Towards nuclear energy **applications other** than **electricity** production

Use of nuclear energy relies on operation of a boiler, involving practically no greenhouse gas emission. Whereas production of electricity is, nowadays, virtually its sole purpose, demand for heat production could equally arise, particularly with the emergence of high-temperature, or even very-high-temperature reactors. With the abilities this involves, as regards the recovery of heavy crude oils from tar sands, seawater desalination, or, most importantly, production of hydrogen by electrolysis, or thermochemistry, this being the energy carrier of tomorrow.

Nuclear energy, to date, has had as its essential application the production of electricity. It can, however, equally produce the heat required for industrial processes, in the context of **cogeneration** concepts, which are at the same time effective in terms of energy efficiency, and produce no **greenhouse gases**. Why go for cogeneration, rather than a dedicated reactor, for a given application? Because the cost of a nuclear reactor – both design and construction costs – is so high that it would not be economic to design one reactor for each type of application, or, in particular, achieve a level of thermal power precisely matched to every need. When designing one standard model (or two at most), its power will most likely prove not to be matched to the heat application; however, the excess heat, not required for the industrial process, can always be converted into electricity, distributed over the grid, thus ensuring optimum utilization of the power from the nuclear boiler. Figure 1 shows a number of these applications, ranged according to the temperature required by the process, and the type of reactor that may meet this requirement.

The insertion of a nuclear power plant into an industrial environment does however raise a number of specific issues, aside from technical ones, which are not be underestimated (education of the players involved, political decisions, social acceptance, safety of supplies, control of materials). Potentially however, as shown in Figure 1, the areas in which nuclear heat may be used are diverse, and many. Three examples may serve as illustrations of such applications: one, at low temperature, is seawater desalination; another, medium-temperature instance is recovery of heavy crude oils from tar sands;[1] and a further, high-temperature use is **hydrogen** production by advanced high-temperature **electrolysis** processes.

With regard to applications involving use of heat, the relevant transport constraint should be borne in mind, restricting as it does to a few tens of kilometers low-temperature heat transport (T < 150 °C, typically for urban district heating), to a few kilometers medium-

CEA/DEN/DTEC/STCF

Charged with iodine in November 2006, the Bunsen reactor, for the investigation of the iodine–sulfur thermochemical cycle, constructed by CEA, and photographed at Marcoule before being sent to the United States, where it is to be coupled to the other two components of a complete pilot plant, under the aegis of a CEA–DOE agreement. The dissociation installations for sulfuric acid, and hydriodic acid are constructed, respectively, by Sandia National Laboratories, and General Atomics.

temperature transport (T ~ 300–500 °C), and to one kilometer or less very-high-temperature transport (T > 800°C).

Seawater desalination

To counter the aridity of certain regions, and mitigate the scarcity of potable water, seawater desalination is commonly resorted to, particularly in Gulf states, and

---

(1) Tar sands: a mixture of crude tar, sand, clay, and water, the largest deposits being found in Canada and Venezuela, providing an "unconventional" source of hydrocarbons.
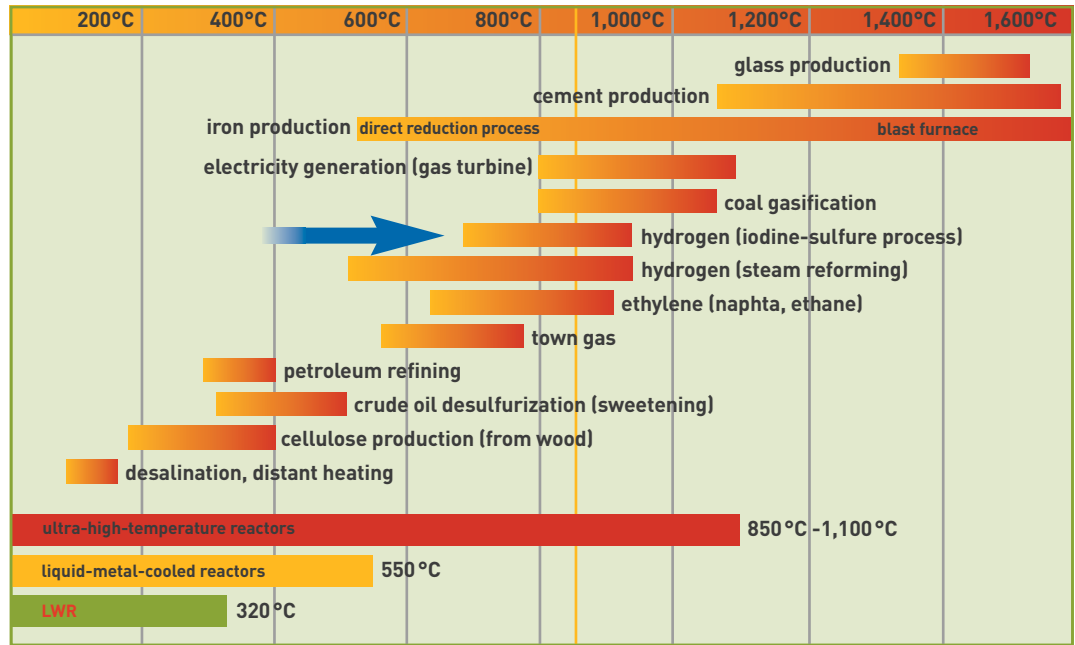
Figure 1. Temperatures required for various industrial processes (source: Forschungszentrum Jülich, 2002).

