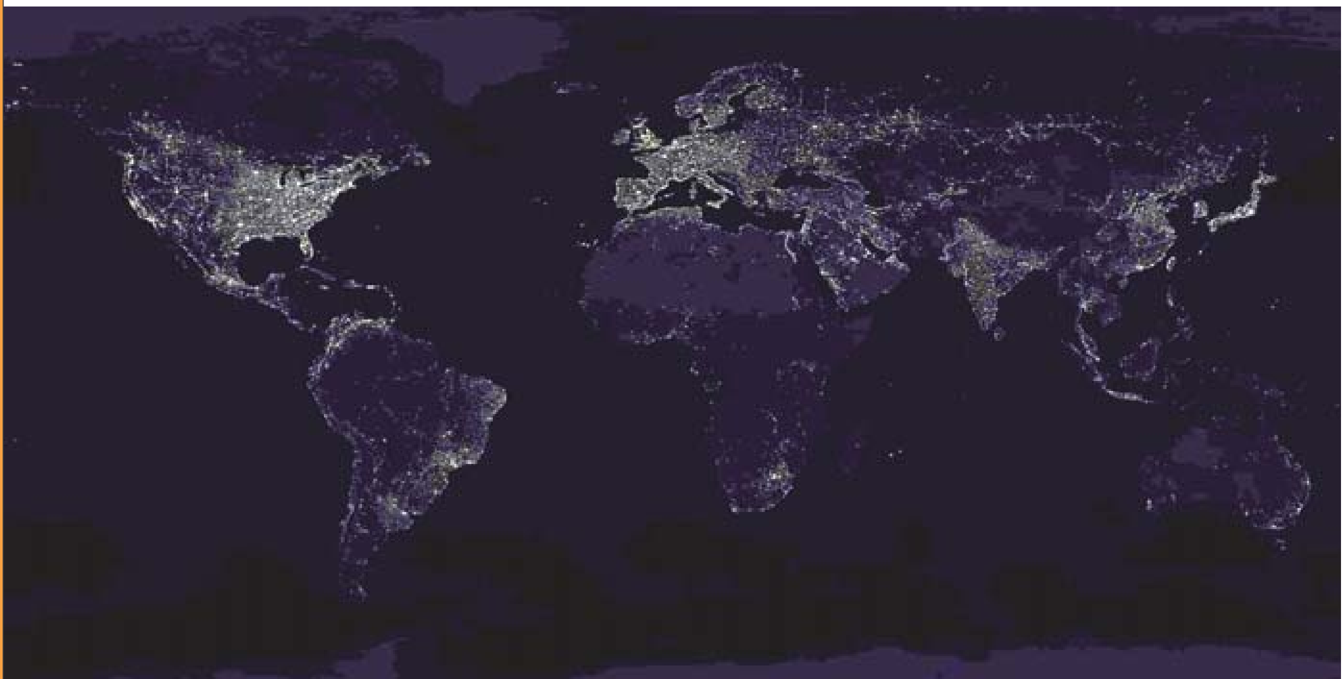
The challenges of **sustainable energy** production

At a time when the issue is to ensure – while protecting the environment –the security of energy supplies, and enable at the same time the development of emerging countries, sustained growth and living standards in developed countries, and guaranteed access to energy for the poorer countries, nuclear energy affords undoubted advantages.



C. Mayhew & R. Simmon (NASA/GSFC) ; NOAA/NGDC ; DSMP Digital Archive.

A mosaic of satellite photographs taken in nighttime, showing up the areas of high electricity consumption.

Globally, the present context is characterized, on the energy scene, by the realization there is a requirement for tighter integration, into any policy, of considerations pertaining to the rapid rise of global demand, to ensuring security of supply, as well as protection of the environment. Predictions of energy demand growth point, for the most conservative forecasts, to a doubling of world consumption by 2050 (20 **Gtoe**), to cater for a population of close to 9 billion humans (see Figure 1). Consequently, meeting humankind's energy needs presents one of the challenges, not to say the major challenge, for these opening years of the 21st century.

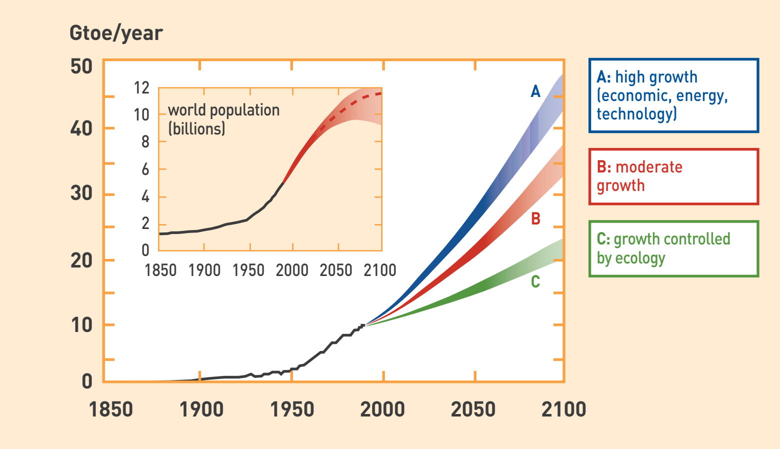


Figure 1.
Evolution of world population and primary energy needs (Source: International Institute for Applied Systems Analysis [IIASA] study for the World Energy Council: Global Energy Perspectives, 1998).

