Gas-cooled fast reactors

The fast reactor using gas as a coolant (GFR) affords an extremely attractive path, for reactors that could emerge around the middle of the century. Employment of helium does still call for some development work, and definition of the fuel for such a system will keep researchers busy for some years, since it must exhibit properties that are clearly at a remove from those of current fuels.

The helium tribometer, set up at the CEA Cadarache Center, developed for the investigation of friction and wear of materials, under conditions most representative of GFRs. It has the ability to operate up to 1,000 °C in helium, with or without presence of steam and/or impurities, and is capable of exerting a contact pressure of 20 MPa.

A. Gonin/CEA

The principle of **nuclear reactor** operation using **fast neutrons**, and a gas **coolant** harks back to the European GBR4 (Gas-Cooled Breeder Reactor 4) and US GCFR (Gas-Cooled Fast Reactor) projects, which attracted, right up to the 1980s, sufficient interest to warrant substantial research programs being carried out. The notion then was to draw on the technological experience gained with **generation-I and generation-II gas-cooled reactors** (see Focus A, ***The components of a nuclear system***, p. 10; Focus B, ***Reactor lines, generations, and neutron spectra***, p. 14). Achieving "high temperature" was not a goal, and designers elected to carry over the **fuel** that had been successfully developed for the **sodium**-cooled **fast reactor** (SFR). The lead the latter concept soon took over the field, together with the downturn in research on reactors for the future resulted in such design studies being shelved prematurely, no demonstrator ever being built.

Combining the benefits of fast spectrum, and high temperatures

The **fourth-generation gas-cooled fast reactor** (**GFR**) takes in the results thus achieved, however it is essentially driven by revamped specifications (see Box, *The six concepts selected by the Gen IV Forum*, p. 6). Its aim is to combine the advantages of the high-temperature reactor (HTR) with those of the SFR (see *High-temperature reactors: a recent past, a near future*, p. 51). *High temperature* allows high-performance **energy conversion cycles** to be considered, with electricity generation efficiencies higher than 45%, while opening up new applications for nuclear energy, such as process heat production. *Fast neutrons* afford the ability, for **fission** nuclear energy, to comply with sustainable development goals and prospects, through the sparing management of natural resources, and minimization of ultimate waste.

In particular, GFR designers are banking on the viability of a novel fuel,(1) exhibiting, with respect to safety, inherent qualities, lacking in the more conventional fuel (**oxide pellets**, stacked in a fuel **rod**) used in the GBR4 and GCFR projects. This fuel must feature high **thermal conductivity**, allowing effective removal of the power generated inside it, and refractory **cladding**, with the ability to ensure containment up to a very high temperature.

The specific characteristic of the fourth-generation GFR will be its use of innovative technologies, to be brought in in accordance with a still-evolving design approach. Anticipated benefits from the use of **helium** lie, essentially, in its chemical inertness, its "transparency" to **neutrons** (no slowing down of, or **activation** by, neutrons), and optical transparency, allowing

(1) J.-C. Garnier *et al.*, "Contribution to GFR design option selection", ICAPP '06, Reno, United States, 4–8 June 2006.

temperature remote measurement systems to be contemplated, using optical, or sighting pyrometers,[(2)] together with visual observation systems, for handling and in-service maintenance operations.

The main design options

An initial stage was completed at the end of 2005, in the guise of a survey of the concept as a whole, fitting the individual options together again.[(3)] The major structuring design choices, for the GFR, are as follows:
• helium pressurized to 7 **MPa** as primary coolant;
• helium temperature at **core** outlet of 850 °C, to achieve high energy conversion efficiencies, whichever **thermodynamic cycle** is adopted (see Focus C, ***Thermodynamic cycles and energy conversion***, p. 23);
• high reactor power, with a capacity to deliver 2,400 **MWth**;
• high core **power density**, standing at around 100 **MW/m³**, to minimize **plutonium** requirements;
• a zero-breeding-gain core, i.e. a core having the ability, in operating conditions, to generate as much **fissile** material as it consumes;
• reactor layout of the **loop** type (as opposed to **integral** designs);
• a so-called **indirect combined** energy conversion **cycle**, allowing high efficiencies, while making use of existing technology;
• reactor **decay heat** removal based on gas circulation in all circumstances, using low-power pumping systems, or even **natural convection**.

The reactor core

The selected reference fuel element takes the form of a flat **plate**. The cladding – the first containment **barrier** – is made of a reinforced **ceramic** (a silicon carbide **composite**); the fissile phase consists of a mixed **uranium** (**U**)–**plutonium** (**Pu**)–**minor actinide** (**MA**) **carbide** (see *What fuel for GFRs?* p. 45).
The plate elements are positioned vertically, to form an **assembly**. The hexagonal geometry of the latter makes for optimal core stability, and mechanical equilibrium. The channels through which helium circulates have a rectangular section. Closure of the assembly, in the **thermal–hydraulic** sense, is effected by the outer casing (wrapper tube).
Core characteristics and performance (see Table 1) are the outcome of an optimization approach, aimed at meeting to the best the individual specification stipulations.
Breeding gain expresses the difference between production, and consumption of fissile **atoms**. In the present case, it is slightly negative. This is a so-called *first-cycle* value, corresponding to a core holding no minor actinides. As **multirecycling** of **actinides** (U, Pu, MAs) proceeds, fuel composition and **isotopy** vary, tending to equilibrium values. Breeding gain rises, to a value close to zero, and the core reaches a zero-breeding-gain regime. The "reactivity coefficients" are characteristics inherently making for safety. The helium

(2) Optical and sighting pyrometers: these devices, used for high-temperature – higher than 600 °C – measurements, measure energy emitted in the **infrared**.

(3) P. Martin, N. Chauvin, "Gas-cooled fast reactor system: major objectives and options for reactor, fuel and fuel cycle", GLOBAL 2005, Tsukuba, Japan, 9–13 October 2005.

core height / diameter (m)	1.55 / 4.44
number of fuel assemblies in the core	387
number of fuel elements per assembly	27
plate fuel element thickness (mm)	7
(U,Pu)C / helium volume fractions (%)	22.4 / 40
operating pressure (MPa)	7
core inlet / outlet temperature (°C)	400 / 850
core **pressure drop** (MPa)	0.044
maximum fuel temperature (°C)	1,260
Pu **inventory** (t/**GWe**)	8.2
breeding gain*	- 0.07 / - 0.04
Doppler coefficient* (pcm)	- 1,872 / - 1,175
helium void coefficient* (pcm)	212 / 253
delayed neutron fraction* (pcm)	388 / 344

* Values at start, and end of **cycle**.

