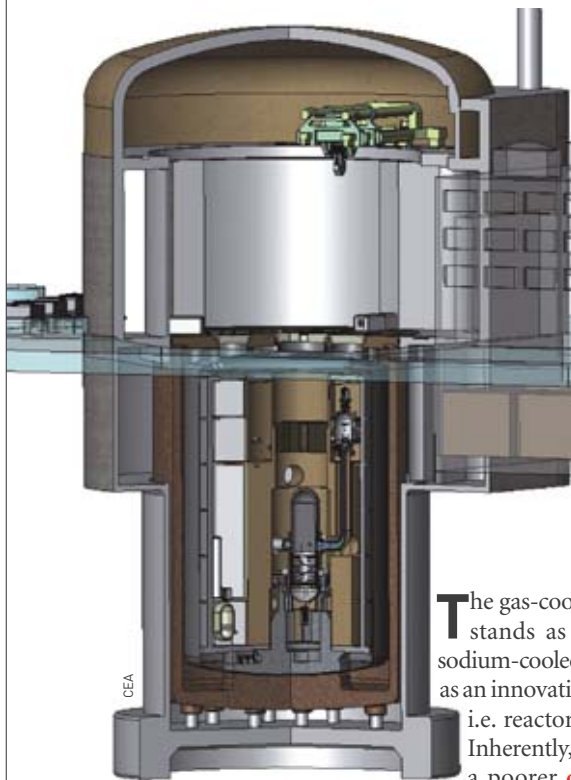




ETDR, a precursor for GFRS



General view of ETDR, showing, in the lower region, the primary circuit with its metallic close containment, and, in the upper region, the reactor building with its concrete confinement, and the traveling crane for the handling of bulky components.

The gas-cooled **fast reactor (GFR)** stands as an alternative to the sodium-cooled fast reactor (**SFR**), and as an innovative path for fast reactors, i.e. reactors using **fast neutrons**. Inherently, even though definitely a poorer **coolant** than sodium, **helium** exhibits attractive characteristics, with respect to reactor design, and safety, since it is a chemically inert gas, and transparent to **neutrons**. Further, the GFR, with its ability to operate at a higher temperature, would allow other requirements than electricity generation to be met, such as the supply of process heat, for **hydrogen** production, or production of synthetic fuel,⁽¹⁾ or even for other industrial applications, while remaining unaffected by tight **uranium** prices (see *Towards nuclear energy applications other than electricity production*, p. 123). On the other hand, this concept cannot draw on operational feedback, at the scale of a reactor with some operating mileage.

A milestone on the way to the industrial prototype

The GFR development schedule, appreciably more protracted as a whole than for its sodium-cooled counterpart, covers three main stages, of which construction of the Experimental Technology Demonstration Reactor (ETDR) is the second, the first stage involving overcoming the main technological barriers, namely development of an innovative, refractory fuel, and choice of safety options. A low-power reactor, ETDR will provide the capability to establish the viability basis for **core** design, **fuel** technology, **neutronic** control, and instrumentation, prior to going on to the third stage: construction of a medium-power, electricity-generating prototype, comparable to that contemplated for the SFR reactor line.

(1) Synthetic fuel: a hydrocarbon fuel obtained through chemical synthesis, rather than by refining natural hydrocarbons.

The gas-cooled fast reactor stands as a highly innovative path, benefiting from no such operational feedback, however, as sodium-cooled reactors. Before construction can be considered – probably in a European context – of a medium-power, electricity generating prototype, CEA is thus laying the ground for an Experimental Technology Demonstration Reactor (ETDR), which could start up in 2020, for the purposes of gaining in-depth knowledge, in actual operating conditions, of gas-cooled fast-neutron reactor technology.

The decision whether to go ahead with construction of ETDR is to be taken in 2012, at the outcome of feasibility and definition studies, with commissioning scheduled for 2020.

The goals set for ETDR

The overall goal set for ETDR is that of establishing the viability of the GFR, and its fuel cycle, with a view to its long-term deployment, and, more specifically, to provide the demonstrations required for qualification of the solutions selected for this reactor line, and for the industrial prototype. ETDR must, first of all, serve to draw up initial safety options and criteria, validated by the safety authorities. ETDR safety options will not necessarily prove amenable, in some respects, to being carried over to the prototype reactor, for which, e.g., the role played by active systems may prove smaller. Nevertheless, these options will stand, obviously, as a definite milestone, in terms of the definition and validation of safety options for a gas-cooled fast reactor, and the stance taken by the safety authorities will, in this respect, stand as a reference for subsequent reactors.