The role for nuclear energy in the global energy mix

There is no need to be reminded that one third of the world's population, as yet, does not have access to an energy system, or that up to 80% of energy consumption is concentrated among the industrialized countries. The rapid growth in developing countries (mainly China, and India, which together account for over one third of the world's population), persisting strain on the **hydrocarbons** market, the arduousness of effecting a change in consumption habits, call for the deployment of all energy sources. Nuclear energy takes its place among present-day solutions, being all the more advantageous for not releasing **greenhouse gases**, and for affording a huge potential for further growth, and development.
A variety of studies of long-term prospects, most notably that carried out by the **International Energy Agency** in 2003, have pointed to the conclusion that resorting to nuclear energy is inescapable, in such

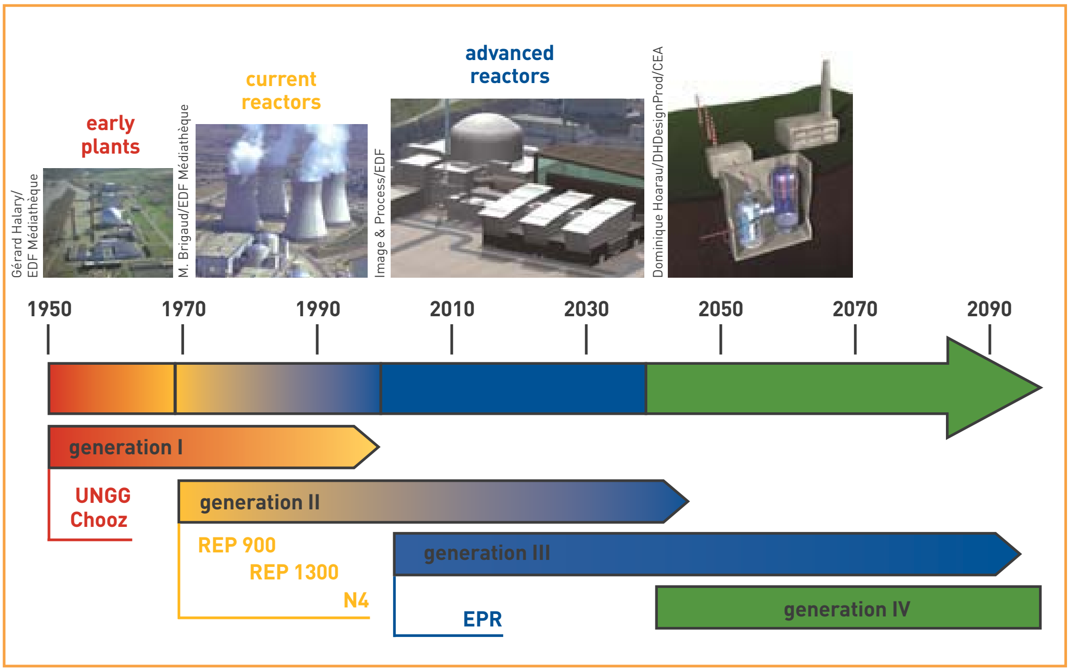


Figure 2.
Milestones of the various generations of nuclear reactors in France.

countries where this energy source is competitive, and may be safely developed; according to these studies, the share for nuclear energy in world electrical energy production should stand, by 2050, at 22%, as against 17% in 2002.

The requirement for a fourth generation of nuclear reactors

In this context, where nuclear energy is establishing its position as an economic, safe, and environmentally-friendly energy source, a number of programs are under way, to launch, or lay the ground for, industrial facilities, on the basis of evolutionary(1) models, drawing on the vast accumulated experience that has been gained, even as they bring further enhancements.
The need to prepare for a fourth generation of nuclear energy systems became apparent, on the international scene, some years ago, in the context of enquiries considering the initiatives that would be required, to meet a likely resurgence in demand for this energy source. In late 1999, the US Congress directed that this be made a major line of development. Early in 2000, Russia's President Putin made a speech in New York, during the ceremonies to mark the Millennium, suggesting a worldwide collaboration, to develop a fourth-generation system.
While matching, as a minimum, the safety and economic competitiveness levels achieved with the third generation, the goal is to secure sustainable development and growth of nuclear energy, with better utilization of resources, while minimizing waste and affording stronger resistance with regard to safety risks, **proliferation**, or terrorism. Further, a broader uptake of nuclear energy in the global energy mix will involve other utilization pathways, aside from electrical power generation: direct heating applications, **hydrogen** production, or seawater desalination units.
Going for a new generation is imperative, owing to the requirement for a switch in technologies, be it for reactors or fuel cycles, and thus for time to develop the innovative systems that will meet the new criteria. The inherent limitations of water reactors, whether in terms of efficiency or of **uranium** consumption, have focused the development effort on **high-temperature reactors**, and **fast-neutron reactors** (see Figure 2). Further, sustainability of the closed cycle on a global scale entails solutions that are effective with regard to ultimate waste management, and robust with respect to proliferation risks.

The Generation IV International Forum

The **Generation IV International Forum** (GIF, also known as Gen IV) was established in 2000, as the outcome of an initiative of the US Department of Energy (**DOE**). The aim was to share with the international nuclear community the thinking of US laboratories as to the future of nuclear power, and the innovations that would need to be developed, to ensure its continuing viability, while meeting the concerns of citizens.
Very early on, the Forum opted to act as an independent organization, aiming not only to elicit a consensus among its members as to the solutions that should be developed, but equally to set in place a strong international cooperation, to carry out the relevant work.
To date, the Forum brings together thirteen participants (Argentina, Brazil, Canada, China, **Euratom**, France, Japan, the Republic of Korea [South], Russia, South Africa, Switzerland, the United Kingdom, United States) who have all, at the outset, underwritten its Charter, whereby they recognize the importance of developing future systems for the production of nuclear energy, and the need to ensure optimum protection of the environment, while addressing proliferation risks, and whereby they commit themselves to work together for the development of such systems.