Table 1.
Characteristics and performance of the GFR core, as of end 2005.

void coefficient induces an increase in **reactivity**, this remaining lower than the margin yielded by delayed neutrons. The **Doppler coefficient** is particularly large for a fast reactor, resulting in a markedly stabilizing effect (see Table 2).

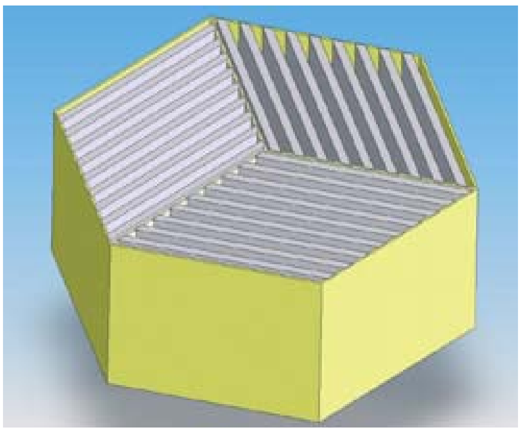
reactor type	delayed neutron fraction = 1 **$**	coolant void coefficient	Doppler coefficient
GFR	~ 350 pcm	+ 0.5—+ 1 $	- 3—- 5 $
SFR	~ 350 pcm	+ 4—+ 5 $	- 2—- 2.5 $

Table 2.
Comparison of safety characteristics for gas- and sodium-cooled fast reactor cores.

These findings were obtained through use of CEA's ERANOS **computation code**.

Energy conversion

The indirect combined cycle option stands as a credible alternative to the **direct cycle**, that had been selected in the initial phase of GFR design studies. Compared to the latter, it affords attractive characteristics. On the one hand, the energy transfer system for the **primary circuit** is more compact, compared to the direct-cycle energy conversion system, which is exceedingly bulky, especially for a high-power reactor. The general layout of the primary circuit is thus easier to arrive at. And, on the other hand, the indirect combined cycle involves

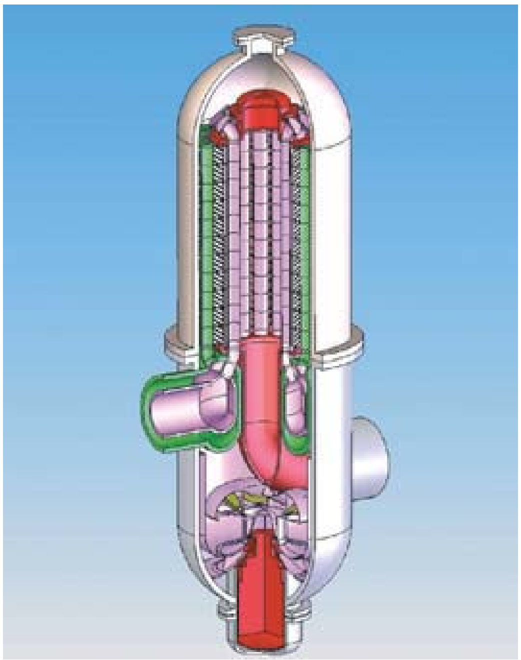


CEA

GFR hexagonal fuel assembly, with the fuel elements, in the form of plates, stacked in rhombic (diamond-shaped) subassemblies.


Artechnique/CEA

The CLAIRE loop, set up at the CEA Grenoble Center, is used in particular for gas-to-gas heat exchanger tests.


CEA

The vessel enclosing the intermediate heat exchanger (IHX) and circulator.

an optimum thermodynamic efficiency that is less sensitive to core inlet temperature, allowing a design to be considered, for the **vessel**–reactor unit, involving no extensive thermal insulation features. This cycle relies on a novel **heat exchanger** technology, development work for which is ongoing under the aegis of the ANTARES (Areva New Technology based on Advanced gas-cooled Reactor for Energy Supply) HTR project, steered by **Areva NP** (see *Gas-technology energy conversion: common ground for the new fast reactors and [V]HTRs*, p. 91; and *High-temperature reactors: a recent past, a near future,* p. 51).

In the energy conversion system opted for (see Figure 1), the primary circuit fluid, helium (He), transfers its energy, by way of an intermediate heat exchanger (IHX), to a **secondary circuit**, using gas – a helium–nitrogen ($He–N_2$) mixture – and comprising a turbine, a **steam generator**, and a compressor. The steam yielded by the steam generator is used in a conventional steam cycle. Electrical energy is generated partly by the secondary circuit gas turbine, and partly by steam turbines mounted in the **tertiary circuit**. An initial evaluation of net efficiency suggests a value of 45.1%. Aside from the IHX, all components are conventional designs. If use of a water–**supercritical** steam cycle is contemplated, combined with improved optimization of the distribution of electricity generation across the secondary and tertiary circuits, it would be feasible to achieve an efficiency close to that for the direct cycle (~ 48%).

Preliminary design studies have mainly focused on the primary circuit components, namely the reactor vessel, the heat exchangers between the primary and secondary circuits (IHXs), and circulators. It has been decided to go for three primary loops (3 × 800 MWth), each fitted with one IHX–circulator unit, enclosed in a single vessel. The reactor vessel is a metallic structure, of great size (inner diameter of 7.3 m) and thickness (20 cm). The material selected, ensuring as it does negligible **creep** at operating temperature, i.e. 400 °C, is **martensitic** 9Cr1Mo **steel** (industrial grade T91, containing 9% by mass chromium, and 1% by mass molybdenum).

Fitting the whole picture together again

The reactor vessel, together with two of the decay heat removal loops, and one of the IHX–circulator unit, is shown in Figure 2a, which further indicates the design principles selected, as regards the **control rods** (with mechanisms located at the bottom of the vessel), and fuel handling (articulated arm system; fuel element loading and unloading *via* lock chambers; sealed vessel). The entire primary circuit is confined within a metallic guard containment structure, of spherical geometry, with a diameter of about 30 m (see Figure 2b). Being of moderate thickness (38 mm),

