



The JHR reactor: a multipurpose asset for materials

The creeping obsolescence of experimental irradiation reactors around the world, in the European Union in particular, means that the Jules Horowitz Reactor (JHR), due to be commissioned by CEA in 2014, at Cadarache, is an essential addition, whether it be for the investigation of materials, or of novel fuels. Its capabilities extend beyond the requirements of investigations relating to fourth-generation systems.



The JHR, seen to the left, in the background, as it will appear when added to the existing facilities at CEA's Cadarache Center (Bouches-du-Rhône département, southeastern France)

Fourth-generation concepts call for major innovations, in terms of materials, and **fuels**. Such innovations go through the stages of selection, qualification, optimization, and safety tests, for which two types of reactor play a crucial role.

In general terms, development of innovative reactors involves a number of stages: an experimental reactor (e.g. Rapsodie, or Joyo, for the **SFR** reactor line; HTTR for the **HTR** reactor line), allowing the validation of general concepts, followed by an electricity-generating prototype (Phénix, Superphénix,

(1) Named after the internationally renowned physicist, and former Director, Nuclear Reactors and Fundamental Research at CEA (he died in August 1995). He had specialized, as early as 1946, in design studies for research reactors, becoming one of the fathers, in particular, of the **natural uranium-graphite-gas (UNGG)** reactor line.

Monju, for the sodium-cooled technology line), before the reactor line is deployed. The technological maturity of the sodium-cooled line makes it possible to pass directly to the prototype stage. It is of course possible, in experimental and prototype reactors, to carry out any number of tests on materials, and fuels, in normal operating conditions (temperature, environment, flux, stresses...).

However, there is also the need for the ability to conduct more elaborate experiments, in a research reactor, by nature more flexible, catering for sophisticated instrumentation, and allowing tunable control of environment conditions and allowing operating limits to be explored. The construction, currently going ahead, of a new, high-performance research reactor, the Jules Horowitz⁽¹⁾ Reactor (JHR), due to be commissioned in 2014, meets this very goal.

JHR: a new test reactor in the European Union

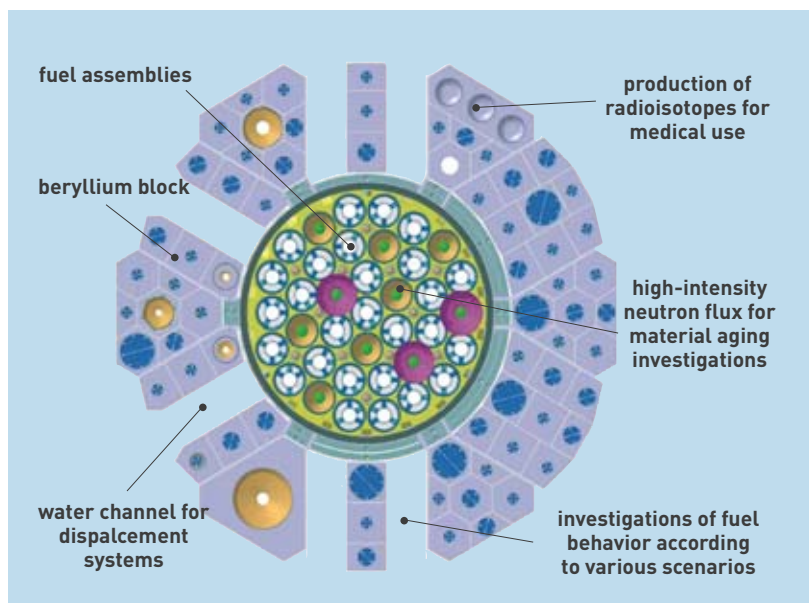
European **irradiation** reactors have provided crucial support for nuclear energy programs, over the past 40 years. In association with **hot laboratories**, for post-irradiation investigations, they form an infrastructure structuring the European Research Area, in the field of **fission**. However, irradiation reactors currently in service within the European Union will be over 50 years old, by the next decade, and are due to be gradually phased out, owing to their obsolescence. Such a situation is incompatible with the sustainable development of nuclear energy, which provides a major component in the energy mix. As a result, CEA launched a project for a new materials and fuel test reactor: JHR.