(1) Evolutionary: featuring development that occurs as a continuum with extant realities. Said in particular of a nuclear reactor technology involving no major technological breakthrough compared with the previous generation.

continued on page 8

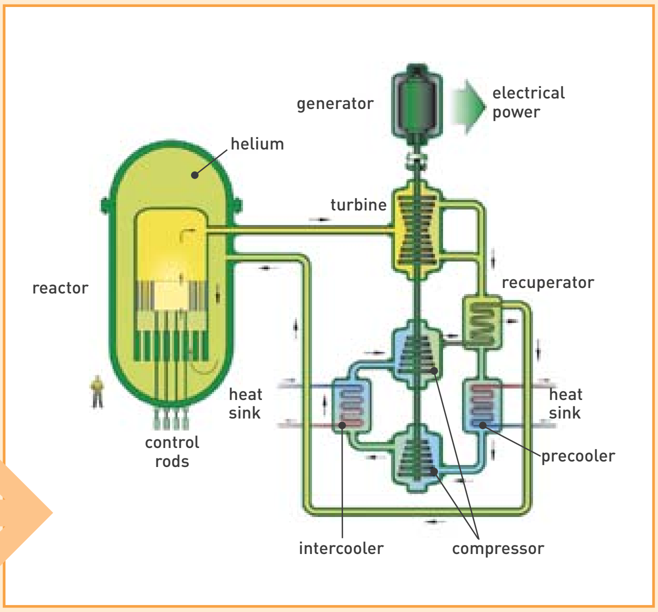
The six concepts selected by the Gen IV Forum

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

GFR

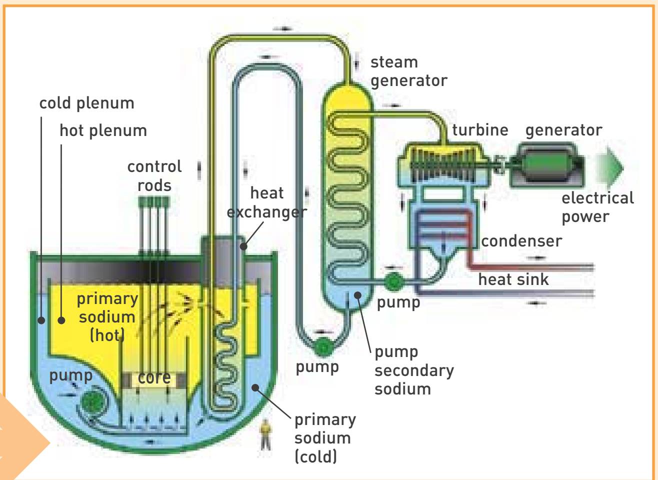
The gas-cooled fast reactor system (GFR) is a high-temperature, gas-cooled (helium-cooled as a rule), fast-neutron reactor allowing actinide recycle (homogeneous, or heterogeneous), while sustaining a breeding capability greater than unity. The reference concept is a helium-cooled, direct- or indirect-cycle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

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



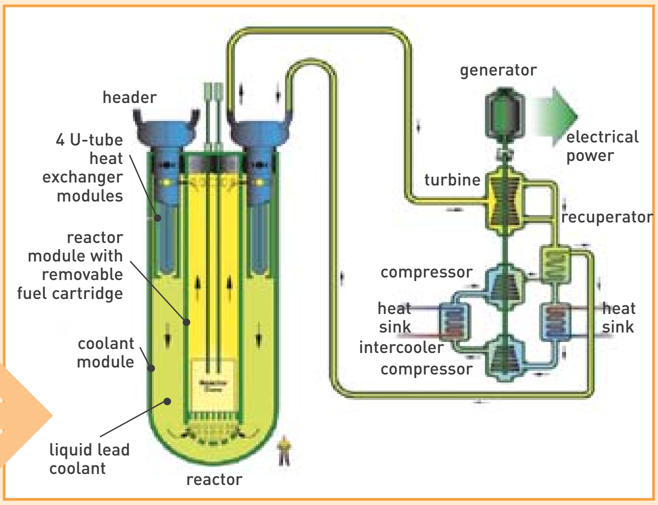
Le SFR

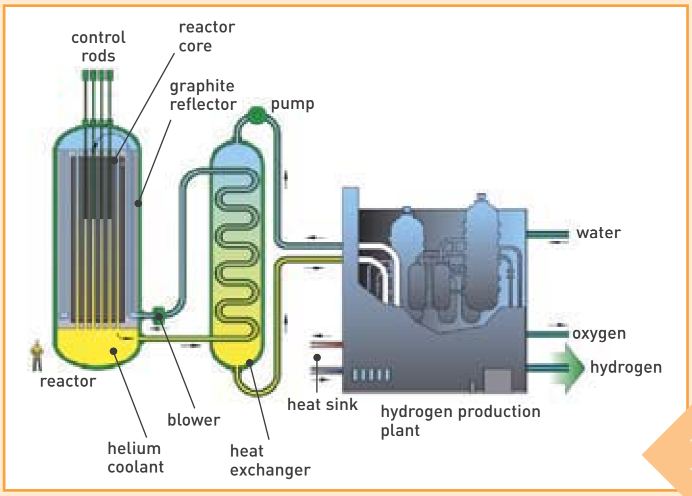
The sodium-cooled fast reactor system (SFR) is a liquid-sodium-cooled, fast-neutron reactor, associated to a closed cycle, allowing full actinide recycle, and plutonium breeding. Owing to its breeding of fissile material, this type of reactor may operate for highly extended periods without requiring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