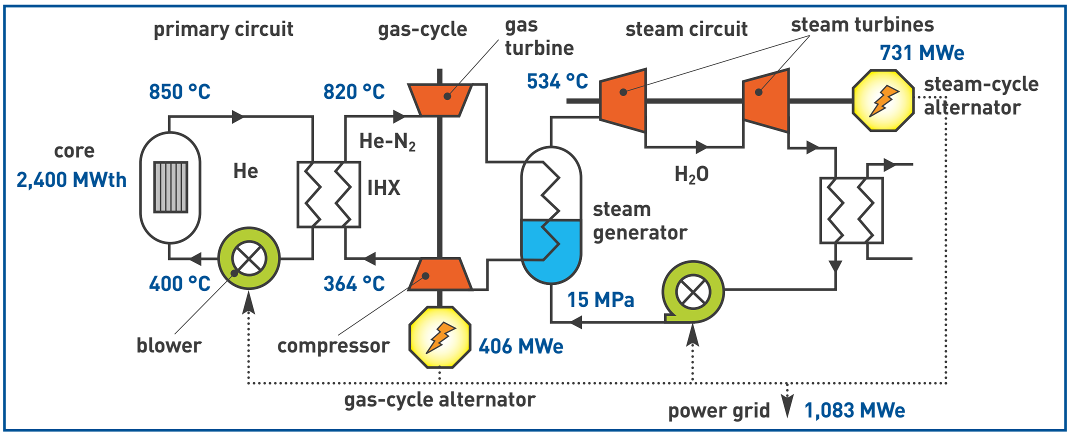


Figure 1.
GFR energy conversion installation, using an indirect combined cycle.

this structure requires no complex operations, such as stress relief measures,[4] when erected on site.
Following the preliminary design studies covering all reactor systems and components, a numerical **model** was drawn up, using the CEA CATHARE code. **Transients** of the blackout type (complete loss of electrical power supply), and **breaches** have been calculated. These initial findings confirm the overall consistency of the options selected, and corroborate the preliminary calculations carried out at the preconceptual design stage.[5]

The safety approach

The technical targets set for the GFR, as indeed for the other systems selected by the **Generation IV International Forum** (see Box, *The six concepts selected by the Gen IV Forum*, p. 6), are consistent as a whole with those drawn up by the French Nuclear Safety Authority (**Autorité de sûreté nucléaire**):[6] control of nuclear and chemical reactions, of removal of the energy generated, of hazardous product containment, of personnel protection, and of **effluents**, and waste, to ensure the protection of populations, and the environment.
These functional targets are complemented by further targets, of a probabilistic character. By way of example, safety specialists have set an overall probability (i.e. taking on board both internal and external initiators) of core degradation, accompanied by significant release of **fission products**, of about 10^{-5} per reactor, per year; and a probability of significant release beyond the final barrier standing at some 10^{-6} per reactor, per year. The sole purpose of such figures is to assist the designer in selecting, in a relevant manner, the number of functional redundancies that should be employed, if the above-mentioned safety targets are to be met.
The list of accident initiators, relative to a new reactor project, depends on the level of detail the designer has reached, in the reactor design, further relying on an intuitive understanding of the resulting scenarios. Initiators themselves are classed according to their frequency of occurrence, which it should be said is not readily determined, for reactor projects with no operational feedback to draw on. An initial list, with no claim to exhaustiveness, but relating to decay heat removal, of major initiators leading to an abnormal rise in fuel temperature is set out in Figure 3. Safeguards against the depressurization accident will be detailed by way of example.

Safeguard systems

The safeguard systems suggested must meet the technical safety targets set for a nuclear reactor. The principle behind the solutions selected at the present stage is indicated, for each one.
For the purposes of controlling containment, the three barriers principle is implemented: **cladding** at the fuel level; vessels and primary circuit; containment building or vault. Opting for a guard containment structure (the sphere in Figure 2b), to maintain gas pressure, forms a fourth barrier, affording further potential guarantees.
The measures taken to control reactivity involve an assembly design guaranteeing core geometrical stability, when under power, a fuel exhibiting favorable natural behavior, together with stabilizing **neutronic feedback**, and, finally, two diversified, redundant control rod systems, actuated according to a gravity drop principle.
As regards controlling heat removal, two independent, individually redundant decay heat removal systems are employed. One, a passive system, operates by means of natural convection, while the second one, an active system, uses circulators having the capability to maintain a minimum gas flow through the core.

(4) Stress relieving: use of a heat treatment, for the purposes of reducing internal stresses, involving no appreciable structural alteration.

(5) P. Dumaz *et al.*, "The thermal–hydraulic studies in support of the GFR pre-conceptual design", NURETH-11, Avignon, France, 2–6 October 2005.

(6) M. Lavérie, *Clefs CEA*, No. 45 (Fall 2001).

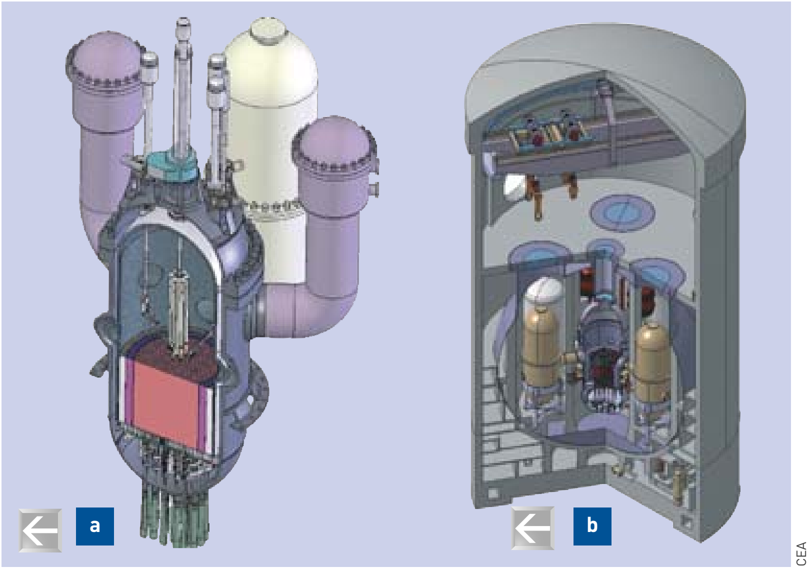


Figure 2.
Shown at a, the reactor vessel, with (purple) two of the decay heat removal loops, and (white) one of the heat exchanger (IHX)–circulator unit. At b, cutaway view of the reactor building, showing the metallic, spherical-geometry guard containment structure, inside which the entire primary circuit is enclosed.

Optimizing the helium purification process on the CIGNE chemical reactor. Helium quality is of paramount importance, since the impurities in this coolant gas determine its corrosive potential with respect to structural materials.