The second goal will be the qualification of GFR fuel and **fuel assembly** technologies. In the GFR fuel development plan, ETDR comes in at a point between sample irradiations (a few grams) in technological

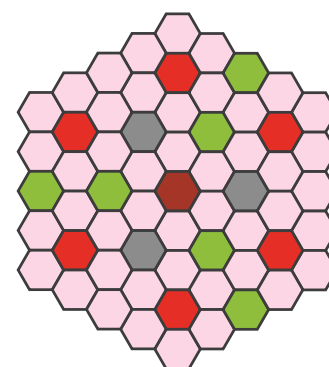


Figure 1. Loading plan for the ETDR core. Shown in red: control rods; in green: test assemblies; in gray: complementary shutdown rods; and in brown: an instrumented assembly.

irradiation reactors, and full-scale demonstration (several tonnes of fuel) in the industrial prototype. It provides the flexibility required for the investigation of the various core and assembly concepts that may be considered, for the reactor line and its prototype. The third goal: the demonstration of operation, of control, and of the associated instrumentation.

The experience gained with ETDR will make it possible to proceed with further development of the GFR reactor line, going on to the industrial prototype stage.

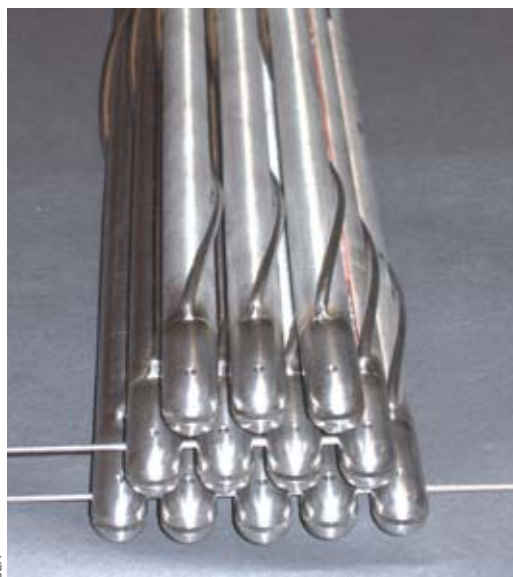
Core reference options

In design terms, as regards the core in particular, ETDR is directly related to the prototype reactor, for which it stands as a scaled-down mockup. It contrasts radically with an MTR- (materials testing reactor) type reactor, such as the Jules Horowitz Reactor (JHR) for which the remit is one of ensuring that irradiation requirements are broadly covered, for the various reactor lines (see *The JHR reactor: a multipurpose asset for materials*, p. 102).

Preliminary dimensioning of the ETDR core will be grounded on representativeness criteria, with respect to irradiation conditions, as compared to those for the reference provided by the commercial 2,400-MWth GFR. The aim is to achieve, at the same time, identical cladding and fuel temperatures, a fast flux and **dose/burnup** ratio close to those to be exhibited in the GFR, along with a core characterized by power of 50 MW, and average **power density** of 100–150 MW/m³, which is seen as affording the best performance/cost tradeoff at this stage.

The fuel used in this so-called “demonstration” configuration will be the innovative fuel selected as reference for the GFR. This comes in the form of flat plates, holding a (U,Pu)C carbide fuel. The **cladding** is made of **SiC-SiCf composite**.

ETDR is primarily designed to operate with the GFR advanced fuel, however it is due to be built before the



Part of an assembly of 19 test pins intended for the ETDR startup core.

GFR fuel can be fully qualified. Consequently, a so-called “startup” core is being contemplated, of proven design, in which irradiation, of an experimental character, of the GFR advanced fuel will be carried out in test assemblies (see Figure 2). For reference purposes, the startup core will comprise (U,Pu)O₂ oxide fuel pins, featuring metallic cladding (SFR technology). These entail a definitely lower core outlet helium temperature, compared with the GFR (560 °C, as against 850 °C). The strong point, with regard to safety, will be the use of steel for cladding, more specifically of a grade known for its good behavior under a **fast** flux. The Table sets out the main core characteristics, for the 2,400-MWth GFR, and ETDR (startup, and demonstration), as obtained at the outcome of exploratory design studies. The configurations selected exhibit irradiation performance

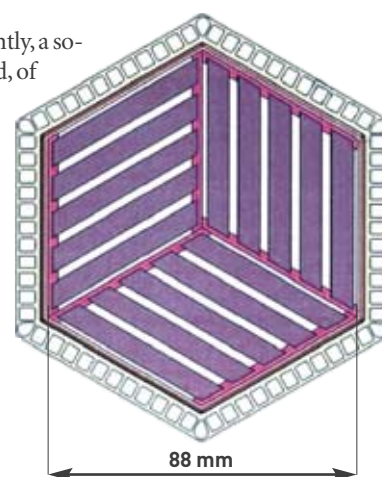


Figure 2. Test assembly holding the advanced, honeycomb-plate GFR fuel.