LFR

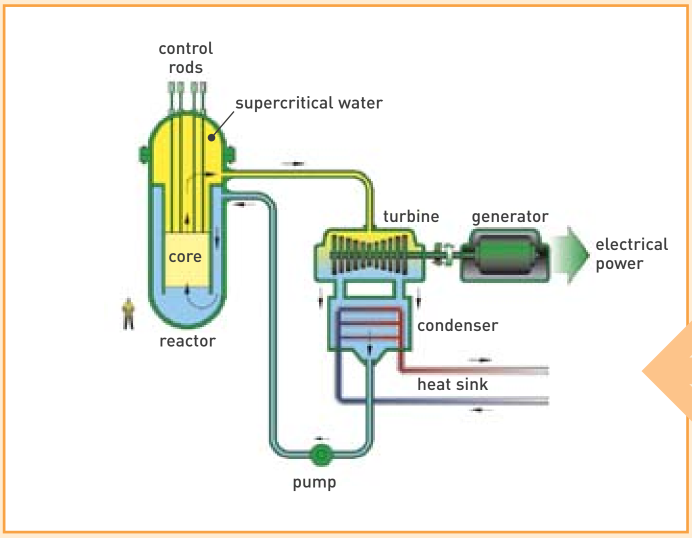
The lead-cooled fast reactor system (LFR) is a lead- (or lead–bismuth alloy-) cooled, fast-neutron reactor, associated to a closed fuel cycle, allowing optimum uranium utilization. A number of reference systems have been selected. Unit power ranges from the 50–100 MWe bracket, for so-called battery concepts,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





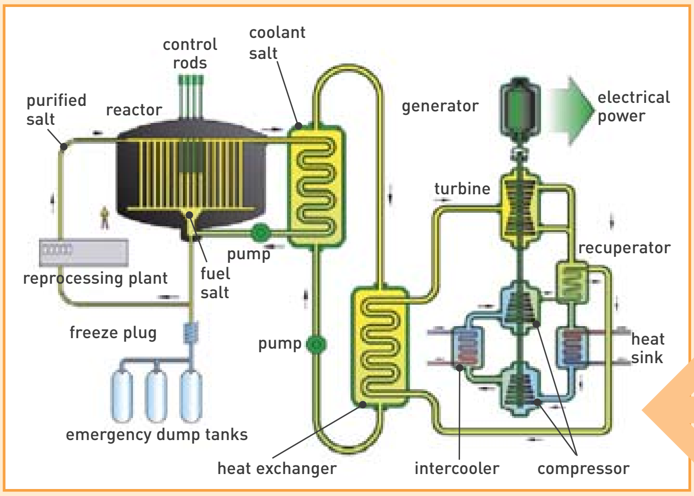
VHTR

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



SCWR

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



MSR

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

cont'd from page 5

An intergovernmental Framework Agreement, initially signed on 25 February 2005, hardens these commitments, setting out more detailed legal provisions for the venture, allowing for a variety of participants to be involved in the research and development (R&D) effort. Of the thirteen members in the Forum, ten have already signed the Framework Agreement, or confirmed their intention of so doing in the near future.

Argentina and Brazil opted to delay signing the Agreement, remaining for the time being with the status of non-active members. The United Kingdom signed the Framework Agreement in 2005, however it did not ratify it, opting to keep its participation restricted, for the immediate future, to its contribution to the European program.

Definition of the criteria to be selected for fourth-generation nuclear systems elicited, early on, a broad consensus within the Forum. Ambitious goals have been set in economic terms, with overnight(2) costs of some $1,000 per **kilowatt** (electric) installed, and production costs capped at $20 per **megawatt–hour**. The aimed-for safety level matches, as a minimum, that of third-generation reactors.

Proliferation resistance, and, more widely, expressly addressing security issues, was identified as one major criterion for the broader growth of civil nuclear energy. However, it was mainly the so-called sustainability criteria, resource conservation and waste minimization, that swayed the balance, when selecting the types of system to be developed.

GIF

Jacques Bouchard, Chairman for three years, starting from November 2006, of the Generation IV International Forum (GIF), speaking at a meeting of the GIF Policy Group, held in Paris in Fall 2006.

These twin criteria resulted in a majority of the systems selected being based on fast-neutron reactors, associated to a **closed cycle**, whereby all **actinides** are burnt. The favored solution would consist in carrying out the group partitioning and global recycling of all actinides – **uranium**, **plutonium**, and so-called **minor actinides** (neptunium, americium, curium). At the same time as meeting the requirement for adequate utilization of **natural uranium**, and of restricting waste to **fission products,** this solution affords, over the entire cycle, proliferation resistance such as to allow considering its worldwide dissemination. Other solutions may be considered, taking the form of more **heterogeneous recycling**, or involving separate processing of the products.

With respect to the development work to be carried out, the Forum further took into consideration the opening up of nuclear energy applications, by way of other pathways than electric energy production. This is imperative, if the share of nuclear energy in primary energy(3) consumption is to rise in a significant manner. It may provide a prime solution for the mass production of hydrogen from water, if this proves indispensable for energy supply for transportation. More broadly, the Forum looked into industrial applications of heat, and seawater desalination. On the basis of this set of criteria, the Forum selected six concepts, seen as the most promising (see Box), and drew up an R&D program with the aim of bringing forth the innovations required, to enable industrial deployment of models based on these concepts from 2030 on. For each system, the development plan covers three phases: the viability phase, the performance phase, and the demonstration phase. The intergovernmental Framework Agreement and the detailed agreements coming under it only cover the first two phases, the construction of demonstrations, pilots, prototypes or other such plants requiring further agreements, of a different kind. The Forum is minded, nevertheless, that, in the event, such demonstrations should be explicitly linked to the R&D effort, on the one hand, in order to benefit from the findings already gained, and to allow subsequent use of the data yielded by the demonstrations for concept development.