JHR design definition studies, funded by CEA and **EDF**, were carried out from 2003 to 2005, by an engineering team some 100 strong, pooling the expertise of **Areva**, and EDF. The construction phase is being conducted under the aegis of an international consortium, bringing together, at the present time, alongside Areva, CEA, and EDF, the European Joint Research Center (JRC, coming under the **European Commission**), CIEMAT (Spain), NRI (Czech Republic), SCK (Belgium), and VTT (Finland). These players are funding construction on a contractual basis, relating rights of access in JHR, to capital outlay for the project. Discussions are ongoing to open up the consortium to further partners.

JHR will provide, over a major part of the 21st century, a high-performance irradiation capability, for the investigation of material and fuel behavior under irradiation, involving a high **neutron** flux (typically double the flux available in current European test reactors). It will allow simulating a variety of environments (in terms of pressure, temperature, flux, **coolant** chemistry), reproducing the conditions prevailing inside power reactors (water reactors, gas-cooled reactors, sodium-cooled fast reactors...). It will stand out in terms of its expanded online instrumentation, consistent with the growing importance of **modeling**, and **numerical simulation**, and its safety specifications are fully compliant with current design guidelines.



Artist's impression of JHR and ancillary buildings.



A 100-MWth research reactor

JHR is a 100-MWth, pool-type reactor. JHR's **core** is held in a pressurized vessel (740 mm in diameter), within a **primary circuit** ensuring water circulation through forced **convection**.

The reactor building will accommodate, on the one hand, the primary circuit, and, on the other, the experiment area (300 m² per level, over 3 levels), communicating with the core, catering for the running of some 10 experimental **loops**. Next to the bunkers housing these loops, a **fission product** analysis laboratory will allow the online characterization of the content of the fluids yielded by fuel samples.

Communicating with the reactor building, the nuclear ancillary buildings will house the pools, laboratories, and **hot cells** indispensable for reactor operation, and experimental processes. The latter will require, aside from a cell for feed fuel management, two hot cells, for the preparation, and treatment of experimental devices, before and after irradiation. In addition, a so-called **alpha** cell will serve as a dedicated cell for experiments investigating fuel operating limits (experiments to explore accident conditions, or to optimize safety margins, that may result in containment failures within the sample). The capability for investigation of degraded fuels, possibly right up to melting, whether involving interaction with the coolant or otherwise, is a significant improvement, meeting a rising demand from end-users.

The core, 600 mm high (see Figure 1), is water-cooled, and water-moderated. The fuel element consists of concentric circular plates, and its central cavity can accommodate an experiment, or house a **control rod**. The fuel elements are held in an aluminum rack. Some of these elements may be removed, being replaced by experiments. The core is surrounded by a reflector, optimizing cycle time, and forms a high **thermal-neutron flux** region. The reflector is mainly made from beryllium⁽²⁾ blocks, cut away to let water channels through. The experiments are positioned

Figure 1. A typical configuration for the JHR core, involving a load of some 20 experiments, while allowing concurrent production of radioisotopes, for medical use.

(2) Beryllium: a light metal (⁷Be), used, among other purposes, for its neutron **moderation** and reflection properties.



in beryllium blocks, or on displacement systems in the water channels passing through the reflector.

This design meets a twofold objective. It allows a high **fast-neutron** flux to be achieved inside the core, hence the high **power density** within the core (600 kW/l); and the positioning, in the reflector, of displacement systems subjected to a high thermal-neutron flux.

Core performance is determined to cater for a load schedule of some 20 experiments (so-called perturbed flux conditions).

At least two decades ahead

JHR will stand as the highest-performance materials test reactor for at least the next two decades. Indeed, a number of existing test reactors, in the European Union or in Japan, are aging, exhibiting limited performance, and are due to be decommissioned in the near future. Other test reactors, in Asia and in Australia, in particular, have been designed as multipurpose installations, with consequently poor performances for nuclear energy users. Finally, the ATR reactor, in the United States, yields a neutron flux as high as that planned for JHR, with lower operating flexibility.

