

How will fourth-generation nuclear systems be deployed?



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If it is to be fully consonant with a sustainable development perspective, nuclear energy needs must rely on systems that will ensure the greatest optimization of fissile material utilization, while minimizing production of ultimate waste. Deployment scenarios, for systems of the new generation, remain very much open.

The Flamanville site (Manche département, western France), where EDF is building its first third-generation (EPR) reactor. In the background, the two existing second-generation units.

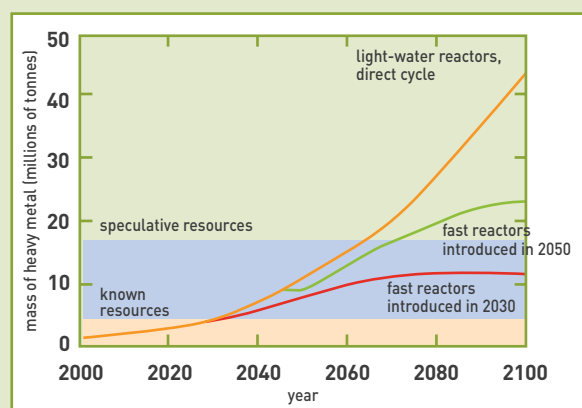
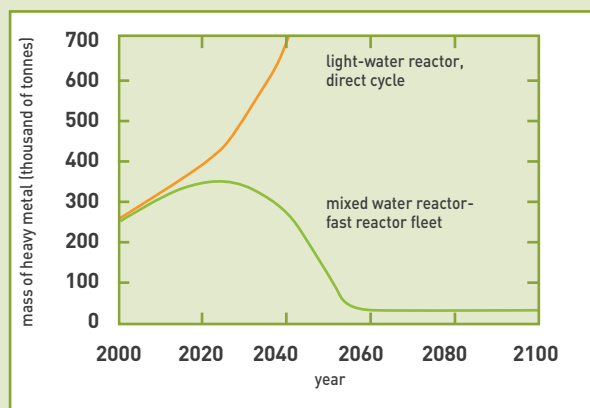
World energy needs are experiencing strong expansion. Increasing scarcity of fossil resources, and concerns relating to climate change will act as a restraint, against turning to fossil hydrocarbons to meet these needs. Renewable energies, to be sure, are bound to experience growth, however, by their very nature, they are too diffuse (solar energy, wind energy), too intermittent (wind), quantitatively limited (hydro energy), or still too expensive (solar energy) to claim a dominant share in the world energy mix. Strong growth as regards the share of nuclear energy in that mix is thus anticipated. How is this to be addressed? Cheap uranium resources are limited (see *What do we know of world uranium resources?* p. 17), and the resumption of nuclear energy programs, around the world, is fast consuming them. Strains on the uranium market, of which the first signs may already be observed, are to be expected: the price of natural uranium (Unat) has risen fivefold, over the past 5 years! Even with spent fuel recycling, to achieve maximum savings in natural resources, second- or third-generation nuclear system (see Focus B, *Reactor lines, generations, and neutron spectra*, p. 14) cannot be operated in sustainable

fashion, since they do not make efficient use of the energy potential from fertile nuclei (see Figures 1 and 2).

Actinide recycling

Sustainable nuclear energy will certainly entail the recurrent, thorough recycling of actinides, this being dictated, first of all, by the need to ensure a proper utilization of fertile resources. Only fast reactors (FRs) have this capability.

To obtain electrical energy corresponding to an output of 1 GW over one year, about 1.4 tonne of matter must undergo fission. For that purpose, an ordinary-water reactor (a PWR, for instance) requires some 140 tonnes of natural uranium, whereas with a breeder system, 1.4 tonne of fertile nuclei would suffice! Apart from the unavoidable “ashes” from nuclear reactions (fission products), fourth-generation reactors will have the ability to recycle all of the materials from spent fuel: materials still usable to yield energy, fertile materials to be turned into fissile material, and (as an option) minor actinides, in order to incinerate them.



Figures 1 and 2.