This entails that a number of challenges be taken up. Three may be mentioned.

The first challenge is that of development of a process to recycle all actinides. If it is to be effective with regard both to waste management and nonproliferation, full actinide recycle will need to comply with certain basic requirements: no partitioning of pure elements, of plutonium in particular; very low actinide quantities in ultimate waste; and efficient burning (through **fission**) of actinides within the reactor, to preclude the buildup of stored **heavy elements**.

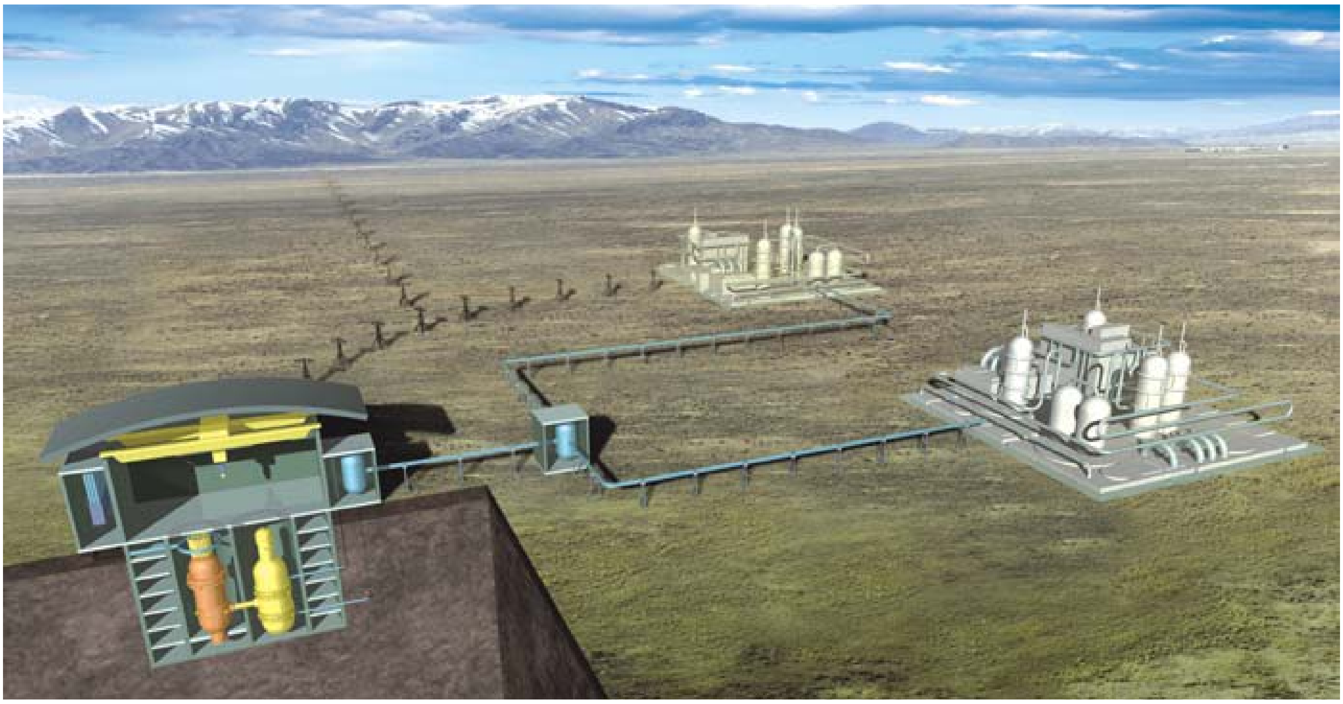
Fast reactors are best suited to meet these requirements, since they are used in **breeding** mode. The design, fabrication and performance of their **fuel** will be significantly affected by full actinide recycle, and will need to take in the concomitant choice of a **reprocessing** technique. Even though actinide content remains low, in a **homogeneous recycle** of this kind, fuel **radioactivity** (**gamma** radiation, and **neutron** emission) will entail a wholly remotely operated fabrication process, which has not, as yet, been developed to an industrial scale.

The second challenge concerns the development of fast reactors. A number of countries have extensive experience of sodium-cooled fast reactors, while Russia has constructed, for its naval equipment program, a number of lead-cooled reactors. As for the gas-cooled fast reactor, none have as yet been built. In its selection process, GIF opted to promote these three technologies, and even to consider a fourth technology: **supercritical**-water cooling.

For obvious reasons, choice of a fast reactor technology should remain open, as long as there is no requirement for industrial deployment in the near term. Sodium cooling is undoubtedly the most economic

(2) Overnight costs: the direct costs of an industrial investment, over the construction phase, without taking into account financial charges (interest on loans) incurred. This would correspond to the total cost of the facility, if it could be completed with no time delay ("overnight").

(3) Primary energy: a form of energy that may be directly accessed in nature, i.e., one of the fossil energies (coal, oil, natural gas), nuclear energy, and renewable energies (hydro energy, biomass energy, solar energy, wind energy, geothermal energy, tidal energy).



INL

The US NGNP (Next-Generation Nuclear Plant), investigated by the Idaho National Laboratory: an example of the association of a fourth-generation nuclear plant with hydrogen production.

option, however it labors under being generally poorly regarded, owing to its chemical reactivity, while many unfortunately still consider the termination of Superphénix to be proof of the failure of this solution (mistakenly, however the issue is not easily unraveled). Be that as it may, this technology requires an intermediate cooling circuit (or double-wall **steam generators**), involving extra costs.