	2,400-MWth GFR	ETDR demonstration core	ETDR startup core
unit power [MWth]	2,400	50	50
core power density [MW/m ³]	100	100	100
helium pressure [bar]	70	70	70
core inlet/outlet temperature [°C]	400/850	400/850	260/560
fuel	(U,Pu)C-SiC	(U,Pu)C-SiC	(U,Pu)O ₂ pellet
fuel assembly	plate	plate	pin
fuel/coolant fraction [% vol]	22.4/40.0	23.5/31.0	36.2/42.0
(matrix + gap)/structure fraction [% vol]	17.6/20.0	18.5/27.0	3.1/18.7
plate thickness [mm]	7.0	6.27	–
pin diameter[mm]	–	–	6.55
max. cladding/fuel temperature [°C]	965/1,260	1,045/1,140	615/1,045
core pressure drop [bar]	0.44	0.58	0.62
Pu/(U + Pu) [%]	15.2	34.9	27.3
average/max. burnup [at %]	10.1/15.7	5.4/8.6	5.2/8.8
max. fast flux [E>0.1 MeV (n/cm ² /s)]	1.4 10 ¹⁵	0.94 10 ¹⁵	1.1 10 ¹⁵
max. fast fluence [E>0.1 MeV (n/m ²)]	3.0 10 ²⁷	1.1 10 ²⁷	1.6 10 ²⁷
max. dose [dpa SiC]	163	61	85
dmax. dose/max. burnup [dpa SiC/at %]	10.3	7.1	9.7
He end-of-life void effect [pcm]	253	-18	63
end-of-life Doppler effect [pcm]	-1,175	- 523	- 360
delayed neutron fraction [pcm]	344	339	353

Table. Comparison of the main characteristics for a 2,400-MWth GFR core, and for the two cores planned for ETDR.

meeting the target set (some 30% lower than the corresponding values for the 2,400-MWth GFR), as regards fast flux, and dose/burnup ratio.

Fissile element (namely **plutonium**) content is notably higher (by about 30% in terms of the plutonium/heavy metal mass ratio), compared with the 2,400-MWth GFR. This is to offset the small size of the ETDR core, which induces a significant increase in **neutron leakage**. With regard to the fuel, and fuel cycle, the values involved are in no way an obstacle: the limit for oxide fuel **reprocessing** thus stands at 40–45% Pu/(U + Pu).

The other parameters of importance are the effects on **reactivity**: the helium depressurization effect is close to zero, which is a favorable consideration, obviously, while the **Doppler effect** is comparable to that for an SFR of equivalent size.

The contribution to fuel cycle demonstrations

GFR fuel fabrication technologies will be validated, initially (test assemblies, in the startup core), for uranium–plutonium assemblies with no **minor actinides**, fabrication of which may be carried out in glove boxes.

An experiment relating to integral recycling (uranium, plutonium and minor actinides), involving

about 1 kg of actinides (a fraction of an ETDR assembly), will form a key point in the development program for this technology line. Reprocessing, followed by refabrication, could allow the carrying out, in ETDR, of a cycle closure experiment. Beyond providing a demonstration of processes, the aim will thus be to investigate in-pile behavior of the fuel assembly after reintroduction of all minor actinides, but also with a residual content of impurities and **fission products** yielded by **recycling**.

Fuel fabrication facilities will have to be developed through rational management of the switchover from R&D facilities to industrial plants. Ultimately, the cycle front end (the supply of fuel) and back end (fuel reprocessing and refabrication, to recycle the actinides) will be part of the approach for qualification of

GFR fuel cycle processes, for core-scale equipment.

Further contributions from ETDR operation

Aside from fuel irradiations, ETDR will allow the experimental determination of a number of other points, starting with coupling effects, between core neutronics and **thermal-hydraulics**. Processes such as neutron **feedback**, subsequent to increased core inlet temperature, core heatup, or overpower, involve all aspects of reactor behavior: core mechanical equilibrium, support for **control rod** mechanisms, structure–gas thermal exchanges, fuel thermal behavior, Doppler effect, fuel element expansion... These feedback processes govern the core's natural behavior, and, for that reason, stand as the major parameters for the

safety analysis. ETDR will further allow investigation of **decay heat** in the core, and the conditions for its removal, to complement tests carried out in inactive **loops**: management of the initial phase of loss-of-coolant **transients**, indeed, will be a key point for the safety of GFR cores, exhibiting as they do low thermal inertia.