For innovative materials and fuels

Development of Generation-IV reactors entails use of innovative fuels, and materials, able to operate in more stressful temperature, flux, and environmental conditions, over a timespan of 60 years.

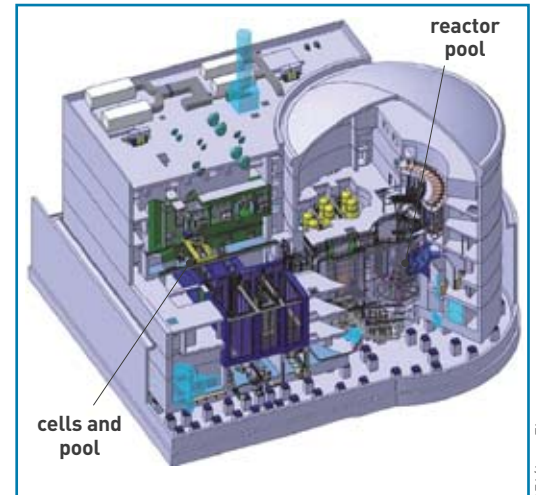
Such materials, whether already existing, or calling for optimization, or development, require extensive characterization, in the operating conditions they are intended for.

The reactor concepts currently being developed require evaluation of the behavior under irradiation of a wide variety of materials, such as **graphite** (VHTR, and **MSR**), **austenitic and ferritic steels** (VHTR, SFR, **GFR**, **LFR**), **nickel alloys** (SCWR), **ceramics** (GFR) (see the Table, p. 75, setting out the materials being considered for in-core operation, for the six systems being investigated under the aegis of the **Generation IV International Forum**; and Focus E, *The main families of nuclear materials*, p. 76).

Selection, optimization, and qualification of such materials, and fuels call for use of several, complementary types of irradiation. Initial selection, for instance, may rely on irradiations involving little instrumentation, in conditions however that are most representative of operating conditions. Ideally, for

a reactor line for which experimental reactors are already available (this being the case for SFRs, and VHTRs), such irradiations may be carried out in these reactors.

Optimization of the materials selected provides a second example. This requires a good understanding of the



Cutaway view of the JHR reactor building, showing, in particular, the antiseismic support structures (bottom), and the pool (right)

processes involved in aging under irradiation, microstructural and dimensional evolutions, behavior under mechanical stress, flux-induced deformations, and, for that purpose, *analytical* and *separate-effect* irradiations are required. In such experiments, parameters such as temperature, stress, deformation, and environmental chemistry are known, and measured, to a high precision.

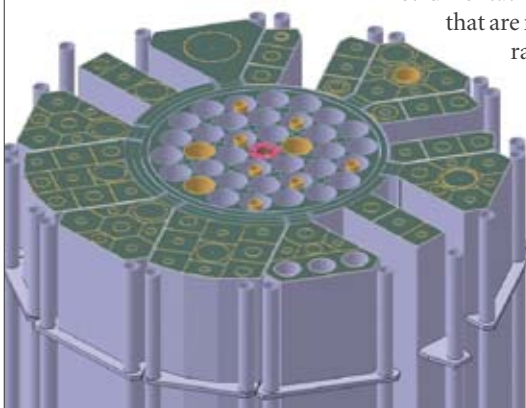
This relies on simulation experiments (irradiation in the JANNUS triple accelerator, for instance: see *The JANNUS platform: ion-beam simulation of materials irradiation*, p. 110), but equally, and most crucially, on irradiations in research reactors providing sophisticated experimental setups, and an appropriate neutron flux.

Finally, qualification experiments will be carried out in experimental, or prototype reactors, reproducing in every aspect normal operating conditions; however, they require the use of research reactors, if they are to have the ability to test materials to their limits, along with their behavior in incident or accident situations. The few examples considered in the following paragraphs will serve to illustrate the experimental requirements that are to be met by existing research reactors, and subsequently by JHR.