around the Mediterranean. Industrial techniques for large-scale desalination fall into two families: those involving bringing seawater to the boil, followed by condensation in a separate chamber (multistage flash [MSF] desalination, and multi-effect distillation [MED]), and those involving reverse-osmosis[2] (RO) filtration.

The former group involves direct use of low-temperature heat, at around 70 °C at the head end of the plant, to evaporate water in a chamber at subatmospheric pressure, the heat recuperated from the condensate being used in the next chamber, and so on, through some 7–10 successive chambers (stages), or even more. It will be seen that the energy used is mostly thermal energy, even though a few electric pumps are required, to transport water from one stage to the next. The heat supply for the desalination plant may be obtained by diverting low-pressure steam from a power-plant steam-turbine stage. This technique is commonly used, coupled with a conventional electricity production plant, or a cogeneration nuclear plant. The advantage afforded by the latter is the lack of any associated greenhouse gas generation.

In the second family, seawater is brought to high pressure – around 7 **MPa** – and filtrated through specific membranes. The greater part of the salts remains trapped in the membranes, which must be regenerated at regular intervals. The energy consumed, in this case, is purely electrical, in pumps, although recent concepts have shown the benefit attached to preheating water to about 40 °C prior to filtration, to enhance efficiency. On average, reverse osmosis uses about 5 kWh (electric) per cubic meter desalinated water.

Where large-volume production is being considered, use of nuclear energy is increasingly looked to, owing to the high cost of **fossil** fuels, and the associated greenhouse gas generation involved. Over the past few years, reverse osmosis would appear to have taken the lead, over evaporation techniques.

(2) Reverse osmosis: a separation process involving the transport through a semi-permeable membrane of the solvent, from a concentrated solution to a dilute solution.

This application is often claimed to number among those calling for high-temperature reactors (**HTRs**). It will clearly be seen that these afford no particular benefit with regard to this application, which only involves low-temperature heat, or use of electricity.

Recovery of heavy crude oils

The recovery of crude oils from tar sands provides a good example of potential large-scale use of medium-temperature nuclear heat. For many years yet, it will prove highly impractical to find substitutes for liquid fuels obtained from petroleum, namely gasoline and diesel fuel, for transportation purposes, owing to their ease of use. However, crude oils tend to be increasingly heavy, and difficult to extract. To take an example, currently experiencing heady expansion, mention should be made of the tar sands, or oil sands, of the Canadian Province of Alberta, for which oil reserves are estimated to be equivalent to those of Saudi Arabia, with stated production intentions rising from a current output of 1 million barrels per day to 5 million bbl/d by 2025–30.

The issues raised by extraction of these tars, buried as they are in geological strata ranging from the surface to several hundreds of meters down, relate to the high viscosity of the product *in situ* (comparable to that of butter inside a refrigerator), which however is perfectly fluid at 70 °C (just like butter at that temperature). A steam injection process (see Figure 2), known as steam-assisted gravity drainage (SAGD), coupled with pumping of the steam–tar mixture, has been developed, and its effectiveness has been proven, as regards operations on these fields. Such an operation obviously requires, in terms both of steam and electricity (for mixture pumping, and transport), large amounts of energy, currently provided, essentially, by burning natural gas. The following consumptions are representative of a typical output of 100,000 barrels/day: 912 MWth used for generation of 460 kg/s steam at 300 °C and 10 MPa (this being the pressure required for steam transport over several kilometers, and injection into

the wells), and about 120 MWe consumed by pumps and various utilities.