These processes will be the object of prior evaluations, by means of **models**, qualified through *separate-effect* tests, whether in **critical mockups**, or through tests in inactive loops. The complementary goal, for ETDR, is the overall qualification of coupled studies and simulations, and, through this, the firming up of the qualification of design computer codes and the narrowing down of uncertainties.

General design of ETDR

The goal for design studies for the reactor, at the exploratory study stage, was to identify key issues and possible solutions, and acquire consistent data, allowing the reactor to be described, and modeled, for the purposes of safety studies. The design principles selected are shown in Figure 3.

The reactor vessel and associated functions

The **vessel**, comprising a cylindrical shell, and hemispherical bottom, is 3 m in diameter, with a height of 8 m, for a thickness of 10 cm (the components are 9Cr1Mo **steel** forgings). Total mass (vessel head included) stands at around 120 tonnes.

Helium enters the vessel through a side pipe connector, descending to the bottom via an annular space, ascends through the core to reach the hot **plenum** above the core, and exits via the side connector. The primary loop layout, outside the vessel, is of the cross-duct type, featuring two concentric ducts: the piping holding the “hot” gas being inserted into the duct channeling the “cold” gas.

The mobile, absorbent part of the control rods is positioned above the core; the mechanisms are located at the bottom of the vessel, in the cool region. Fuel handling is effected without removing the vessel head, an articulated arm system allowing vertical and lateral movement of assemblies inside the vessel; loading and unloading is effected via lock chambers, at the top of the vessel.

Cooling in nominal operating conditions

Owing to its low power, and in order to simplify the system, and minimize capital cost, ETDR features no energy conversion system, and is not intended to produce electricity. It will be cooled by pressurized helium, by means of an isobaric⁽²⁾ circuit fitted with blowers. The **secondary** loop is a pressurized-water circuit, the ultimate heat sink being the atmosphere. The technology selected for the **heat exchanger** in the main circuit is the *tube* and *calandria* technology, already used for the Japanese HTTR reactor. Power in operating conditions is removed via a single loop (see Figure 3). A power extraction of 10 MW, for a circuit branching off from the main one (not shown), is earmarked for possible experimental activities, at a later date, e.g. testing components such as a high-temperature intermediate heat exchanger (**IHX**)...



The ESTHAIR (Essais thermoaérodynamiques en air: Gas Thermal-Hydraulic Tests in Air) facility, at CEA's Grenoble Center, intended, in particular, for the evaluation of friction and heat exchange coefficients for the pin-type fuel element for the ETDR startup core.

D. Michon-Artequique / CEA

Safeguard systems

As is the case for the GFR, decay heat removal relies on gas circulation, and on the presence of a leaktight close containment, enclosing the primary circuit. Such circulation is ensured by redundant, diversified means, primarily active pumping equipment (circulators). Pumping power is set by reference to the cooling requirement in accident conditions (depressurization), the **backup pressure** available within the guard containment (about $3 \cdot 10^5$ Pa) making it possible to keep to an altogether reasonable power level (< 50 W per machine). In the event of loss of primary flow, with circulators failing to start up in the decay heat removal loops, the pressure level inside the reactor is sufficient to allow core cooling through **natural convection**.

The safeguard system comprises three independent loops, each one of which has the capability to carry out, operating singly, its mission. Each loop features a heat exchanger, predimensioned to remove up to 3% of nominal power, located 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.

ETDR development calls for completion of the feasibility study by 2009, and the compiling of the mandatory safety options report. The study approach, similar to that adopted at the same time for the GFR, takes in the safety approach, and the study of safeguard systems. For the safety approach, the aim is go for an exhaustive analysis, looking to combine the various tools (probabilistic, deterministic⁽³⁾), and set out the rationale to justify why certain situations are taken on board, and why others are precluded by the design. The design and performance study, for the safeguard systems, relies on **numerical simulations**, making it possible to establish that operating limits, for barriers and components, will not be overrun.

The safety demonstration will finally prove conclusive when it is possible to set out the grounds both for the decoupling criteria to be met, and for the relevance of the models used, at the various levels involved. As regards the first issue, the strong point is precisely that, when reasoning on the startup core, fuel element operating limits (cladding resistance/failure) can be precisely established, and, consequently, so can the performance demanded from the safeguard systems. The second issue primarily concerns the core but also the circulators, heat exchangers, the interaction with structures and the close containment...