Lead, or lead–bismuth, cooling is free from reactivity issues, however the **coolant** fluid is not as satisfactory, and corrosion hazards bring about new difficulties.

Helium cooling has been under investigation for many decades, however it has remained without any practical application. This affords the potential for achieving higher temperatures, and hence better efficiency; it further opens up an easier path for hydrogen production. On the other hand, this technology raises delicate issues, with regard to fuel design, and emergency cooling.

Supercritical-water cooling is a new area of investigation. This seeks to take advantage of the extensive experience gained with **light-water reactors**, while opening the way to higher temperatures, and a fast neutron **spectrum**. The issues that arise concern safety, owing to the specific characteristics of supercritical fluids, and **neutronics**, owing to these fluids' hydrogen content.

Most of the countries planning to build large numbers of nuclear power plants over the next two or three decades have no intention of turning to fast reactors in this context, with the sole exception of India, with the announcement of a tranche of ten 500-**MWe** sodium-cooled fast reactors over the next twenty years. On the other hand, a number of countries (the United States, Japan, France, Russia, or China) are planning to build prototype fast reactors in that timescale. Hence, it will undoubtedly prove fruitful to set up an international cooperation program, leading to better understanding of the various technologies, before choices are made, that may impact the future of an essential source for our energy supply, over many tens of years. There can be no underestimating the work that needs to be carried out to develop other cooling technologies, and bring proof of their viability. Nevertheless, a multiannual international cooperation program may, undoubtedly, result in enhanced understanding of these matters, and facilitate the indispensable choices to be made, as a prerequisite to the construction of new demonstration power plants in the decade from 2020, or beyond.

The third challenge that needs pointing out relates to hydrogen production. Nuclear energy may provide the means for the mass production of hydrogen through water splitting (via chemical or **electrolytic** processes), and thus contribute to setting up new energy carriers.[4]

The Generation IV R&D program ranges further afield than the issues relating to these three challenges. This is a highly ambitious program, the success of which will depend on strong international cooperation, and adequate leadtimes.

Since industrial deployment is scheduled for beyond 2030, this brings the ability to take the time required to develop innovative solutions, meeting the Gen IV criteria. Demonstrations, some of them full-scale, will likely be set up earlier than that date. Practical applications may also be taken up, at intervening stages, by some countries. In no way should such intermediate demonstrations or applications be allowed to bring about a premature termination of major development efforts, which may contribute to meeting more fully the energy needs of tomorrow.

> Jacques Bouchard
Chairman, Generation IV International Forum

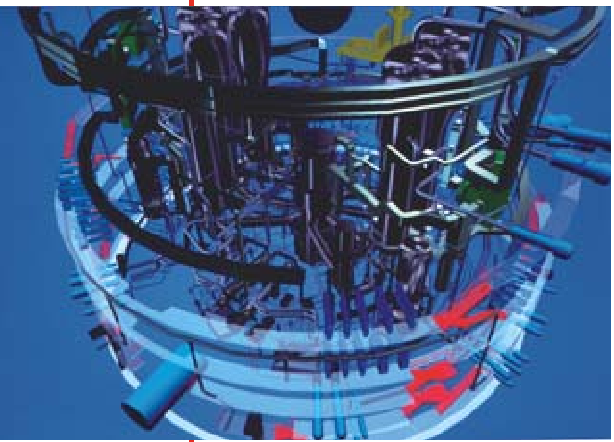
(4) Energy carrier: enabling the transport of energy. Currently, the two main such carriers are electricity, and heat; hydrogen may take up a major role in the future.

FOCUS A

The components of a nuclear system

A nuclear system comprises a nuclear reactor and the fuel cycle associated to it. It is the object of overall optimization, when industrially deployed – from raw materials to waste. In such a system, for which it forms the lynchpin, the reactor is given the ability to recycle fuel – so as to recover for value-added purposes fissile materials (uranium, plutonium), or even fertile materials (uranium, thorium) – and to minimize, through transmutation, production of long-lived waste, by burning, to a large extent, its own waste – namely, the minor actinides (MAs). Some systems may also feature online reprocessing plants.

The reactor itself, whichever technology line it may come under (see Focus B,



Areva NP

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

Reactor lines, generations, and neutron spectra, p. 14), invariably comprises the same main components (as regards fission technology at any rate, since fusion reactors make use of altogether different nuclear processes).

The core, i.e. the area where chain reactions are sustained, holds the fuel, bearing fissile, energy-yielding materials (heavy nuclei), as well as fertile materials which, subjected to the action of neutrons, turn in part into fissile materials. The fuel may come in a number of forms (pellets, pebbles, particles), and fuel elements may be brought together in rods, pins, or plates, these in turn being grouped together in assemblies, as is the case, in particular, in water-cooled reactors.

The moderator, when required, plays an essential part. This is a material consisting in light nuclei, which slow down neutrons by way of *elastic scattering*. It must exhibit low neutron-capture capability, if neutron "wastage" is to be avoided, and sufficient density to ensure effective slowing down. Thermal-spectrum reactors (see Focus B) require a moderator – as opposed to fast-spectrum reactors (which, on the other hand, must compensate for the low probability of fast-neutron-induced fission through a steep rise in neutron numbers) – to slow down the neutrons, subsequent to the fission that yielded them, to bring them down to the optimum velocity, thus ensuring in turn further fissions. One example of a moderator is graphite, which was used as early as the first atomic "pile," in 1942, associated to a gas as coolant fluid.