Testing innovative metal alloys...

The improvements that are required as regards fuel **cladding**, for SFRs, involve development of a novel oxide-dispersion-strengthened (**ODS**) ferritic steel grade, the aim being to achieve higher **burnup**, this being restricted, to date, by cladding resistance under irradiation. Indeed, at 100 **dpa**, the 15–15Ti austenitic grade exhibits **swelling** such that it proves impossible to proceed with further irradiation, for intrinsic reasons (embrittlement threshold), and on design grounds (volume). Target dose for cladding is set at 200 dpa, corresponding to a doubling of burnup value, to 200 GW · d/t. Austenitic steels of the 15Cr/20–25Ni type may allow doses of 150 dpa to be achieved, however, in order to achieve 200 dpa, it will be necessary to employ **ferritic-martensitic steels**. A large amount of data, yielded by irradiations in a number of fast reactors (see Figure 1, p. 71), indicate the good swelling resistance demonstrated

The JHR core, surrounded by beryllium blocks



by 9–12% Cr steels, owing to the body-centered cubic crystal structure, and high density of irradiation defect elimination sinks provided by the martensitic structure. Such alloys are restricted to temperatures lying in the 400–550 °C range. To enhance their high-temperature **creep** properties, these alloys must be strengthened by means of oxide dispersion. It is essential, however, to make sure the distribution of oxide particles is not affected by irradiation, and to understand the way such oxides act on irradiation creep, for which **dislocation** climb processes are dominant. Other types of hardening particles (**carbides**, **nitrides**) are also being investigated. Candidate grades will further have to be characterized, in order to overcome major barriers relating to other critical points for this type of material, such as workability, or weldability.

Such increased burnup will make for improved resource management, and will result in major savings in terms of waste. Further, use of low-swelling alloys may allow the interval between fuel **pins** to be brought down, this being one potential way of improving safety, by reducing the **void effect**.

... and ceramic-matrix composites (CMCs)

In the case of gas-cooled fast reactors, operating as they do at even higher temperatures, employment of **ceramic** materials is required. One candidate affording some benefits is silicon carbide (SiC). However, in order to enhance its mechanical characteristics, it must imperatively be used in **composite** form. This makes it possible to contemplate applications up to 1,200 °C, in operating conditions, up to 1,600 °C

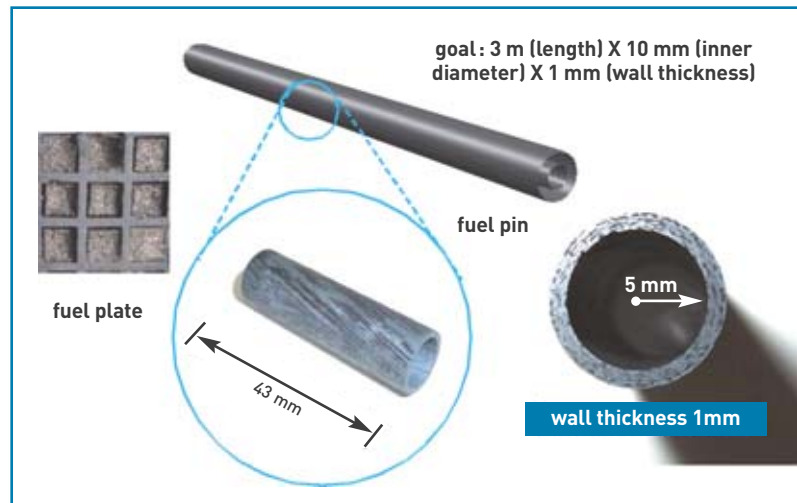


Figure 2. SiC-SiC cladding for GFR fuels (plate and pin concepts).

flux; or quantification of the damage incurred under irradiation and under load.