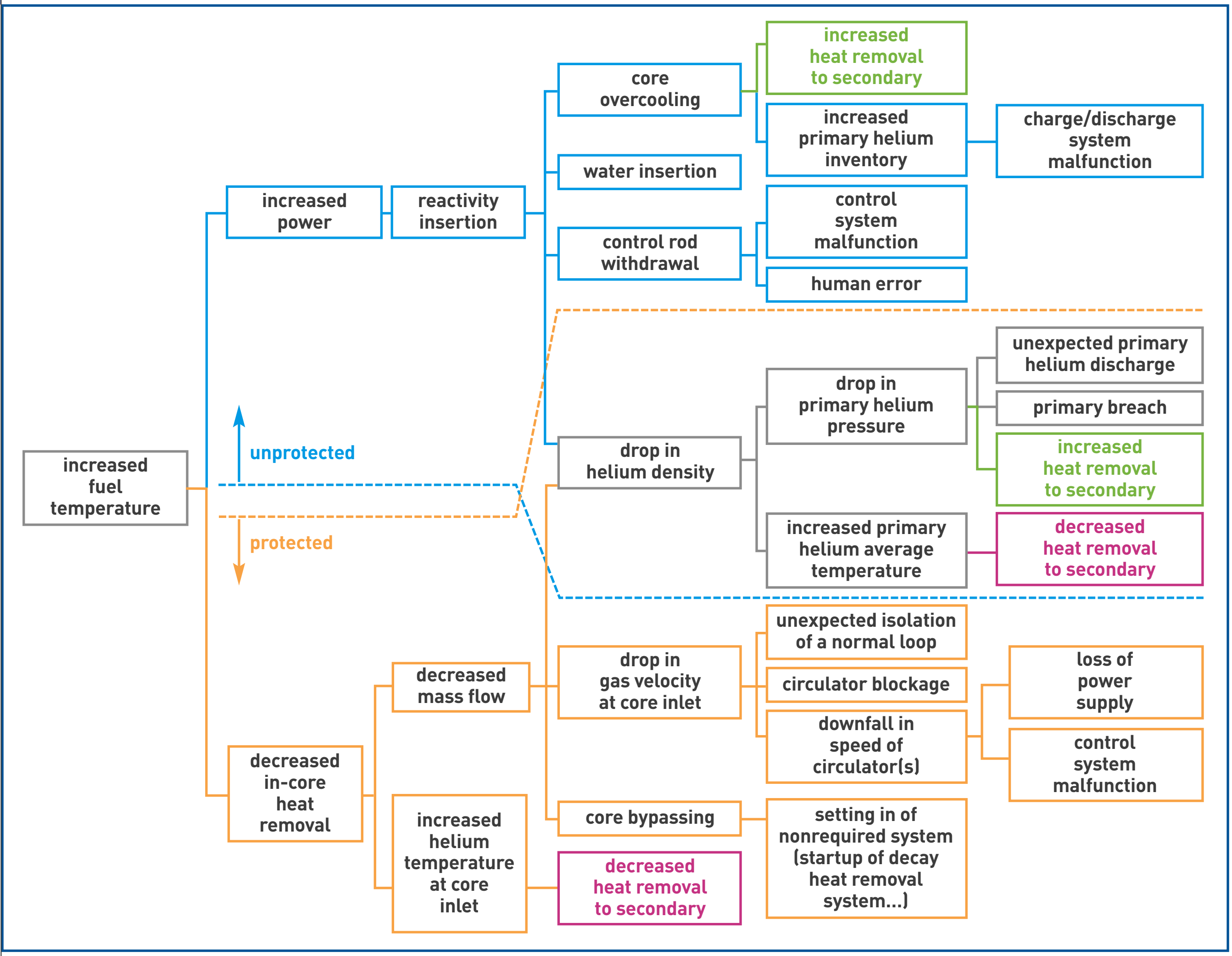
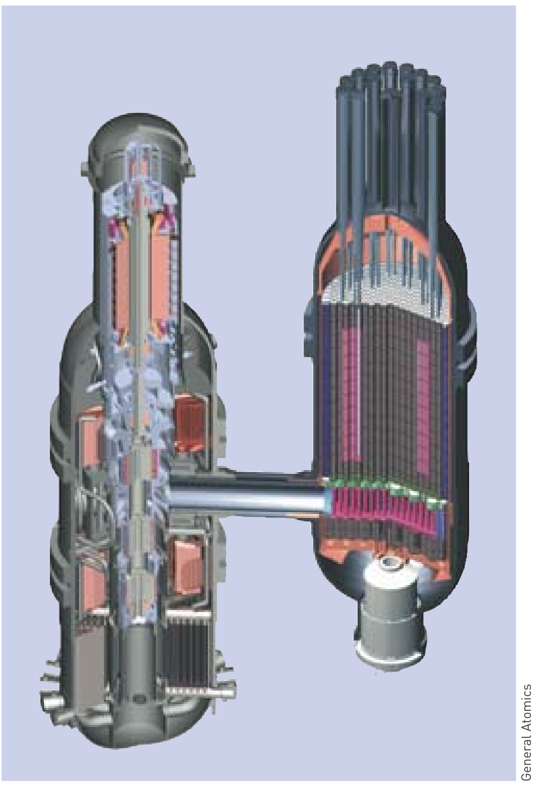


Figure 3.
Simplified event tree, with regard to events resulting in an abnormal fuel temperature rise, for an indirect combined energy conversion cycle GFR.



To meet the twin challenges of a fast spectrum, and high temperatures, the GFR brings advantages inherited from the operational feedback from past gas-cooled reactors, and modern high-temperature reactor concepts. The GT–MHR project, from General Atomics, in the United States, is a modular, direct-cycle HTR. Its power of 600 MWth, core geometry, and vessel size are optimized to ensure maximum fuel temperature, during a decay heat removal transient, with no gas injection, does not rise above 1,600 °C.

Finally, to control possible chemical reactions, inerting[(7)] the guard containment enclosure may be considered.

The depressurization accident

With modular HTRs – of which the GT–MHR (Gas-Turbine Modular Helium Reactor, a direct-cycle reactor designed by **General Atomics**, in the United States) or the PBMR (Pebble-Bed Modular Reactor, a direct-cycle reactor developed by **Eskom**, in South Africa) are instances – design solutions to limit the consequences of the depressurization accident consist, on the one hand, in going for low unit power, which restricts core volume, and low **volumetric power**, thus diluting the fuel; and, on the other hand, in counterbalancing the gas's low heat capacity through the thermal inertia of a large volume of **graphite**, which at the same time acts as **moderator**. With GFRs, the **fast spectrum** entails doing away with graphite inside the core, and going for significantly increased volumetric power. Novel solutions are required, if the consequences of a rapid depressurization event are to be limited.

(7) Inerting: a preventative technique, involving the substitution, for an explosive atmosphere, of a noninflammable gas (most commonly nitrogen), or gas mixture, further having no ability to act as oxidizer (inert atmosphere).