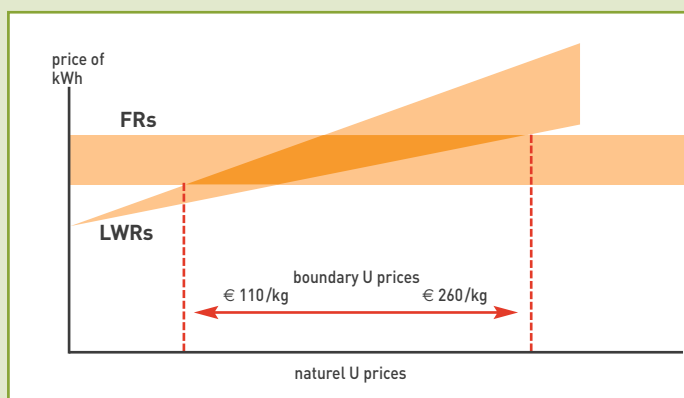
Forecasting indications: world "production" of spent fuel (at left), and evolution of world uranium requirements (right): water reactors are fast consuming fissile resources, and building up actinides. Fast reactors do not involve these drawbacks.

In this manner, these reactors will turn the natural resource into energy, and have the ability to burn their own waste (aside from fission products). This **closed fuel cycle** concept, i.e. such recycling of all of the fuel, is indissolubly bound up with the concept of sustainable nuclear energy.

The chief argument, warranting deployment of fast reactors remains that of savings in terms of fissile materials; that concerning their "clean" operation only comes second. Fast reactors may thus only be expected to emerge if – or when – their specific advantage, namely savings in fissile materials, becomes a key factor for success.

The contribution of fuel to the price of a **kilowatt-hour** is small, however it is bound to rise, chiefly because of the increasing price of natural uranium, due to the aforementioned reasons. A time will come when fourth-generation breeder systems will become economic, in spite of the higher capital cost they involve. The break-even point, for fast reactors, is largely dependent on the extra cost incurred, and natural uranium prices. For an **SFR** (sodium-cooled), assumed to involve a capital outlay greater by 25% than for a **light-water reactor (LWR)** of equivalent power, the competitive threshold lies in a range of about €110–260/kg Unat, depending on the assumptions made for the economic computation. Current Unat spot prices, of late, have entered this bracket. This may be a temporary

Figure 3.
With respect to light-water reactors (LWRs), the competitive threshold for fast reactors (FRs) is dependent on uranium prices. The range of figures shown here for a sodium-cooled fast reactor is taken from the findings of a technical-commercial study recently carried out at CEA (H. Safa et al., ICAPP 2007).



effect, due to speculative movements, however, should operators consider that that price is due to stay durably pegged at such levels, fast reactors will stand a good chance of being deployed, as it is likely that the economic argument will carry the day, over all other considerations. While the figures illustrating this paper carry the dates 2040–50, it must be said that there is still a great deal of uncertainty as to the actual date for such a transition (see Figure 3).

The issue of unit power, for fourth-generation reactors, is likely to loom large for their deployment. Should large reactors be considered, with the anticipation of thus achieving economies of scale, or small, modular reactors, affording advantages in terms of operational flexibility, spreading of capital outlay, ability to adjust to demand, and economies from standardization?

The demonstration of safety

There will also be a need for the safety of fourth-generation systems to be convincingly shown. Safety is one area where operational feedback plays a crucial part. Demonstrating such safety is thus a problematic phase, for reactors involving a break with existing technologies. In this respect, much work remains to be done, whichever type of reactor is contemplated! Fast reactors are inherently more "responsive" than their **thermal** counterparts, since their **neutron** balance involves fewer **delayed neutrons**. At the same time, the SFR is handicapped by sodium flammability, while the GFR suffers from its low thermal inertia. None of this would appear to be beyond a solution, however the demonstration of safety involves a process of building up confidence, which will necessarily take time.

Fourth-generation nuclear systems involve a technological breakthrough, and consequently require an extensive research effort. Such research work is currently conducted in a highly international context, chiefly through the **Generation IV International Forum**. To foster mastery of the technical aspects of fourth-generation systems, the President of France took the decision to launch, at CEA, design work for a prototype fourth-generation reactor, to be operational by 2020. That date was not chosen by chance. Indeed, there is an important given, that must be taken on

board in any scenario of nuclear energy deployment around the world: and that is the length of the time constants involved. Development of new reactor lines is a protracted, long-term effort, for which the ground must be laid well in advance (20 years, as evidenced by the transition from generations I to II, and later from II to III).

At the same time, the lifetime for such installations is invariably long: 40 years for second-generation reactors, 60 years for third-generation reactors. The 21st century will thus witness the coexistence of third- and fourth-generation systems, which will have to operate in a complementary, symbiotic fashion (see Figure 4).

A symbiosis between third and fourth generations

This symbiosis chiefly concerns the fuel cycle. Such coexistence could well prove fairly harmonious: the **depleted uranium** yielded by uranium **enrichment** operations, for LWR fuel, carefully stockpiled as it has been for 20 years, will find a use, as fertile material in fast reactors. At the same time, LWRs have the ability to generate **plutonium** (Pu), which the fast reactors could use for startup.