High-temperature reactors are particularly well suited to this type of application, in cogeneration mode, owing to their ability to generate steam as specified, together with the electrical power required: two 600-MWth modules such as ANTARES (see *High-temperature reactors: a recent past, a near future,* p. 51) yield the full complement of energy required for an SAGD output of 100,000 bbl/d, with no upgrading.[3]

The power levels involved would benefit from a nuclear plant operating in cogeneration mode, which would obviate burning large amounts of natural gas (some 2.5 million m³/d), along with the related pollution (at least 6,000 tonnes CO_2 per day). Extrapolated to the output levels projected for 2025, the energy and ecology implications of such an application become apparent.

Hydrogen production

Climate warming, and increasing scarcity of oil resources present major challenges, the reality of which is now proven, but equally afford opportunities for nuclear energy. Finding feasible, reasonable paths to meet this sets real challenges in turn. In this context, the combined production of hydrogen and electricity by means of nuclear energy, allowing a mass supply of hydrogen involving no greenhouse gas emissions, could prove advantageous, provided it is economically acceptable.

Industrial hydrogen is chiefly produced, nowadays, by the reforming[4] of natural gas (48%), or of liquid **hydrocarbons** (18%), and by coal gasification (18%), processes that generate large amounts of CO_2, while

(3) Upgrading involves the hydrogen enrichment of tars, turning them into a synthetic petroleum product, comparable to light crude oil. The production of hydrogen, and its incorporation into the tar involve energy consumption and CO2 production, which are not taken into account here.

(4) Reforming: a chemical cracking operation, applied to a hydrocarbon or alcohol molecule, to convert it into its major constituents.

(5) Reference: Jérôme Perrin, "L'hydrogène et l'électricité, un mariage de raison", *Revue géopolitique*, No. 93, 2006.

(6) Endothermic/exothermic: involving a concomitant absorption/release of heat.

(7) Electrolyte: a liquid or solid medium enabling ion conduction; the electrodes being the conducting elements ensuring the functions of electron or ion emission, or capture.

water electrolysis, the only process involving no CO_2 emission, only accounts for 4% of production.[5] All gasification and reforming processes yield a mixture, consisting of hydrogen, CO, and CO_2.

Hydrogen production processes involving use of nuclear energy fall into two families, electrolytic processes, and **thermochemical** processes (some hybrid processes involving both techniques), these involving a series of reactions, some of them endothermic,[6] with elements (iodine, sulfur, bromine…) that are fully recycled, so that only water, and heat stand as the "raw materials" for a hydrogen, and oxygen production. Both pathways are currently targeted by investigations at CEA.
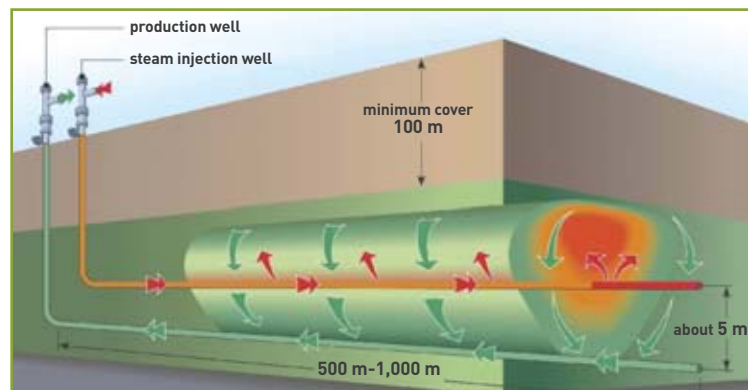
Electrolytic processes

Investigation of the advantages of, and optimum conditions for, the coupling of high-temperature electrolyzers (HTEs) to a **pressurized-water** nuclear **reactor** (PWR), or to a high-temperature nuclear reactor (HTR) is being addressed by research conducted both in Europe and in the United States.

Other types of reactor may also be considered, HTRs affording the advantage of being able to provide heat at higher temperature than other types.

