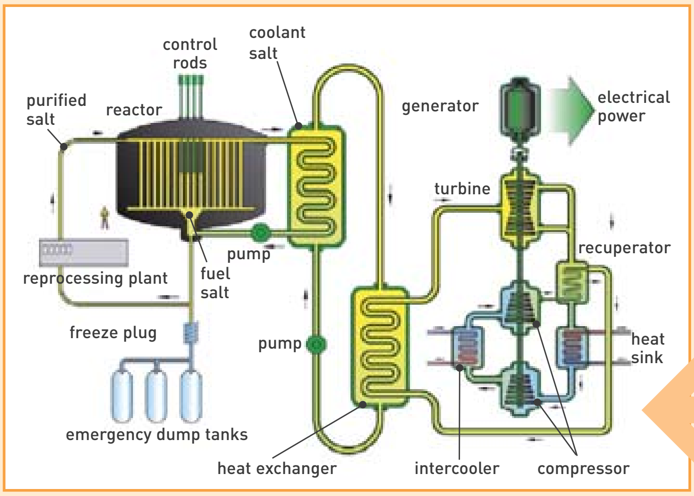
VHTR

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.