However, the speed with which fast reactors may be deployed across the world may be restricted by availability of materials. While it is true that a fast reactor with a power of 1 **GWe** only consumes one to two tonnes of (natural, or depleted) uranium, it does require, for startup, a considerable amount of fertile material: 40 t uranium, enriched to 20%, or 10 t plutonium (this then being bred constantly). A typical **UOX** PWR (1 GWe) yields 210 kg plutonium per year, which has made it possible to build up, in France, a Pu inventory of some 270 t, present in spent fuel. Currently, recycling of Pu, in the form of **MOX** fuel in some water reactors in the present fleet, allows the growth of this

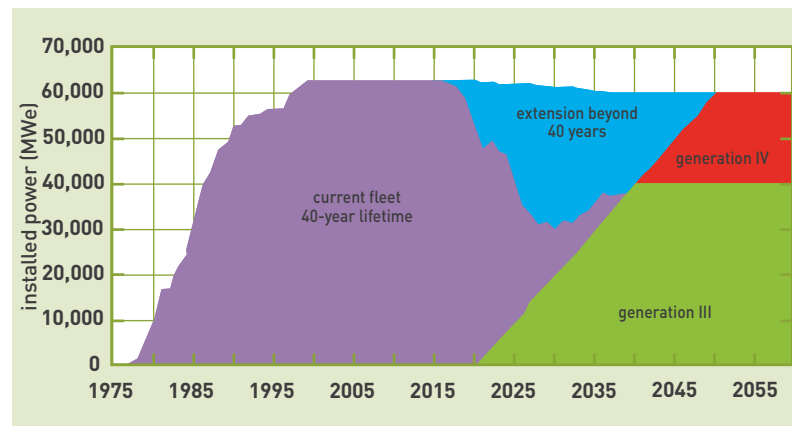


Figure 4. Replacement scenario for the French nuclear reactor fleet, as envisaged by EDF, according to a highly conservative assumption, of a slight reduction in overall installed power, down to 60 GW. The operator will doubtless seek to extend the lifetime of existing reactors, as far as is reasonably feasible, and allowed by regulations. The plan is to initiate replacement of part of the fleet in a “wedge” approach, as early as 2020, to even out the funding effort, initially with third-generation, then fourth-generation reactors.

inventory to be kept under control – an inventory that may, in the longer term, turn into a strategic resource for fast reactor expansion. Depending on the way it is utilized in the meantime, in the form of MOX in water reactors, the French Pu stockpile, in 2040, should stand somewhere between 300 t and 500 t, this being sufficient for the startup of 30–50 fast reactors. As may be seen, France is well equipped to ensure deployment of fast reactors, however, the same does not hold for the rest of the world. For the purposes of ensuring availability of the plutonium required, countries that, as China, or India, do not have stocks of spent fuel, or do not carry out the reprocessing of such fuel will have to meet a number of conditions.

They will, first of all, have to change their policy, and achieve a change in mindsets, so that plutonium can pass from the status of waste, to that of recoverable resource. They will further need to have operated, over a long enough interval, a sufficient number of water reactors to have enough plutonium available in spent fuel, bearing in mind that it takes about 50 reactor-years to produce the amount of Pu required for the startup

Reactor generations, and associated fuel cycles

The sequence of reactor generations naturally entails that of the associated fuel cycles (see Figure). In France, for instance, **plutonium** (Pu) from **PWRs** is currently **recycled** in the form of **MOX**. In 2020, the **Generation-II** PWRs, primarily consuming uranium (U), will still be around, however the Pu they generate will be burnt (partly, but more efficiently) by the **Generation-III** reactors deployed by that time. The **minor actinides** (MAs) yielded by this mixed “Gen II–Gen III” fleet could then be **partitioned**, and stored. In 2040, the first **Generation-IV fast reactors** will be deployed, burning the Pu that will have been set aside for their startup, together with – possibly – the minor actinides that will have been building up. The uranium complement required for their

operation could be provided by **depleted uranium**, currently stockpiled. Around 2050, such “Gen IV” reactors

should have the ability to operate by recycling their own actinides, including (as an option) their own minor actinides.