The designer works on the basis of an initiator seen as more severe than most loss-of-coolant-related accidents, and incidents. This is the rapid depressurization due to a large breach in the circuit. The large breach causes rapid depressurization of the primary circuit, typically over an interval lasting 5–40 seconds. Control rod dropdown occurs during the very first few seconds. As the core has a positive void coefficient, the depressurization accident induces a rise in reactivity, however this is controllable, being lower than the margin yielded by delayed neutrons. Once the rods have dropped down, the core is **subcritical**, however it does still release a significant amount of power: **residual power** (decay heat) (see Table 3).

time t	1 min	2 min	4 min	10 min	20 min	1 h	4 h
P/P_N	5%	4%	3%	2.5%	2%	1.5%	1%

Table 3.
Residual power decay, subsequent to reactor shutdown, approximately conforms to the curve $P/P_N = 0.15 \cdot t^{-0.28}$, where P_N is the reactor's nominal thermal power.

A. Gonin/CEA

The HETIQ (HElium TIghtness Qualification) test rig, set up at the CEA Cadarache Center, developed to carry out testing, and qualification of all gastight sealing devices likely to be encountered in a GFR, under representative conditions.

The principle selected, in the first instance, for the removal of decay heat from the GFR core is gas circulation, the breach notwithstanding. Transition between nominal regime, and the new regime, of convection at lower pressure, takes place within a short time interval. One essential parameter, for system performance, is gas pressure. To act on the way this varies, engineers are working on a guard containment concept – functionally, a gastight envelope surrounding the primary circuit. In the situation considered here, the envelope contains the gas it already holds, together with the gas escaping from the primary circuit. Ultimately, the system reaches equilibrium, with the containment and primary circuit at the same pressure: the *backup pressure*.

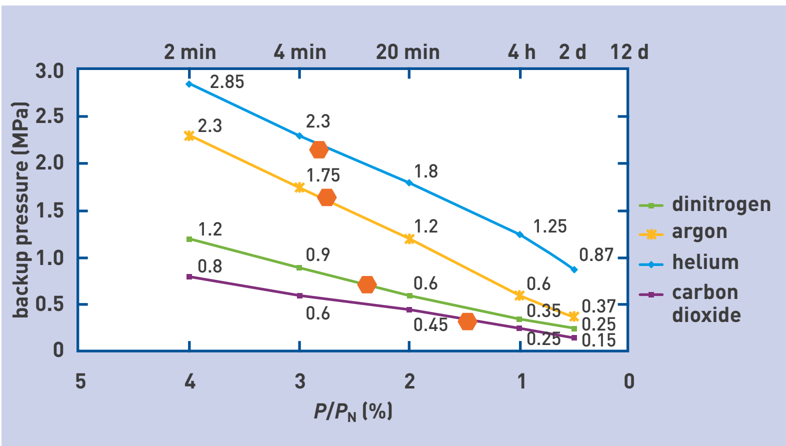
For design purposes, engineers sought to ascertain whether gas natural convection could prove sufficient to ensure removal of the power delivered by the core. The criteria to be met are many. Conservatively, it should be taken that fuel maximum temperature, and maximum temperature of structures outside the core, should not exceed, respectively, 1,600 °C, and 1,250 °C, over a limited time interval. Consolidating such criteria will be a crucial point for the investigations to be carried out, be it with regard to the fuel, or reactor components exposed to these high temperatures, such as thermal protections, and heat exchanger walls.

What are the attractions, and limitations of natural convection? If natural convection is to prove adequate to ensure decay heat removal, one crucial condition is that backup pressure should remain within the 0.7–2.5 MPa range, depending on the criterion selected, and the gas employed. The values obtained, according to the gas, and the heat level to be removed, are shown in Figure 4, for a core volumetric power of 100 MW/m³. Dinitrogen and carbon dioxide are included, to cater for the emergency injection case, and improved system cooling performance. With respect to helium, the backup pressure required in the first few minutes is particularly high, standing at 2–3 MPa. Dinitrogen and carbon dioxide bring down the pressure requirement to less than 1.5 MPa. In any event, setting up such a pressure calls for a gastight guard containment, ensuring that leaking gas be contained. It should be noted that core cooling by means of natural convection, at normal operating pressure, is assured: this is the blackout case.

What are the best design options? Ensuring a backup pressure as high as 2–3 MPa is a peculiarly demanding target. The guard containment structure is a complex design and construction task (prestressed concrete vault, of large size, and great thickness). It must be kept constantly pressurized, when the reactor is operating.

The option selected for the GFR, at this stage, is to go for a lighter, initially unpressurized containment. Target backup pressure is thus brought down to about 1 MPa. Consequently, it is only feasible to rely on natural convection once a sufficient interval has elapsed, to allow residual power to decay to less than 1% of nominal power. In the short term, core cooling is to be ensured by low-power forced circulation systems (circulators, or other devices). To provide for this, it is specified that the power supply required for **forced convection** functioning has to be sufficiently low to be delivered from batteries. Figure 5 illustrates this approach. Terminating forced convection

Figure 4.
Backup pressure requirement to ensure decay heat removal through gas natural convection, according to the nature of the gas injected. The quantity P/P_N, plotted along the x-axis, is the ratio of residual power (to be removed), over reactor nominal power.
The orange hexagon, on each curve, corresponds to the pressure reached inside the guard containment, after injection of 50 m³ of gas at 60 °C.



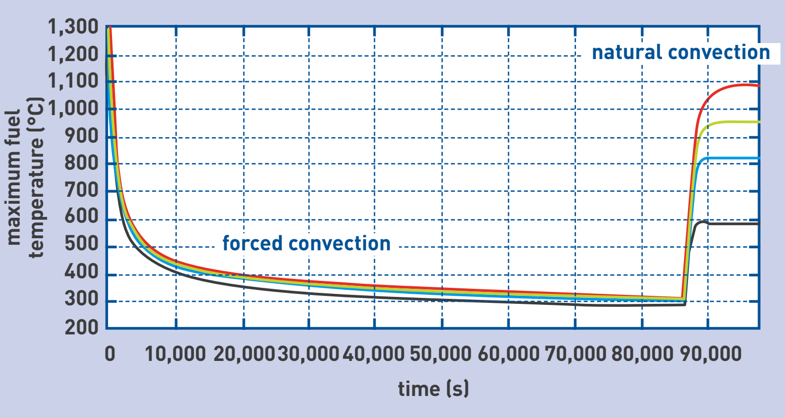


Figure 5.
Evolution of core temperature during a depressurization accident, using low-power pumping for 24 hours, followed by natural convection (CATHARE results). The curves correspond to different core channels (maximum, average, or minimum power).

after 24 hours results in rising fuel temperatures, these then stabilizing due to natural convection cooling setting in.