For the purposes of understanding, and predicting the behavior under neutron flux, and under load of such materials, in order to optimize them, it is crucial the behavior be determined, of each component in this structure. For this purpose, an irradiation creep experiment, involving variable stresses, and *in situ* measurements, on SiC microcomposites is being developed in the OSIRIS materials testing reactor, at CEA's Saclay Center.

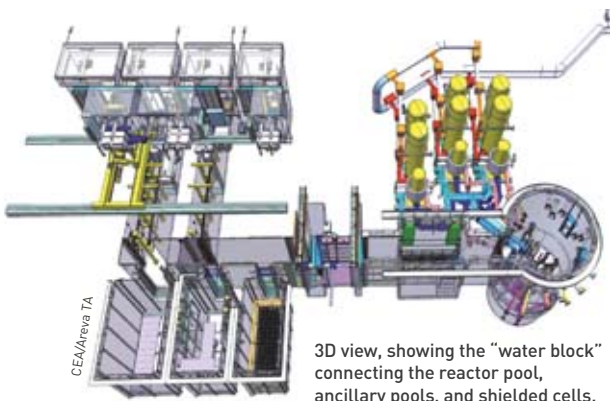
Finally, in order to meet the Gen IV sustainability targets, it becomes necessary to **close the cycle**, and incinerate **actinides**. A fuel exhibiting higher density may also be favored. The priority focus is thus on oxides, and nitrides, even though carbides, and metallic fuels are also being investigated. Evaluation of the behavior of such fuels in normal, incident, and accident conditions will be required.

Controlling **fission gas** release, for instance, is a major challenge, since these gases impact reactor economics (they may restrict achievable burnup), safety (fuel-coolant interaction, and **fission product** retention, in the event of a severe accident), along with its **minor actinide** management capability. JHR has been optimized to cater for this issue, by online measurement of fission products and helium, and will feature a **high-activity** cell, for the purposes of damaged fuel management.

A lynchpin for nuclear energy

Innovation in the area of materials, and fuels is a lynchpin for nuclear energy development. This truth, manifest as it is in practical, economic, and safety terms, implies that scientific, and technical mastery be achieved, with respect to material behavior under irradiation. This entails a renewed drive in terms of experiment, as of modeling. Completion of a new research infrastructure such as JHR, affording as it does high-performance experimental capabilities, alongside existing, and future experimental reactors, will be a contribution to this essential effort.

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3D view, showing the "water block" connecting the reactor pool, ancillary pools, and shielded cells.

in accident conditions. The fiber architecture is, as a rule, component-specific, for the purposes of supporting the thermal-mechanical load. Examples of cladding, used for the plate concept, and the pin concept, are shown in Figure 2. **Toughness** values of around 25 MPa · m^{1/2} are obtained, through choice of an appropriate interphase, between matrix and fibers. Satisfactory traction resistance, and dimensional stability have been obtained to date, only up to a dose of 10 dpa. There is thus still a long way to go before these innovative materials are qualified with respect to dimensional stability, and long-term physical properties; and as regards the effects of irradiation on the interphase, and its ability to enhance toughness, by deflecting cracks; and fiber creep resistance at high temperature, under high fast-neutron

The components of a nuclear system

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

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

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

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

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

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

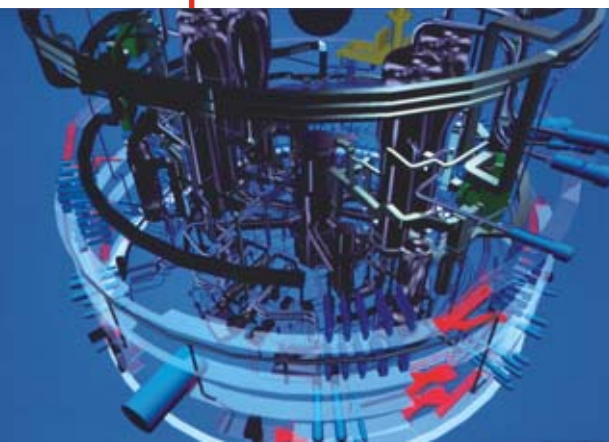
population to be regulated and, in the process, by acting on its **reactivity**, to hold reactor power at the desired level, or even to quench the chain reaction. The rods, held integral and moving as one unit (known as a **cluster**) are inserted more or less deeply into the core. Poisons, on the other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