Once feasibility is established, preconceptional design studies will be initiated, to lay the ground for the decision on construction (in 2012), by setting out cost factors for the reactor, along with complementary viability indications (for the initial core, in particular).

Concurrently with design studies, technological investigations will be required, to achieve mastery of helium

technology, and demonstrate the viability of the solutions put forward, drawing on the experience gained with thermal-spectrum reactors. Specific tests in helium, whether for thermal-hydraulic purposes (assembly mockup, at a scale close to 1), or for more technological concerns (mockups of the circulator, of the decay heat removal heat exchanger...), are planned, in a facility currently undergoing construction, the HELITE loop, at Cadarache.

The indispensable international collaboration

Standing as it does as the indispensable prerequisite for completion of the project, international collaboration is intended to share the research efforts, speed up technological developments, and optimize utilization of human and financial resources. This takes place in two major contexts, the first being the **Generation IV International Forum (GIF)**, which has the remit of specifying and sharing the R&D required to lay the ground for the demonstration phases, taking the form of prototypes, to be built by the various participant countries. The GIF context will have to be put to best advantage, whenever possible, while not precluding specific arrangements (e.g. bilateral agreements). The second context, within the European Union, covers initiatives launched at the level of the **European Commission**, and **Framework Programs**, along with bilateral initiatives with EU member states wishing to enter into swifter involvement on certain points, such collaborations being seen as playing the role of driving forces with respect to the Framework Programs. The GFR reactor line may act as a project bringing a variety of factors together, for the nuclear energy of the future.

Looking at the broader picture, the French players have, as a matter of fact, suggested to their European counterparts the holding of a comparative evaluation of the three systems at hand, GFR, **LFR**, and **ADS**, with a view to selecting, in 2012, a fast reactor technology to stand as an alternative to the SFR, which could result in construction of an experimental reactor in Europe. CEA, together with its partners, are supporting, for their part, the GFR for this endeavor.

The fast-neutron variant of the gas-cooled reactor line stands as a radical breakthrough, compared to thermal-neutron reactors involving the same coolant, particularly with respect to the fuel, and safety systems. A precursor experimental reactor is indispensable, if the basic technologies are to be developed, and qualified. ETDR meets this aim, in the context of a stepwise approach, allowing, by drawing on existing technologies, the evaluation of a variety of innovative technologies, and making it possible to support their development and qualification, to the level required for their implementation in the industrial prototype.

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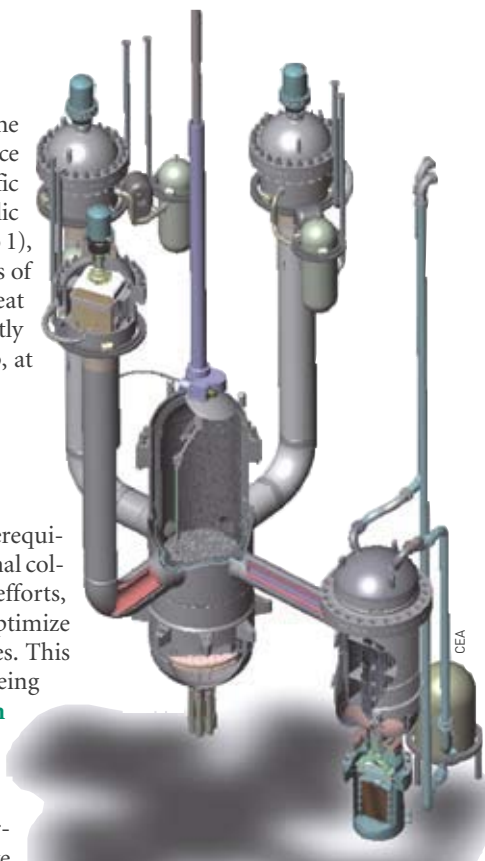


Figure 3.
3D view, showing, in particular, the ETDR vessel (at center of picture) and primary circuit. At top, the three decay heat removal system loops. Bottom right, the intermediate heat exchanger, and, below it, its circulator.

(2) Isobaric: exhibiting constant, uniform pressure, within a fluid.

(3) Deterministic: this term specifies, in particular, the type of computation, computation codes, and algorithms used for the numerical resolution of equations governing, e.g., in the realm of neutronic quantities, transport of neutrons inside matter, after "discretization" – i.e. after assigning definite, discrete values – of the variables (space, energy, time).