In practice, the system comprises three independent loops, each of which has the capability to carry out, operating singly, the decay heat removal mission (functional redundancy). The layout is of the cross-duct type, i.e. featuring a hot duct immersed in the cold leg. Each loop comprises a heat exchanger predimensioned to remove up to 3% of nominal power, positioned in the upper region (at a height of some 15 m above the core's median plane), together with a circulator and an isolation valve, mounted in the cold leg.

Reactivity accidents

Abnormal variations in core reactivity also stand as a crucial point, as regards reactor core design. In particular, ingress of water into a fast reactor core would be a cause for alarm, since water, as it moderates neutron energy, induces increased **fission cross-sections** in plutonium, thus leading to rising core reactivity.

(8) Absorption resonance: a high cross-sectional value, exhibited for a definite value of the energy yielded to a target **nucleus** by an incident neutron, indicating an excited state of the *compound nucleus*. In certain incident energy domains, neutron–nucleus interaction cross-sections are seen to be highly dependent on the neutron's energy, owing to the existence of such resonances.

P. Dumas/CEA

Nitride and carbide fuel pellets and spheres – the materials best meeting the specifications for the GFR core – fabricated at LEFCA (Laboratoires d'études et de fabrications expérimentales de combustibles nucléaires avancés: Advanced Nuclear Fuels Design and Experimental Fabrication Laboratories, CEA/Cadarache). Irradiation experiments in the Phénix reactor will allow investigation of their behavior and performance, under conditions representative of those in GFRs.

Calculations have been carried out for an older, 600-MWth version of the GFR type. It has been shown that limited-volume water ingress results in reduced core reactivity. Indeed, **capture cross-sections** in absorption resonances(8) rise faster than fission cross-sections, for low water contents. It is only for large amounts, typically for a volume percentage higher than 70%, that the core becomes over-reactive, and that water may bring about a **power excursion**. Such findings, however, must be taken altogether cautiously, since computation of the relevant situation is one of the most difficult tasks around, for **simulation** tools. The large variation in the neutron spectrum, between the initial, dry core condition, and the flooded condition, together with the lack, for the time being, of definite data as to the precise geometry for the core, and layout of the various materials, result in considerable uncertainties, which will be resolved as design progresses.

The reactor design must thus take this process on board, by precluding risks of massive water ingress, through an adequate layout for water-holding circuits, and, conceivably, by introducing a **neutron poison** into these circuits, so as to render the water harmless. Finally, an air ingress is of no direct neutronic consequence, however it may involve a possible chemical effect on core materials.

Towards a viability demonstration by 2012

At this stage in the project, the very design basis of the GFR is still evolving, however major design directions have been decided on, as regards the fuel, core materials, reactor architecture, and safeguard systems. Assessment of GFR viability, and performance evaluation will become more detailed as design progresses. An interim report is due to be presented at the end of 2007.

Fuel behavior and performance in normal operating conditions, along with fuel behavior, and that of installation components, in accident situations certainly stand as the salient viability issues for the GFR. By 2012, a demonstration of viability for the fuel, and, more broadly, for the reactor, for this technology line, will be a necessary prior condition, to embark on construction of a low-power reactor, REDT (see *REDT, a precursor for GFRs*, p. 114). Fuel viability will have been corroborated by representative trials in reactor. Qualification of the computation tools will provide the foundation for the demonstration of safety the designer will seek to adduce. The main such tools are the ERANOS computation code, for neutronics, and CATHARE, for core and system thermal–hydraulics. While they have already seen widespread employment, these codes are still not wholly qualified, for GFR applications. Matching calculations with experience will allow a narrowing down of uncertainties, and ultimately their resolution.

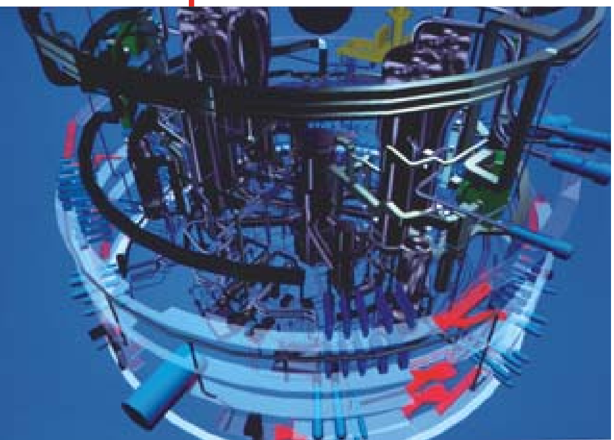
> Jean-Claude Garnier*, **Patrick Dumaz***
and **Pascal Anzieu****
Nuclear Energy Division
**CEA Cadarache Center*
***CEA Saclay Center*

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


M. Brigaud/ EDF Médiathèque

The four PWR units of EDF's Avoine power station, near Chinon (central France), belong to the second generation of nuclear reactors.

Nuclear reactor lines correspond to the many combinations of three basic components: **coolant**, **moderator** (when required), and **fuel** – almost invariably **uranium**, possibly mixed with **plutonium** (see Focus A, ***The components of a nuclear system***, p. 10).

Numerous setups have been experimented with since the onset of the industrial nuclear energy age, in the 1950s, though only a few of these were selected, for the various generations of operational power generating reactors.

The term **technology line**, or **reactor line**, is thus used to refer to one possible path for the actual construction of nuclear reactors having the ability to function under satisfactory safety and profitability conditions, and defined, essentially, by the nature of the **fuel**, the energy carried by the **neutrons** involved in the **chain reaction**, the nature of the moderator, and that of the **coolant**.

The term is used advisedly, implying as it does that this combination stands as the origin of a succession of reactors, exhibiting characteristics of a technological continuum. More or less directly related to this or that line are research and trials reactors, which are seldom built as a series. Such reactor lines are classified into two main families, depending on the **neutron spectrum** chosen: **thermal**, or **fast** (an operating range partly straddling both domains is feasible, for research reactors), according to whether neutrons directly released by **fission** are allowed to retain their velocity of some 20,000 km/s, or whether they are slowed down to bring them into thermal equilibrium (**thermalizing** them) with the material through which they scatter. The neutron spectrum, i.e. the energy distribution for the neutron population present within the **core**, is thus a **thermal spectrum** in virtually all reactors in service around the world, in particular, in France, for the 58 **PWRs** (**pressurized-water reactors**) in the **EDF** fleet. In these reactors, operating with **enriched uranium** (and, in some cases, **plutonium**), heat is transferred from the core to **heat exchangers** by means of water, kept at high pressure in the **primary circuit**.