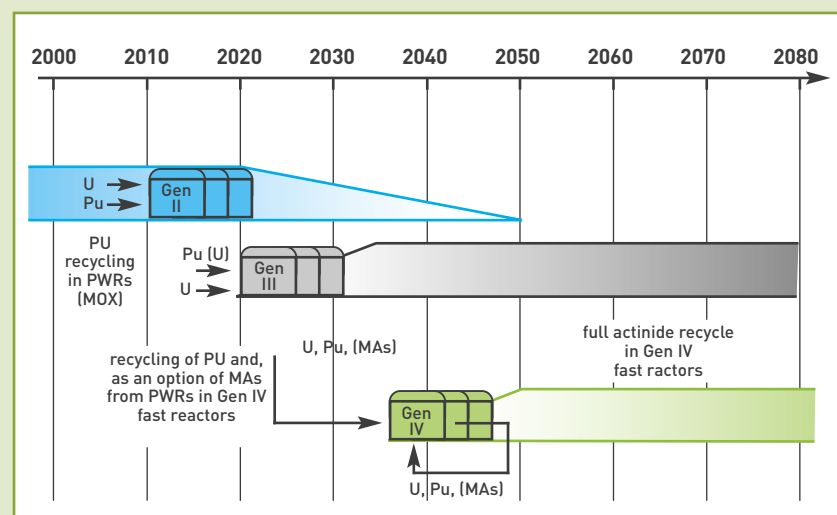


Figure. The manner in which the various nuclear reactor generations may successively take over the burden of recycling.



Gloveboxes used for the cladding of MOX fuel rods, in the Melox plant, at Bagnols-sur-Cèze (Gard département, southern France).

of just one fast reactor. This point need not be an impediment: even assuming a stagnation in the world nuclear plant fleet, the amount of plutonium building up in the form of spent fuel, by 2040, will stand at around 4,000 t, which is enough to ensure the startup of a fleet of fast reactors.

They will also have to procure large-capacity **reprocessing** plants, to reprocess that spent fuel, and turn it into a material suitable for use in fast reactors. For reference purposes, the French reprocessing plant at La Hague produces 11 t Pu per year, i.e. enough to start up just one fast reactor. This stands as a very real brake on the worldwide deployment of a fleet of reactors of this type. To get things going more quickly, it is feasible to start up part of the fast reactors by using enriched uranium, at the cost however of incurring economic losses, and a depletion of fissile resources.

No fast reactor without a cycle plant

In any event, there can be no fast reactor unless a cycle plant is available, for yet another, very good reason: once their deployment is ensured, fast reactors will in turn require reprocessing–recycling capacities for their own fuel. However, fast reactor fuel **burnup** is, as a rule, higher than is the case for water reactor fuel, so that the capacity required for the reprocessing of fuel from a fleet of water reactors will be adequate to meet the requirements of a fast reactor fleet of equivalent power, once it reaches stable operation (see Box).

Cycle plants will have to prove suitable for the reprocessing–recycling of fourth-generation system fuel. While other processes are being investigated, it is likely that the process selected will, as is the case at present, be a **hydrometallurgical** one. Indeed, the separative performance of this type of process is already outstanding, proving hard to match by other processes (see *Closing the cycle with fourth-generation reactors*, p. 118). Further, using new, recently discovered **complexing molecules**, prospects for advances remain high, both in terms of performance, and of flexibility with respect to actinide management. The modifications required for the industrial deployment of such improvements are not extensive.

Concerns relating to **proliferation** will probably loom large, with respect to the political decisions regarding deployment of nuclear energy. Reprocessing–recycling processes having the capability to ensure management

of actinides *as a group* would render any diversion or hijacking of fissile materials impractical. Such processes are currently being developed, with highly promising results.

It can hardly be envisaged that every country seeking to deploy a sustainable nuclear energy capacity procures its own reprocessing–recycling plant. A more realistic scenario, both in economic and political terms, would involve the setting up, in a limited number of countries where nuclear energy technology is already well developed, reprocessing plants providing a large capacity, with the ability to meet world demand. The **GNEP (Global Nuclear Energy Partnership)** initiative, recently launched by the United States, seeks to promote this option.

The strategy being contemplated in the United States, for the introduction of these nuclear systems of the future, emphasizes, as a priority, a clean, nonproliferating nuclear energy, and sees fast reactors as actinide burners. As this strategy entails a closed cycle, with respect to nuclear fuel, the United States are arguing the world should be divided into states having the capability to reprocess nuclear fuel, and states not having it. The latter would not be debarred from access to nuclear energy, however they would purchase, from the former, such services as uranium enrichment, or spent fuel reprocessing. Time will tell whether the “second-class” states will agree to such conditions...

A future for thorium, and molten salts?