The principle of HTE involves the electrolysis of water along a solid electrolyte,[7] at high temperatures (500–1,000 °C, depending on the technology selected). This makes it possible to achieve higher efficiencies than is feasible with conventional water electrolysis (60%, against 40%, referred to primary thermal energy). Part of the energy required is supplied in the form of electricity, the remainder in the form of heat. The increa-

Figure 2.
Principle schematic of steam-assisted gravity drainage (SAGD), for the recovery of heavy crude oils, as used by Total in the Surmont Concession pilot project, in the Athabasca area (Province of Alberta, Canada).

production well
steam injection well
minimum cover 100 m
500 m-1,000 m
about 5 m

Wellheads at the Joslyn site, in the Athabasca area (Province of Alberta, Canada), used by the Total Group for the extraction of tar sands (containing extra-heavy crude oil) by steam-assisted gravity drainage (SAGD process).

L. Zylberman /Total

The various component of an experimental ceramic high-temperature electrolysis (HTE) cell, developed under the aegis of the GENHEPIS project (GENérateur d'hydrogène à éléments plans et interconnexions souples: Plane-Component and Flexible-Interconnection Hydrogen Generator). Fabricated at CEA's Grenoble Center, on the principle of an inverse SOFC fuel cell, the stack comprises ceramic membranes (white) and electrodes (green).

D. Michon-Artechnique/CEA

sed temperature allows bringing down, at the same time, the relative share of electricity to be supplied, excess voltage at the electrodes,[7] and losses due to polarization. Two types of high-temperature electrolysis, both currently still at the research stage, may be considered: electrolysis by ionic conduction (O^{2-} **ion**), involving operations at 850–1,000 °C, and electrolysis by protonic conduction (**H+ proton**), operating at 400–600 °C. The main advantage afforded by HTE is its lower specific consumption (2.6–3.2 kWh/Nm³ hydrogen as against 4.5 kWh/Nm³ for conventional water electrolysis[8]). However, behavior of the materials for the various components in the electrolyzer is one of the main issues.

Areva NP has completed a study for an electrolysis plant, on the basis of electrolyzers using H+ protonic conduction, in order to estimate its cost, and compute hydrogen production costs. If the latter are to be brought down, it proves necessary to achieve significantly lower plant capital costs, in particular by increasing electrolysis cell size, and going for larger electrode active surface areas. Further, a technological breakthrough with respect to materials would likewise allow a significantly lower capital cost to be achieved (longer electrolyzer lifetimes, lower operating temperature).[9]
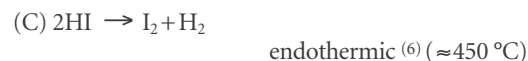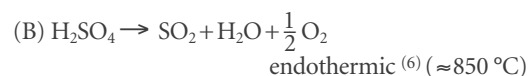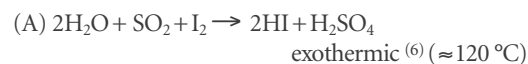
The cost of hydrogen obtained, on a large scale, by HTE would prove competitive ($\sim$ €2.5/kg hydrogen, using either technique), compared to that of hydrogen obtained by natural gas reforming, costing €9/GJ, assuming a carbon tax of $100/t,[10] for electricity priced at €37/MWh, with electrolyzer lifetimes longer than 5 years, and electrolysis cell costs about 30 times lower than the current cost of a prototype laboratory cell.

Thermochemical processes

Other processes, of a thermochemical kind, are being actively investigated at CEA, as in the United States, and Japan. These involve, as a rule, endothermic reac-

(8) Nm³ (normal cubic meter): 1 normal cubic meter (Nm³) of gas is a cubic meter of that gas at 0 °C, at a pressure of 101,325 Pa (1 atmosphere).

(9) Reference: Jean-Pierre Py, Alain Capitaine, "Hydrogen production by high temperature electrolysis of water vapour and nuclear reactors", 16th World Hydrogen Energy conference, 8–13 June 2006.

(10) By way of comparison, hydrogen production by conventional water electrolysis involves a higher cost ($\sim$ €2.8/kg H2, for electricity priced at €35MWh, output of 118,000 Nm³/h, a plant having a lifetime of 20 years, and a discount rate of 15%/year).

tions, exhibiting satisfactory efficiency at 800 °C, or at even higher temperatures. At such temperatures, only HTR reactors would have the ability to provide the required heat. Major technological challenges still have to be met, as regards heat transport, from the reactor to the chemical plant; catering for the aggressiveness of the chemical reactants involved (e.g. sulfuric acid); and the resistance of materials in such conditions. Siting a large plant very close to a nuclear power plant, or even on site, to circumscribe the heat transport issue, would unavoidably involve serious issues of principle (administrative agreements, etc.).