Together with **BWRs** (**boiling-water reactors**), in which water is brought to the boil directly within the core, PWRs form the major family of **light-water reactors** (**LWRs**), in which ordinary water plays the role both of coolant, and moderator.

Use of the **fast spectrum** is, currently, restricted to a small number of reactors, operated essentially for experimental purposes, such as Phénix, in France, Monju and Joyo, in Japan, or BOR-60, in Russia. In such **fast reactors** (**FRs**), operating as they do without a moderator, the greater part of **fission** processes are caused by neutrons exhibiting energies of the same order as that they were endowed with, when

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 → 2); burning the air–fuel mix inside the combustion chamber (2 → 3); and allowing the hot gases to expand inside a turbine (3 → 4). On exiting the turbine, the exhaust gases are discharged into the atmosphere (this forming the cold source): the cycle is thus termed an *open* cycle. If the hot source is a nuclear reactor, open-cycle operation, using air, becomes highly problematical (if only because of the requisite compliance with the principle of three confinement barriers between nuclear fuel and the ambient environment). In order to *close* the cycle, all that is required is to insert a heat exchanger at the turbine outlet, to cool the gas (by way of a heat exchanger connected to the cold source), before it is reinjected into the compressor. The nature of the gas then ceases to be dictated by a combustion process.

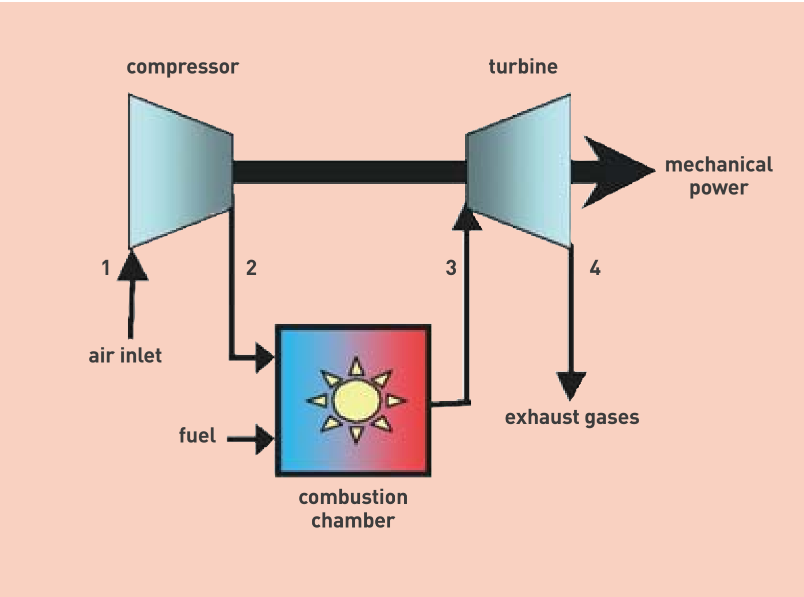


Figure.
Brayton cycle, as implemented in an open-cycle gas turbine.

FOCUS D

What is multiphysics, multiscale modeling?

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

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

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

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

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

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

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

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

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

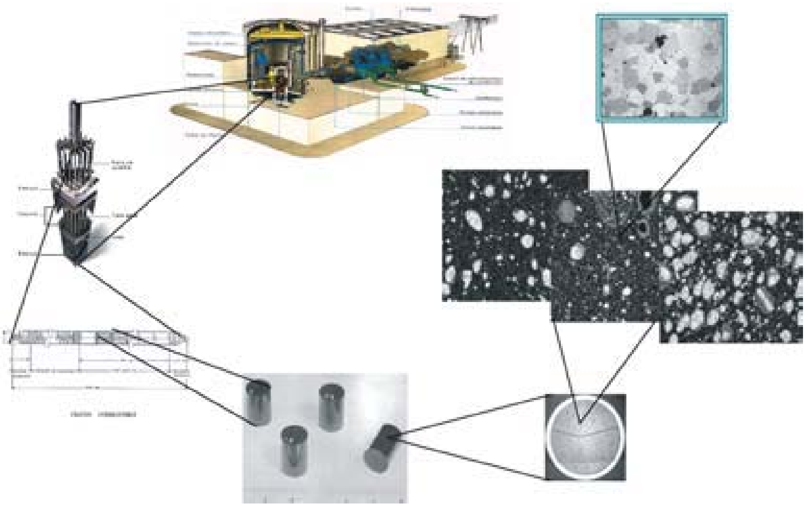


Figure.
Improving nuclear fuel reliability, and cost-effectiveness calls for finescale modeling of that fuel, through a multiscale approach, from reactor to fuel microstructure (in this instance, MOX fuel). Microstructural characteristics (porosity, cluster size and distribution, grain size...) have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

The main families of nuclear materials

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

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

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

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

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

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

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

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

Crystal structures

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

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

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

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

Steel families

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

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

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

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

Areva NP

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

FOCUS (Cond't) E

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

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

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

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

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

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

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

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

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

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

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

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

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

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

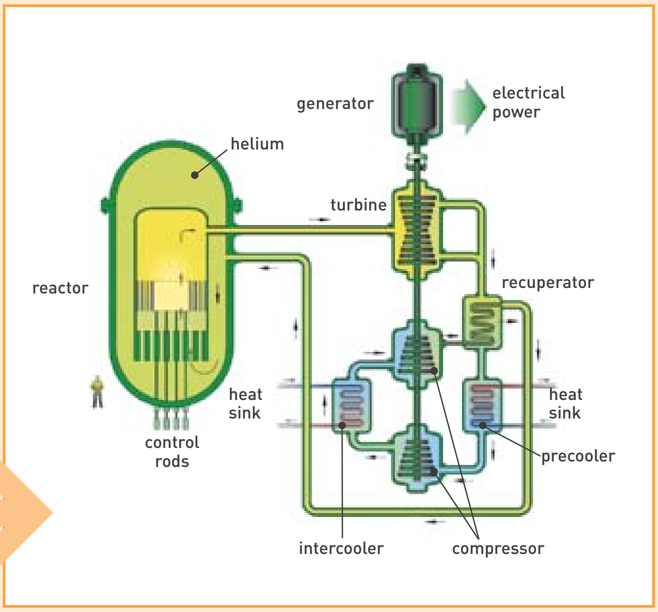
The six concepts selected by the Gen IV Forum