The implicit assumption, in this paper, was that the nuclear systems of the future will be operating on the basis of a fertile uranium 238–fissile plutonium cycle. Another cycle may also be considered: that involving fertile **thorium 232**, and fissile uranium 233. This cycle can bank on two major advantages: the abundance of thorium (2–3 times greater than for uranium), and the low amounts of actinides yielded by the cycle. It would be peculiarly suitable for value-added utilization in **molten-salt reactors**, these forming one of the systems selected by the Gen IV Forum. However, the thorium cycle also carries handicaps: its chemistry is delicate, and further complicated by radioprotection issues; also, it has been little investigated. Its deployment may thus not be realistically contemplated before the end of this century.

The advent of **fusion** energy is probably further off still, and speculative, for both purely technical reasons, and on economic grounds.

The way ahead to the deployment of fourth-generation systems is thus well laid out, since the specifications for, and the nature of, the nuclear systems of the future are somewhat better delineated. However, there can be no question of a deployment of fast reactors around the world until one, indispensable prerequisite is met: the deployment of fuel reprocessing plants. Many questions remain: Who will the lucky winners be, at the outcome of the natural selection process between the fourth-generation systems currently battling it out? Will there indeed be a single winner? And when will such systems prove economically attractive?

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The components of a nuclear system

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

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

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

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

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

Control devices (on the one hand, **control rods**, or **pilot** and **shutdown rods**, made of neutron-absorbent materials [boron, cadmium...], and, on the other hand, **neutron “poisons”**) allow the neutron

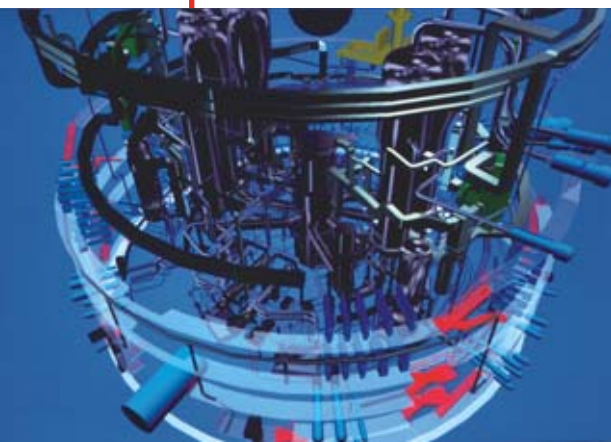
population to be regulated and, in the process, by acting on its **reactivity**, to hold reactor power at the desired level, or even to quench the chain reaction. The rods, held integral and moving as one unit (known as a **cluster**) are inserted more or less deeply into the core. Poisons, on the other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

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

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

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

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



Areva NP

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

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

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

The **moderator**, when required, plays an

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

Reactor lines, generations, and neutron spectra

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

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

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

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



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

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

transferred from the core to **heat exchangers** by means of water, kept at high pressure in the **primary circuit**.

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

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

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

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

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

The **second generation** covers those reactors, currently in service, that came on stream in the period from the 1970s to the 1990s. Solely

built for electricity generation purposes, most of these (87% of the world fleet) are water-cooled reactors, with the one outstanding exception of the British-built **AGRs** (advanced gas-cooled reactors). The standard fuel they use consists of **sintered enriched uranium-oxide pellets**, to about 4% uranium-235 enrichment, stacked in impervious tubes (**rods**), which, held together in bundles, form **assemblies**. PWRs hold the lion's share of the market, accounting for 3 nuclear reactors out of 5 worldwide. This line includes the successive "levels" of PWR reactor models built, in France, by Framatome (now trading as **Areva NP**) for national power utility EDF. Russian reactors from the **VVER** 1000 line are comparable to the PWRs in the West. While operated in smaller numbers than PWRs, **BWRs** (boiling-water reactors) are to be found, in particular, in the United States, Japan, or Germany. Finally, natural-uranium powered reactors of the **CANDU** type, a Canadian design, and their Indian counterparts, form a line that is actively pursued. These are also pressurized-water reactors, however they use **heavy water** (D₂O) for their moderator, and coolant, hence the term **PHWR** (pressurized-heavy-water reactor) used to refer to this line.

The **third generation** corresponds to installations that are beginning to enter construction, scheduled to go on stream from around 2010. This covers, in particular, the French-German **EPR**, designed by Areva NP (initially: Framatome and Siemens), which company is also putting forward a boiling-water reactor, the SWR-1000, at the same

time as it has been coming together with Japanese firm Mitsubishi Heavy Industries. This generation further includes the AP1000 and AP600 types from Westinghouse, a firm now controlled by Toshiba; the ESBWR and ABWR II from General Electric, now in association with Hitachi; the Canadian ACRs, and the AES92 from Russia; along with projects for smaller integral reactors.