The coolant fluid removes from the core the thermal energy released by fission processes, and transports the calories to systems that will turn this energy into useable form, electricity as a rule. The coolant is either water,[1] in "water reactors" (where it also acts as moderator), or a liquid metal (sodium, or lead), or a gas (historically, carbon dioxide, and later helium, in gas-cooled reactors [GCRs]), or yet molten salts. In the last-mentioned case, fuel and coolant are one and the same fluid, affording the ability to reprocess nuclear materials on a continuous basis, since the actinides are dissolved in it.

The choice of technology line has major repercussions on the choice of materials (see Focus E, *The main families of nuclear materials*, p. 76). Thus, the core of fast-neutron reactors may not contain neutron-moderating substances (water, graphite), and their coolant must be transparent to such neutrons.

Control devices (on the one hand, control rods, or pilot and shutdown rods, made of neutron-absorbent materials [boron, cadmium…], and, on the other hand, neutron "poisons") allow the neutron population to be regulated and, in the process, by acting on its reactivity, to hold reactor power at the desired level, or even to quench the chain reaction. The rods, held integral and moving as one unit (known as a cluster) are inserted more or less deeply into the core. Poisons, on the other hand, may be adjusted in concentration within the cooling circuit.

A closed, leakproof, primary circuit contains the core, and channels and propels (by means of circulators – pumps or compressors) the coolant, which transfers its heat to a secondary circuit, by way of a heat exchanger, which may be a steam generator (this being the case equally in a pressurized-water reactor, or in the secondary circuit of a fast reactor such as Phénix). The reactor vessel, i.e. the vessel holding the core immersed in its cooling fluid, forms, in those cases when one is used, the main component of this primary circuit.

The secondary circuit extends out of the "nuclear island," to actuate, by way of a turbine, a turbo-alternator, or to feed a heat-distribution network. In heavy-water reactors,[1] and in some gas-cooled reactors, heat is transferred from gas to water in conventional heat exchangers.

A tertiary circuit takes off the unused heat, by way of a condenser, to a cold source (water in a river, or the sea), or the air in a cooling tower, or yet some other thermal device (e.g. for hydrogen production).

Other components are only found in certain reactor lines, such as the pressurizer in pressurized-water reactors (PWRs), where pressurization keeps the water in the liquid state by preventing it from boiling. On the other hand, boiling is put to work in boiling-water reactors (BWRs), the other line of light-water reactors (LWRs), where the primary circuit water comes to the boil, and directly actuates the turbine.

(1) *Heavy water*, in which deuterium is substituted for the hydrogen in ordinary water, was the first kind of moderator, used for reactor concepts requiring very low neutron absorption. *Light water* became the norm for operational, second-generation reactors. For the future, *supercritical water*, for which thermodynamic and transport properties are altered as it goes through the critical point (temperature of 374 °C, for a pressure higher than 22 MPa [221 bars, i.e. some 200 times atmospheric pressure]), may be used, to enhance the reactor's Carnot efficiency (see Focus C, *Thermodynamic cycles and energy conversion*, p. 23).

FOCUS B

Reactor lines, generations, and neutron spectra

M. Brigaud/ EDF Médiathèque

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

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

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

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

The term is used advisedly, implying as it does that this combination stands as the origin of a succession of reactors, exhibiting characteristics of a technological continuum. More or less directly related to this or that line are research and trials reactors, which are seldom built as a series.

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

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

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

FOCUS (Cond't) B

yielded by fission. A few reactors of this type have been built for industrial production purposes (Superphénix in France, BN600 in Russia), or investigated with such a purpose in mind (mainly **EFR**, a European endeavor, in the 1980s and 1990s, BN800 in Russia, CEFR in China, PFBR in India).

Electrical power generation reactors fall into four generations. The ***first generation*** covers reactors developed from the 1950s to the 1970s, which made possible the takeoff of nuclear electricity production in the various developed countries, comprising in particular the **UNGG** (or **NUGG**: **natural uranium–graphite–gas**) line, using graphite as moderator, and carbon dioxide as coolant, in France; the **Magnox** line, in the United Kingdom; and, in the United States, the first land-based[1] pressurized-water reactor (**PWR**), built at Shippingport.

While comparable in some respects to first-generation reactors, the Soviet Union's **RBMK** line (the technology used for the reactors at Chernobyl) is classed under the second generation, owing, in particular, to the time when it came on stream. RBMK reactors, using graphite as moderator, and cooled with ordinary water, brought to boil in pressure tubes, or channels, were finally disqualified by the accident at Chernobyl, in 1986.

The ***second generation*** covers those reactors, currently in service, that came on stream in the period from the 1970s to the 1990s. Solely built for electricity generation purposes, most of these (87% of the world fleet) are water-cooled reactors, with the one outstanding exception of the British-built **AGRs** (advanced gas-cooled reactors). The standard fuel they use consists of **sintered enriched uranium-oxide pellets**, to about 4% uranium-235 enrichment, stacked in impervious tubes (**rods**), which, held together in bundles, form **assemblies**. PWRs hold the lion's share of the market, accounting for 3 nuclear reactors out of 5 worldwide. This line includes the successive "levels" of PWR reactor models built, in France, by Framatome (now trading as **Areva NP**) for national power utility EDF. Russian reactors from the **VVER** 1000 line are comparable to the PWRs in the West. While operated in smaller numbers than PWRs, **BWRs** (boiling-water reactors) are to be found, in particular, in the United States, Japan, or Germany. Finally, natural-uranium powered reactors of the **CANDU** type, a Canadian design, and their Indian counterparts, form a line that is actively pursued. These are also pressurized-water reactors, however they use **heavy water** (D_2O) for their moderator, and coolant, hence the term **PHWR** (pressurized-heavy-water reactor) used to refer to this line.