One of the most favored processes, currently, is the iodine–sulfur cycle, which may be represented by the following chemical reactions:

(A) $2H_2O + SO_2 + I_2 \longrightarrow 2HI + H_2SO_4$
exothermic [6] ($\approx$120 °C)

(B) $H_2SO_4 \longrightarrow SO_2 + H_2O + \frac{1}{2}O_2$
endothermic [6] ($\approx$850 °C)

(C) $2HI \longrightarrow I_2 + H_2$
endothermic [6] ($\approx$450 °C)

Investigations on these processes are at the laboratory stage, or at the pilot facility stage; anticipations of high performance are widely entertained, but have yet to be demonstrated.

Researchers are also investigating the feasibility of bringing down the temperature required for the iodine–sulfur path.

Practical deployment issues

Potential applications, outside of electricity, for nuclear energy are thus varied; however, the technical and institutional issues they raise should not be underestimated. First of all, the manufacturer looking to use the process heat has no desire, as a rule, to become the operator, or even the owner of a nuclear power plant: a new line of expertise, and business, has yet to be developed. The application using this heat must, moreover, be a heavy consumer (> 200 MWth), since a small nuclear power plant is hardly economic. Such applications are not that numerous. Finally, an industrial application frequently calls for continuous operation, over a number of years. Should several nuclear boilers then be put on line (costs!), to ensure the required redundancy, if only during reactor refueling?

The foregoing examples, heavy energy consumers as they are, do allow the potential contribution from nuclear energy to be appreciated, with respect to production free of greenhouse gas emissions, but equally allows the practical deployment issues to be to foreseen, that will arise. Strong fossil fuel price pressure, along with the taking on board of the economic aspects of environmental issues will no doubt be required, before any major growth can occur of nuclear energy for non-electrical power applications.

> **Michel Lecomte**
Scientific Adviser
to the HTR Project Director
Areva NP

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,

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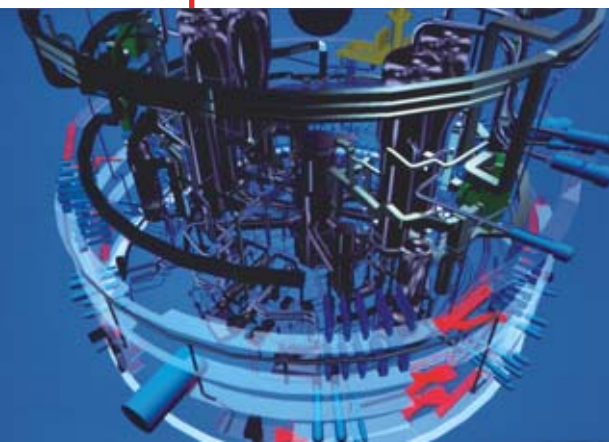
population to be regulated and, in the process, by acting on its reactivity, to hold reactor power at the desired level, or even to quench the chain reaction. The rods, held integral and moving as one unit (known as a cluster) are inserted more or less deeply into the core. Poisons, on the other hand,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

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

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.

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

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

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.

Thermodynamic cycles and energy conversion

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

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

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

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

Any system built around a nuclear generator is a heat engine, making use of the principles of thermodynamics. Just as fossil-fuel- (coal-, fuel oil-) burning thermal power plants, nuclear power plants use the heat from a "boiler," in this case delivered by **fuel elements**, inside which the **fission** processes occur. This heat is converted into electric energy, by making a fluid