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

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

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

Thermodynamic cycles and energy conversion

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

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

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

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

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

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

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

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

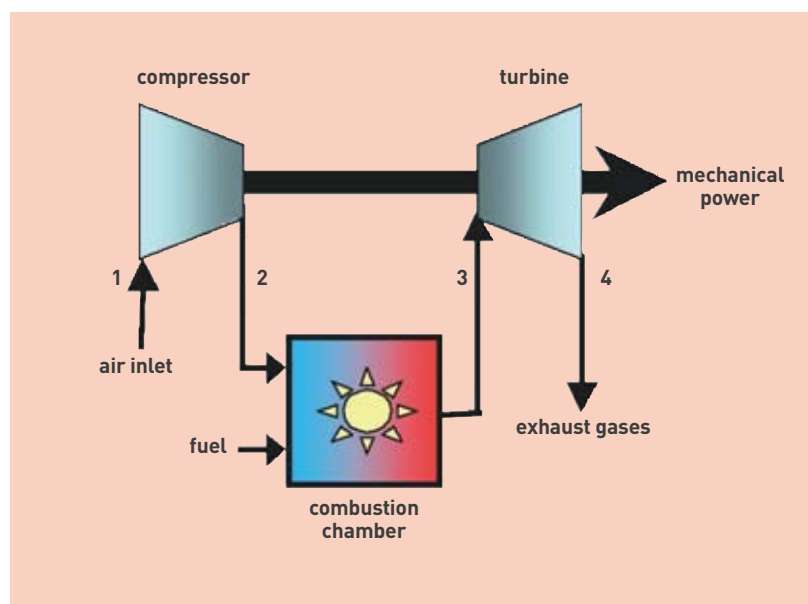


Figure.
Brayton cycle, as implemented in an open-cycle gas turbine.

What is multiphysics, multiscale modeling?

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

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

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

atom), by way of nested levels of description (molecular dynamics, dislocation dynamics).

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

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

Finally, multiscale modeling joins up, through superposition techniques, numerical models at different scales. This makes it possible – to stay with the example of materials – to “zoom in” on

regions that are particularly sensitive to stresses, such as fissures, welds, or supporting structures.

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

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

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

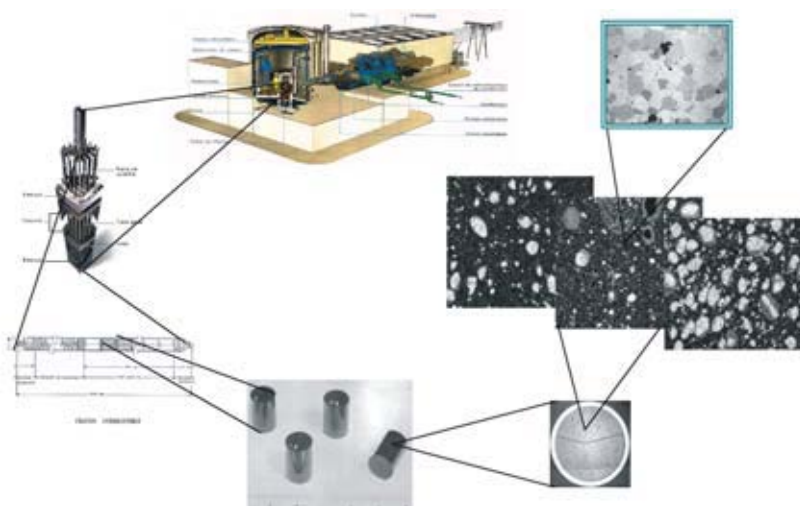


Figure.

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

The main families of nuclear materials

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

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

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

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

Neutron **irradiation** can cause a major alteration in the properties exhibited by the materials employed in the various components of a reactor. In metals, and metal alloys, but equally in other solid materials, such as ceramics,⁽¹⁾ such alterations are related to the evolution of the **point defects** generated by this irradiation, and to the

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

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

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

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

Crystal structures

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

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

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

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

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

Steel families

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

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

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

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



Areva NP

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

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

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

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

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

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

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

to that of austenitic steels. They currently provide the reference solution for fuel cladding, for future sodium-cooled reactors.

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

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

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

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

In particle fuels, uranium and plutonium oxides are coated with several layers of insulating **pyrocarbons**, and/or silicon **carbide** (SiC), possibly in fibrous form (SiCf). These are known as coated particles (CPs). While SiC-coated UO₂, or **MOX** balls stand as the reference, ZrC coatings might afford an alternative.