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,

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

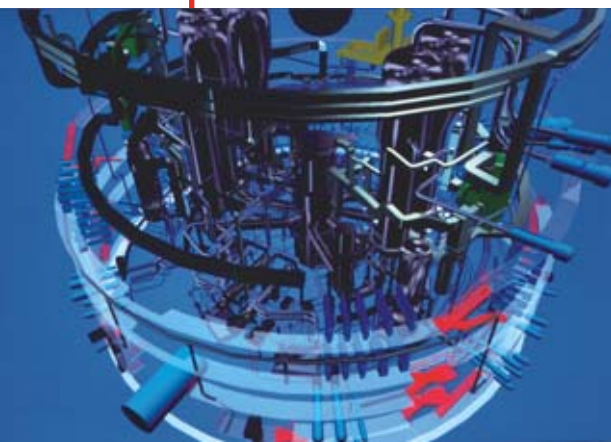
population to be regulated and, in the process, by acting on its **reactivity**, to hold reactor power at the desired level, or even to quench the chain reaction. The rods, held integral and moving as one unit (known as a **cluster**) are inserted more or less deeply into the core. Poisons, on the other 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.



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

(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).

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



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

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

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](#) (D₂O) 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.

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 T_c 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.

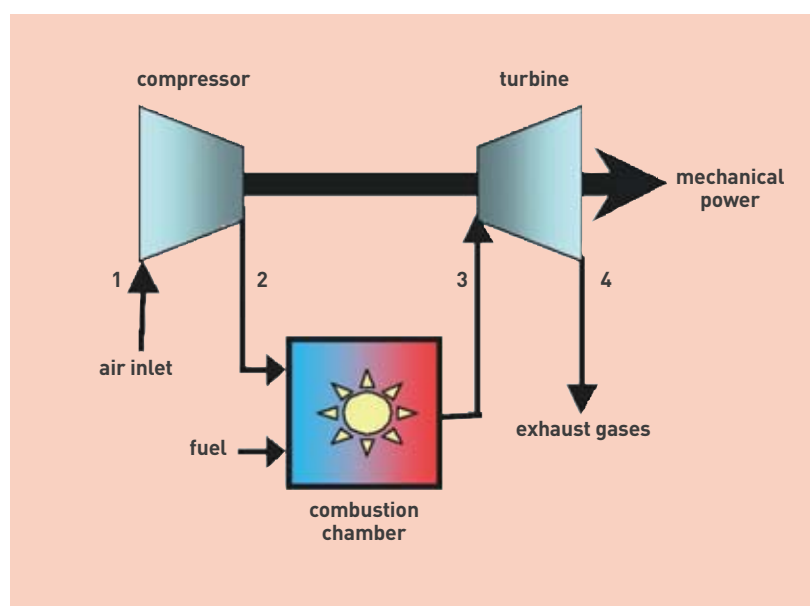


Figure.
Brayton cycle, as implemented in an open-cycle gas turbine.

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.

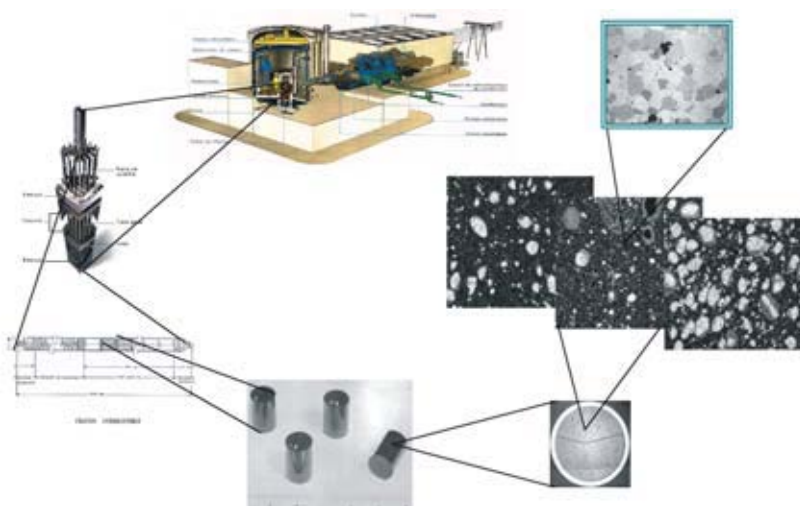


Figure.

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 super-numerary 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 $3 \cdot 10^{19} \text{ n} \cdot \text{cm}^{-2}$, for neutrons of energies higher than 1 **MeV**.