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

GFR

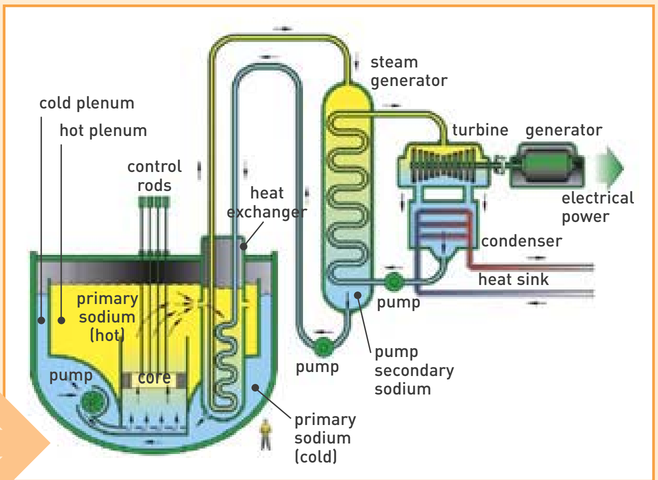
The gas-cooled fast reactor system (GFR) is a high-temperature, gas-cooled (helium-cooled as a rule), fast-neutron reactor allowing actinide recycle (homogeneous, or heterogeneous), while sustaining a breeding capability greater than unity. The reference concept is a helium-cooled, direct- or indirect-cycle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

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



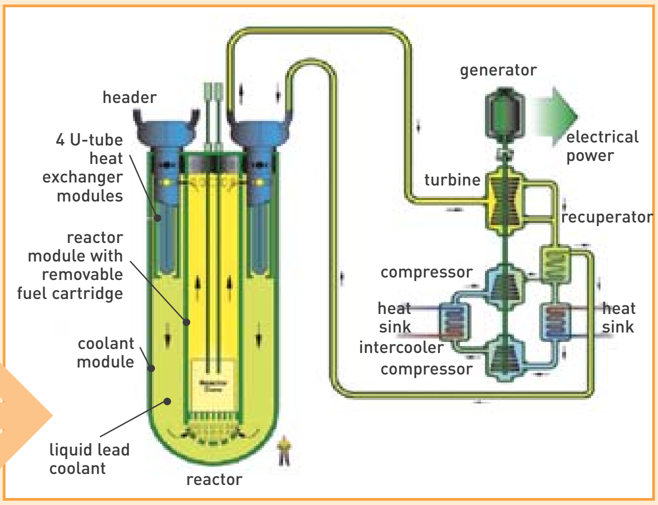
Le SFR

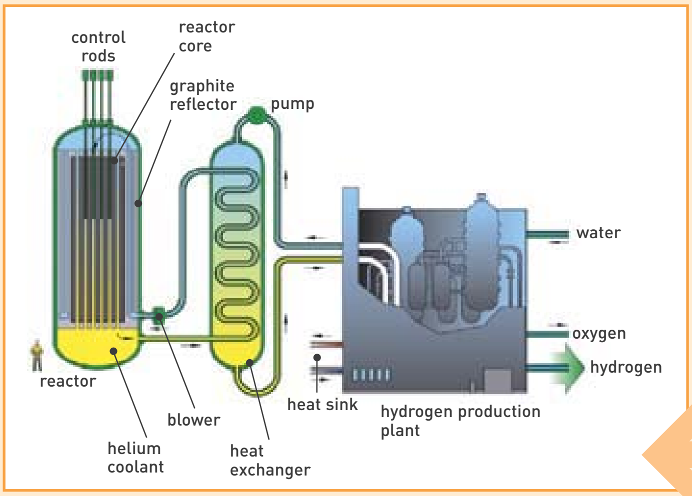
The sodium-cooled fast reactor system (SFR) is a liquid-sodium-cooled, fast-neutron reactor, associated to a closed cycle, allowing full actinide recycle, and plutonium breeding. Owing to its breeding of fissile material, this type of reactor may operate for highly extended periods without requiring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



LFR

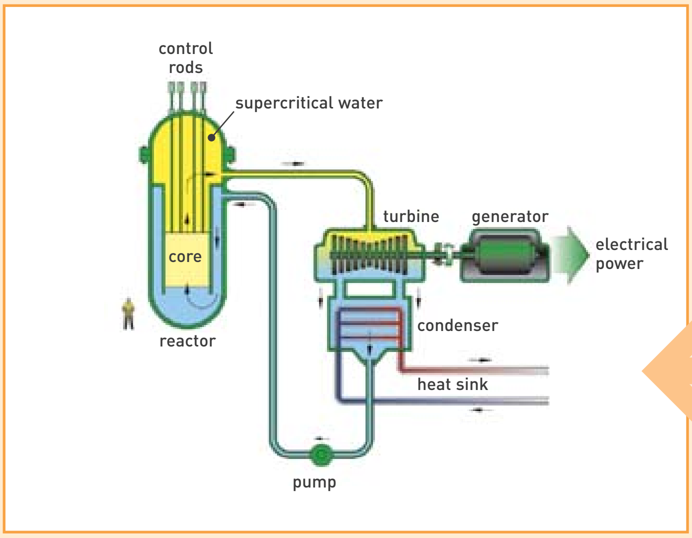
The lead-cooled fast reactor system (LFR) is a lead- (or lead-bismuth alloy-) cooled, fast-neutron reactor, associated to a closed fuel cycle, allowing optimum uranium utilization. A number of reference systems have been selected. Unit power ranges from the 50–100 MWe bracket, for so-called battery concepts,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





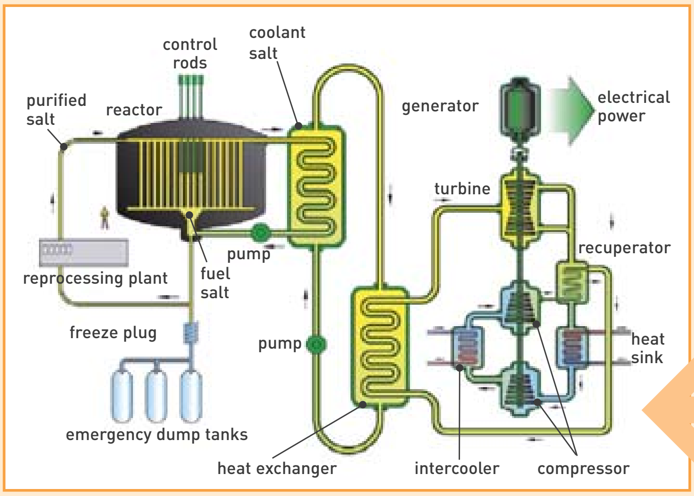
VHTR

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



SCWR

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



MSR

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