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

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

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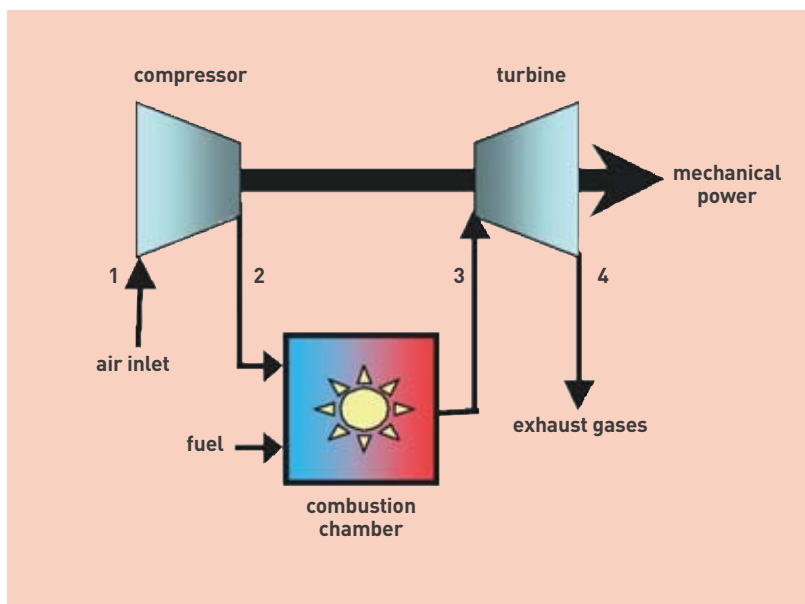

Figure.
Brayton cycle, as implemented in an open-cycle gas turbine.

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.
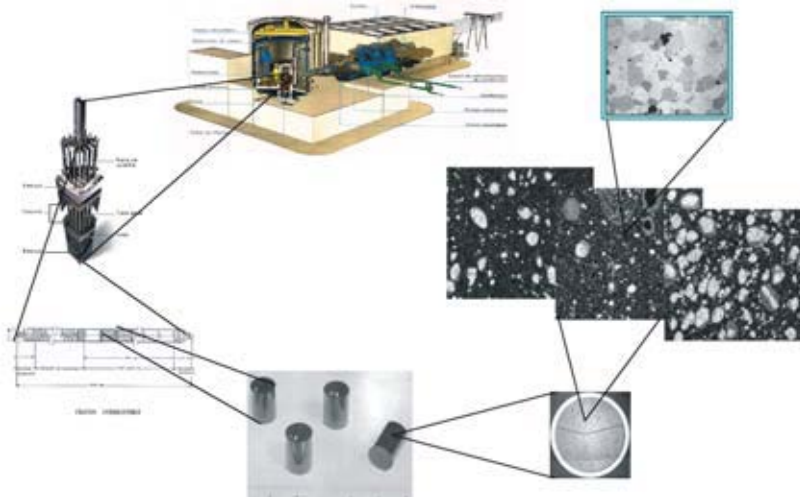

Figure.
Improving nuclear fuel reliability, and cost-effectiveness calls for finescale modeling of that fuel, through a multiscale approach, from reactor to fuel microstructure (in this instance, **MOX** fuel). Microstructural characteristics (porosity, **cluster** size and distribution, **grain** size...)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

(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.

extraneous **atoms** generated by nuclear reactions, substituting for one of the atoms in the **crystal** lattice. The nature, and number of such defects depends both on the neutron flux, and neutron energies, however the neutrons that cause appreciable structural evolutions are, in **thermal-neutron reactors** as in **fast-neutron reactors** (**fast reactors**), the **fast neutrons**.

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

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

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

Crystal structures

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

• the **body-centered cubic structure** (that found in iron at ambient room temperature, chromium, vanadium); such materials as a rule exhibit a ductile–brittle behavior transition, depending on temperature;

• the **face-centered cubic structure** (nickel, aluminum, copper, iron at high temperature);

• the **hexagonal structure** (that of zirconium, or titanium).

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

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

Steel families

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

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

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

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

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

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

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

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

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

Oxide-dispersion-strengthened (ODS) **ferritic and martensitic steels** were developed to combine the swelling resistance exhibited by ferritic steels, with a creep resistance in hot conditions at least equal