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

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



Areva NP

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

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

Austenitic steels, characterized by a *face-centered cubic* structure, contain some 17–18% chromium, 8–12% nickel (this enhancing corrosion resistance: the greater part, by far, of stainless steels are austenitic steels), little carbon, possibly some molybdenum, titanium, or niobium, and, mainly, iron (the remainder). They exhibit remarkable **ductility**, and **toughness**, a high expansion coefficient, and a lower **heat conductivity** coefficient than found in ferritic-martensitic steels. Of the main grades (coming under US references AISI⁽²⁾ 301 to 303, 304, 308, 316, 316L, 316LN, 316Ti, 316Cb, 318, 321, 330, 347), 304 and 316 steels proved particularly important for the nuclear industry, before being abandoned owing to their excessive swelling under irradiation. Some derivatives (e.g. 304L, used for internal structures and fuel assembly end-caps, in PWRs; or 316Ti, 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 **Zircalloys** (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 UO₂, 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

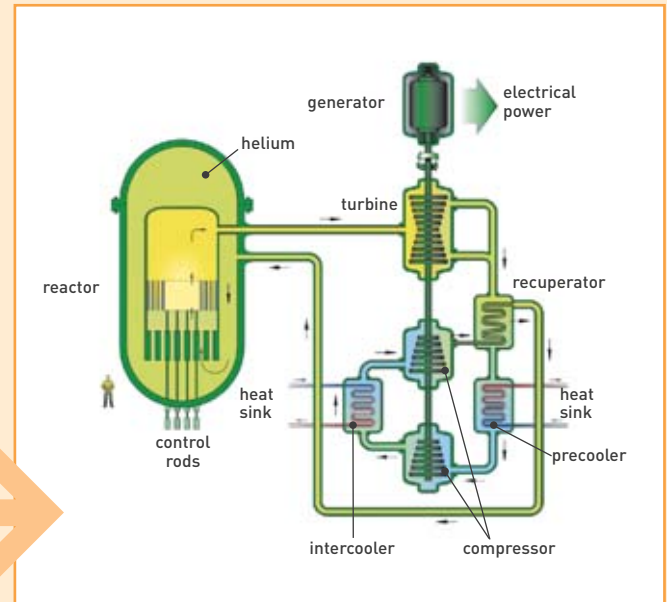
Of the six concepts selected by the **Generation IV International Forum** for their ability to meet the criteria outlined, three – and ultimately four – make use of **fast neutrons**, while three (ultimately two) use **thermal neutrons**. At the same time, two of the six concepts use gas as a coolant (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.

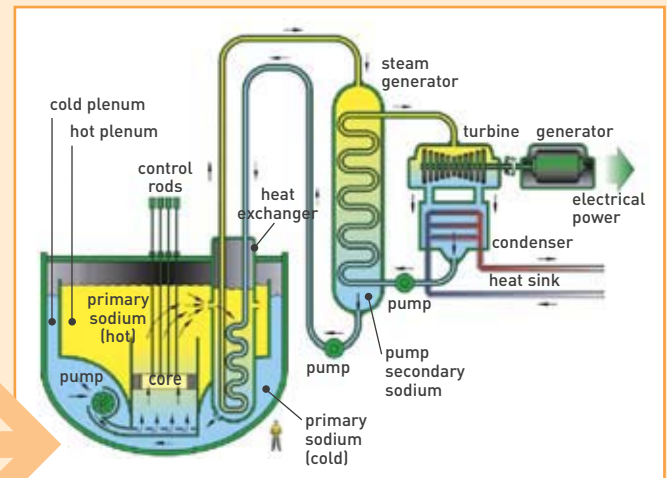
(1) GT-MHR: Gas-Turbine Modular Helium Reactor.

(2) PBMR: Pebble-Bed Modular Reactor.