The ***third generation*** corresponds to installations that are beginning to enter construction, scheduled to go on stream from around 2010. This covers, in particular, the French–German **EPR**, designed by Areva NP (initially: Framatome and Siemens), which company is also putting forward a boiling-water reactor, the SWR-1000, at the same time as it has been coming together with Japanese firm Mitsubishi Heavy Industries. This generation further includes the AP1000 and AP600 types from Westinghouse, a firm now controlled by Toshiba; the ESBWR and ABWR II from General Electric, now in association with Hitachi; the Canadian ACRs, and the AES92 from Russia; along with projects for smaller integral reactors.

Programs for modular **high-temperature reactors**, of the GT–MHR (an international program) or PBMR (from South African firm **Eskom**) type, belong to the third generation, however they may be seen as heralding ***fourth-generation*** reactors.

The fourth generation, currently being investigated, and scheduled for industrial deployment around 2040, could in theory involve any one of the six concepts selected by the **Generation IV International Forum** (see Box, in *The challenges of sustainable energy production*, p. 6). Aside from their use for electricity generation, reactors of this generation may have a **cogeneration** capability, i.e. for combined heat and power production, or even, for some of models, be designed solely for heat supply purposes, to provide either "low-temperature" (around 200 °C) heat, supplying urban heating networks, or "intermediate-temperature" (500–800 °C) heat, for industrial applications, of which seawater desalination is but one possibility, or yet "high- (or even very-high-) temperature" (1,000–1,200 °C) heat, for specific applications, such as **hydrogen** production, **biomass** gasification, or **hydrocarbon** cracking.

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

FOCUS C

Thermodynamic cycles and energy conversion

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

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

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

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

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

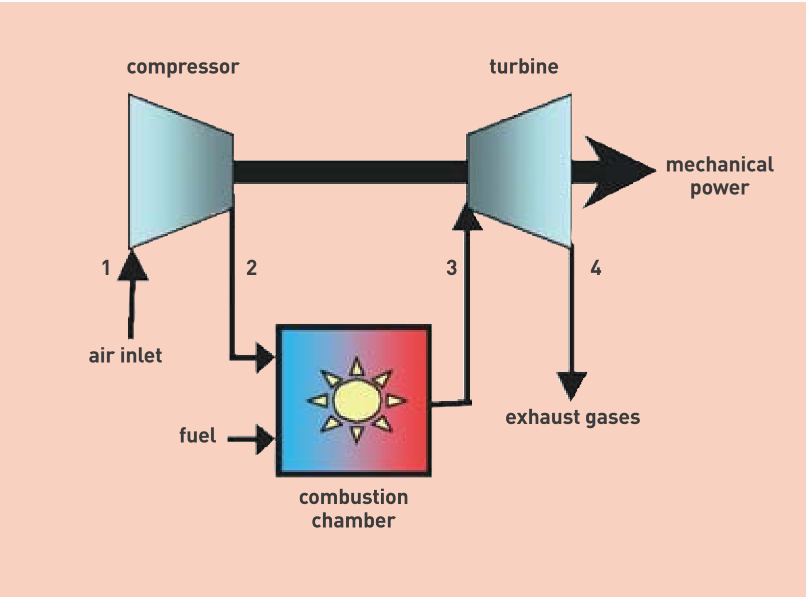


Figure.
Brayton cycle, as implemented in an open-cycle gas turbine.

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.

FOCUS D

What is multiphysics, multiscale modeling?

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

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

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

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

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

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

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

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

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

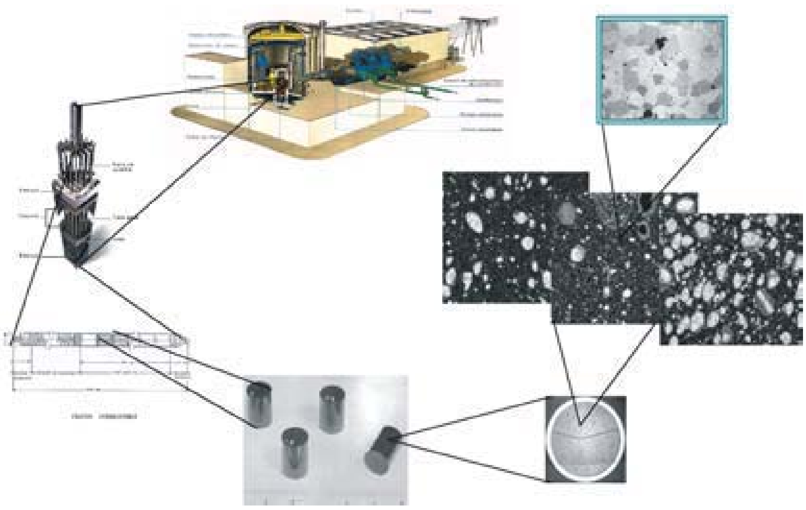


Figure.
Improving nuclear fuel reliability, and cost-effectiveness calls for finescale modeling of that fuel, through a multiscale approach, from reactor to fuel microstructure (in this instance, MOX fuel). Microstructural characteristics (porosity, cluster size and distribution, grain size...) have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

FOCUS E

The main families of nuclear materials

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

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

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

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

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

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

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

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

Crystal structures

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

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

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

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

Steel families

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

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

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

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

Areva NP

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

FOCUS (Cond't) E

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

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

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

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

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

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

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

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

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

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

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

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

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

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