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

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

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

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

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

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

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

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

# The six concepts selected by the Gen IV Forum

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

GFR

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

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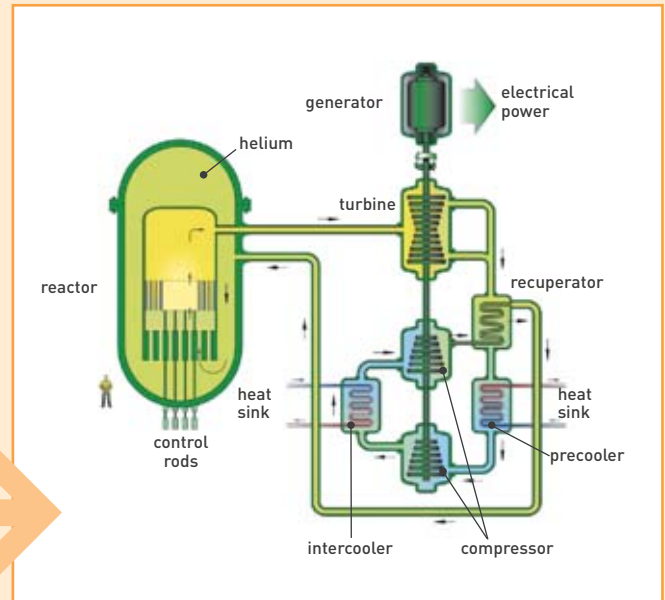
## Le SFR