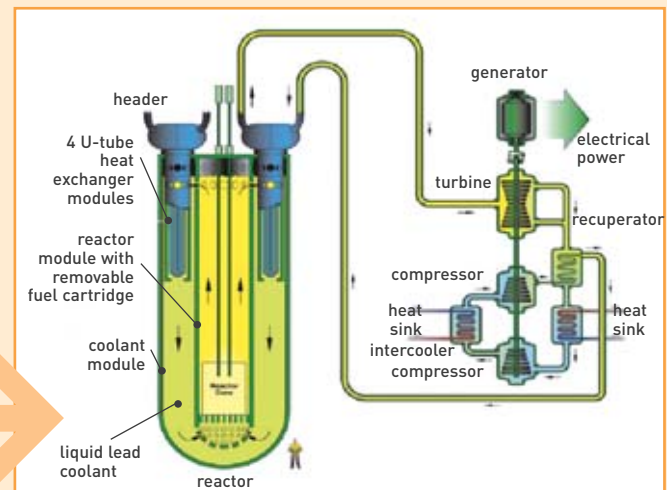
Le SFR

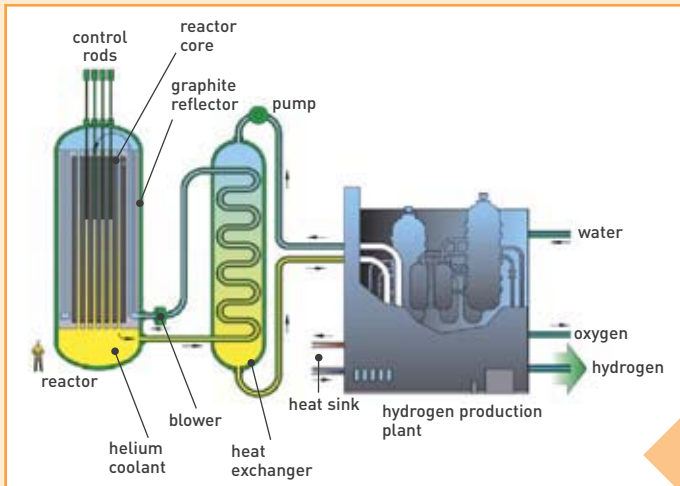
The sodium-cooled fast reactor system (**SFR**) is a liquid-**sodium**-cooled, fast-neutron reactor, associated to a **closed cycle**, allowing full actinide recycle, and **plutonium** breeding. Owing to its breeding of **fissile** material, this type of reactor may operate for highly extended periods 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 **radio-active**, 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 CO₂** as the working fluid for the power conversion system is also being investigated.



LFR

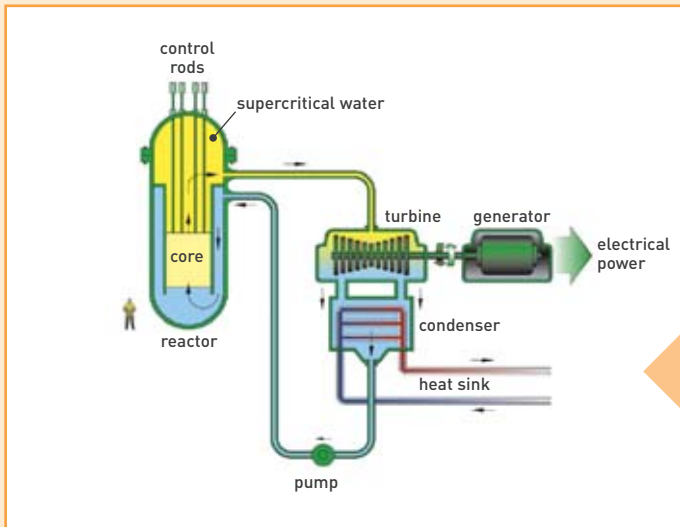
The lead-cooled fast reactor system (**LFR**) is a lead- (or lead-bismuth alloy-) cooled, fast-neutron reactor, associated to a closed fuel cycle, allowing optimum **uranium** utilization. A number of reference systems have been selected. Unit power ranges from the 50–100 MWe bracket, for so-called battery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





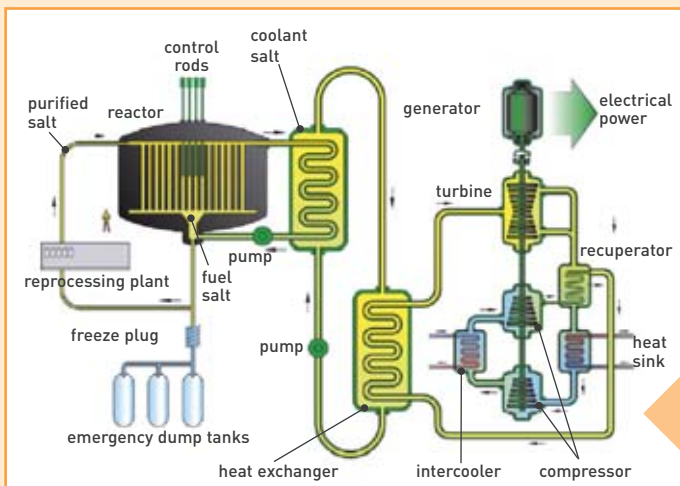
VHTR

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 ($> 1,000\text{ }^{\circ}\text{C}$), to provide the heat required for water splitting processes, by way of **thermo-chemical** 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 $550\text{ }^{\circ}\text{C}$, 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 in 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.