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

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

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

The six concepts selected by the Gen IV Forum

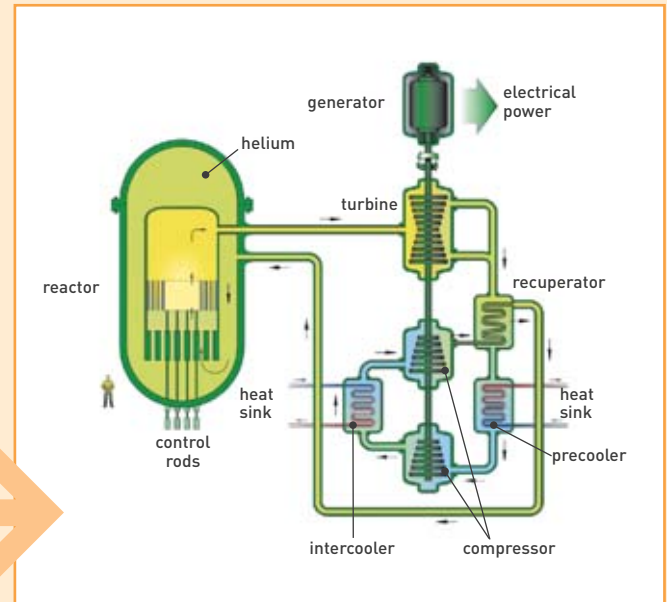
Of the six concepts selected by the **Generation IV International Forum** for their ability to meet the criteria outlined, three – and ultimately four – make use of **fast neutrons**, while three (ultimately two) use **thermal neutrons**. At the same time, two of the six concepts use gas as a coolant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

GFR

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

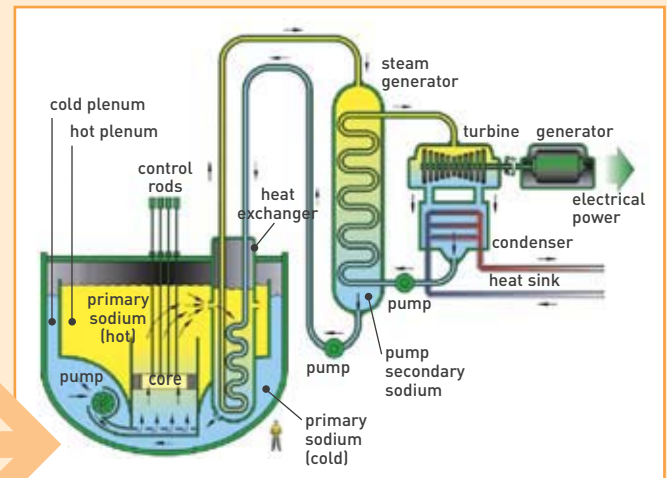
(1) GT-MHR: Gas-Turbine Modular Helium Reactor.

(2) PBMR: Pebble-Bed Modular Reactor.