The sodium-cooled fast reactor system (SFR) is a liquid-sodium-cooled, fast-neutron reactor, associated to a closed cycle, allowing full actinide recycle, and plutonium breeding. Owing to its breeding of fissile material, this type of reactor may operate for highly extended periods without requiring any intervention on the core. Two main options may be considered: one that, associated to the reprocessing of metallic fuel, results in a reactor of intermediate unit power, in the 150–500 MWe range; the other, characterized by the Purex reprocessing of mixed-oxide fuel (MOX), corresponds to a high-unit-power reactor, in the 500–1,500 MWe range. The SFR presents highly advantageous natural resource utilization and actinide management features. It has been assessed as exhibiting good safety characteristics. A number of SFR prototypes are to be found around the world, including Joyo and Monju in Japan, BN600 in Russia, and Phénix in France. The main issues for research concern the full recycling of actinides (actinide-bearing fuels are radioactive, and thus pose fabrication difficulties), in-service inspection (sodium not being transparent), safety (passive safety approaches are under investigation), and capital cost reduction. Substitution of water with supercritical CO_2 as the working fluid for the power conversion system is also being investigated
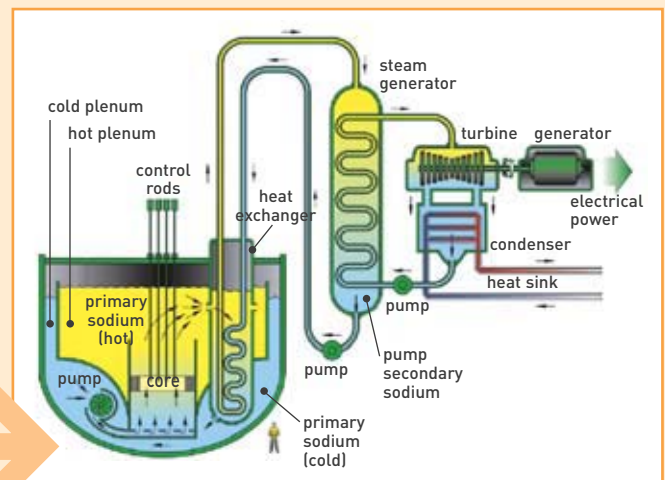
LFR
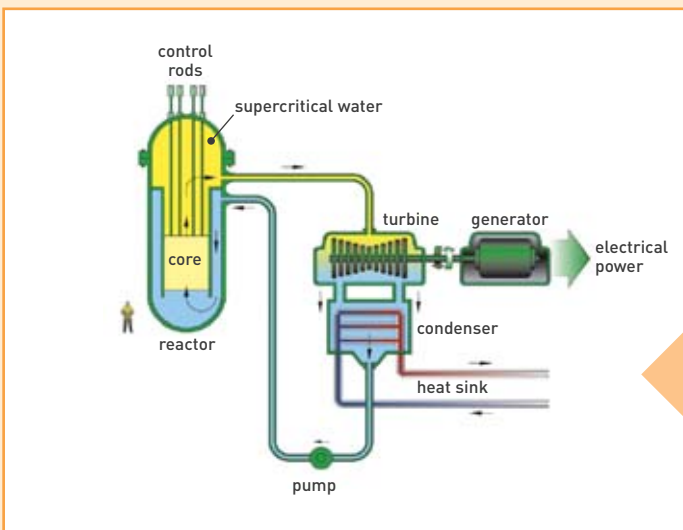
The lead-cooled fast reactor system (LFR) is a lead- (or lead–bismuth alloy-) cooled, fast-neutron reactor, associated to a closed fuel cycle, allowing optimum uranium utilization. A number of reference systems have been selected. Unit power ranges from the 50–100 MWe bracket, for so-called battery concepts, up to 1,200 MWe, including modular concepts in the 300–400 MWe bracket. The concepts feature long-duration (10–30 years) fuel management. Fuels may be either metallic, or of the nitride type, and allow full actinide recycle.
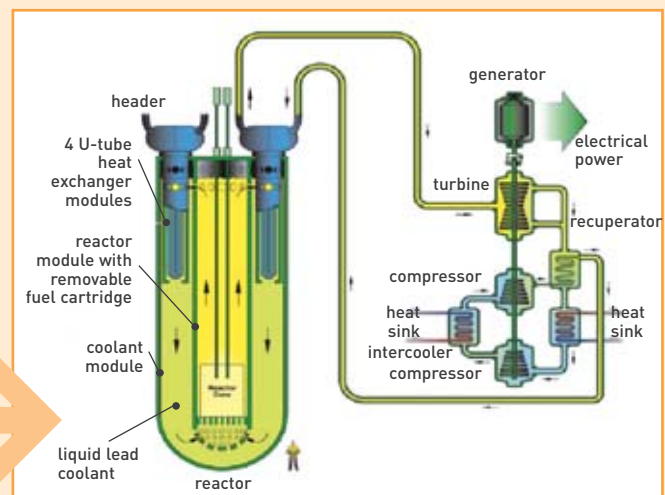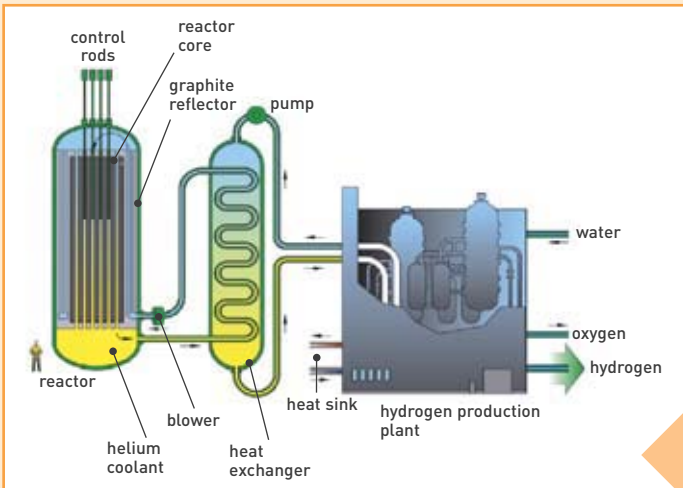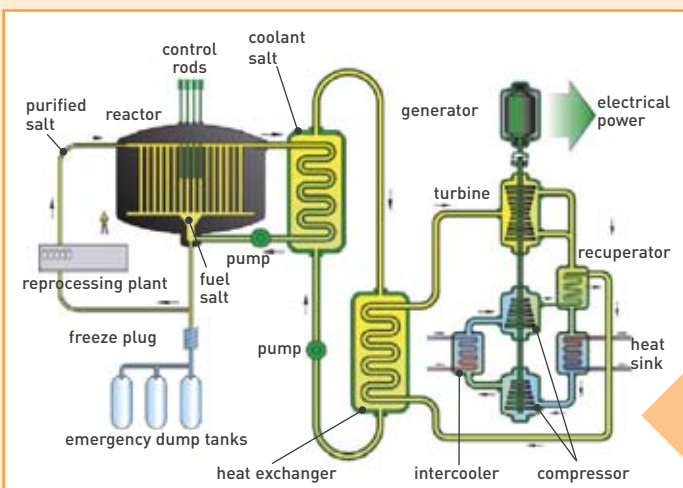
VHTR

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

VHTR diagram labels: control rods, reactor core, graphite reflector, pump, water, oxygen, hydrogen, hydrogen production plant, heat sink, heat exchanger, blower, helium coolant, reactor

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.

SCWR diagram labels: control rods, supercritical water, turbine, generator, electrical power, core, condenser, reactor, heat sink, pump

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.

MSR diagram labels: coolant salt, control rods, purified salt, reactor, generator, electrical power, turbine, recuperator, reprocessing plant, pump, fuel salt, heat sink, freeze plug, pump, emergency dump tanks, heat exchanger, intercooler, compressor