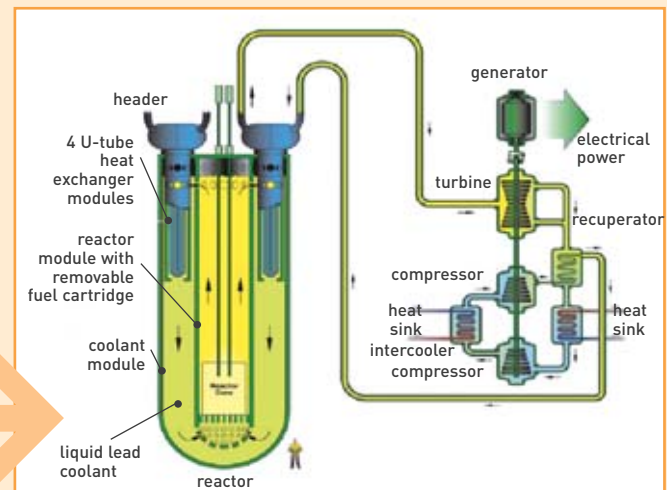
Le SFR

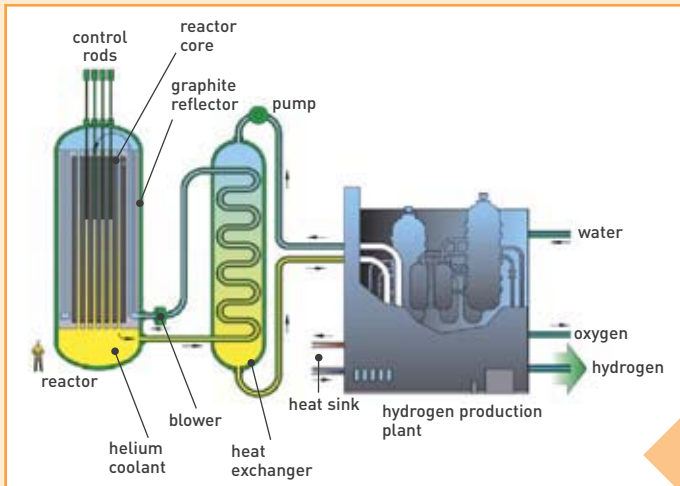
The sodium-cooled fast reactor system (**SFR**) is a liquid-**sodium**-cooled, fast-neutron reactor, associated to a **closed cycle**, allowing full actinide recycle, and **plutonium** breeding. Owing to its breeding of **fissile** material, this type of reactor may operate for highly extended periods without requiring any intervention on the **core**. Two main options may be considered: one that, associated to the **reprocessing** of metallic fuel, results in a reactor of intermediate unit power, in the 150–500 MWe range; the other, characterized by the **Purex** reprocessing of mixed-oxide fuel (**MOX**), corresponds to a high-unit-power reactor, in the 500–1,500 MWe range. The SFR presents highly advantageous natural resource utilization and actinide management features. It has been assessed as exhibiting good safety characteristics. A number of SFR prototypes are to be found around the world, including Joyo and Monju in Japan, BN600 in Russia, and Phénix in France. The main issues for research concern the full recycling of actinides (actinide-bearing fuels are **radio-active**, and thus pose fabrication difficulties), in-service inspection (sodium not being transparent), safety (**passive** safety approaches are under investigation), and capital cost reduction. Substitution of water with **supercritical CO₂** as the working fluid for the power conversion system is also being investigated.



LFR

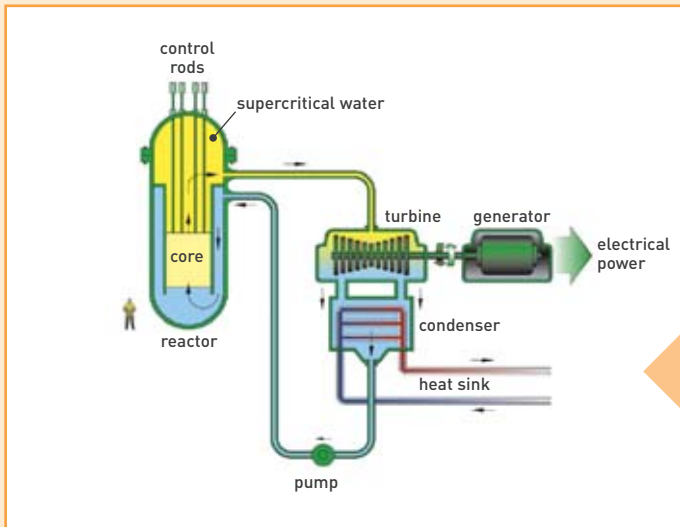
The lead-cooled fast reactor system (**LFR**) is a lead- (or lead-bismuth alloy-) cooled, fast-neutron reactor, associated to a closed fuel cycle, allowing optimum **uranium** utilization. A number of reference systems have been selected. Unit power ranges from the 50–100 MWe bracket, for so-called battery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





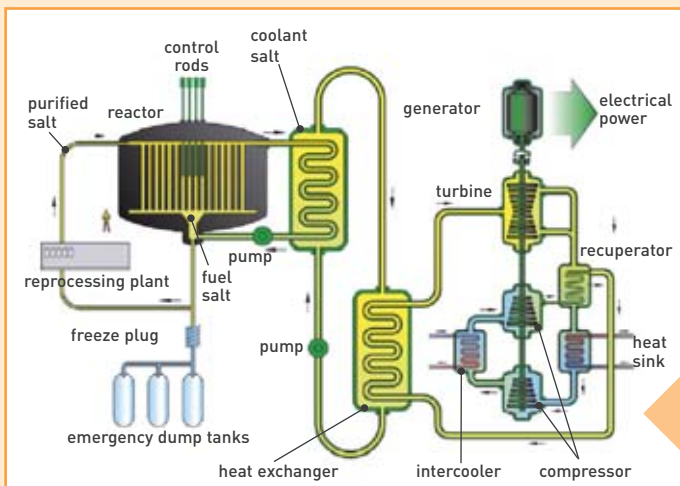
VHTR

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



SCWR

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



MSR

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