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

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

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

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



Areva NP

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

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

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

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

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

Reactor lines, generations, and neutron spectra

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

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

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

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



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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

Thermodynamic cycles and energy conversion

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

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

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

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

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

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

The thermodynamic properties of a coolant gas such as helium make it possible to go further, by allowing a target core outlet temperature of at least 850 °C. To take full advantage of this, it is preferable, in theory, to use a **direct** energy conversion cycle, the **Joule-Brayton cycle**, whereby the fluid exiting the reactor (or any other "boiler") is channeled directly to the turbine driving the alternator, as is the case in natural-gas, **combined-cycle** electricity generation plants, or indeed in a jet aero-engine. Using this cycle, electricity generation efficiency may be raised from 51.6% to 56%, by increasing T_c from 850 °C to 1,000 °C. Indeed, over the past half-century, use of natural gas as a fuel has resulted in a spectacular development of gas turbines (GTs) that can operate at very high temperatures, higher than around 1,000 °C. This type of energy conversion arrangement stands, for the nuclear reactors of the future, as an attractive alternative to steam turbines. GT thermodynamic cycles are in very widespread use, whether for propulsion systems, or large fossil-fuel electricity generation plants. Such cycles, known as **Brayton cycles** (see Figure) simply consist of: drawing in air, and compressing it to inject it into the combustion chamber (1 → 2); burning the air-fuel mix inside the combustion chamber (2 → 3); and allowing the hot gases to expand inside a turbine (3 → 4). On exiting the turbine, the exhaust gases are discharged into the atmosphere (this forming the cold source): the cycle is thus termed an *open* cycle. If the hot source is a nuclear reactor, open-cycle operation, using air, becomes highly problematical (if only because of the requisite compliance with the principle of three confinement barriers between nuclear fuel and the ambient environment). In order to *close* the cycle, all that is required is to insert a heat exchanger at the turbine outlet, to cool the gas (by way of a heat exchanger connected to the cold source), before it is reinjected into the compressor. The nature of the gas then ceases to be dictated by a combustion process.

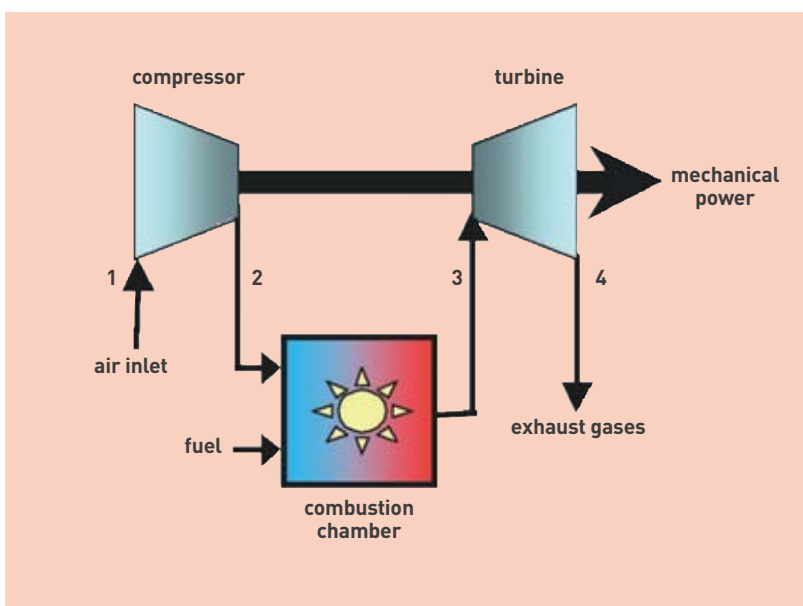


Figure. Brayton cycle, as implemented in an open-cycle gas turbine.

What is multiphysics, multiscale modeling?

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

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

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

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

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

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

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

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

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

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

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

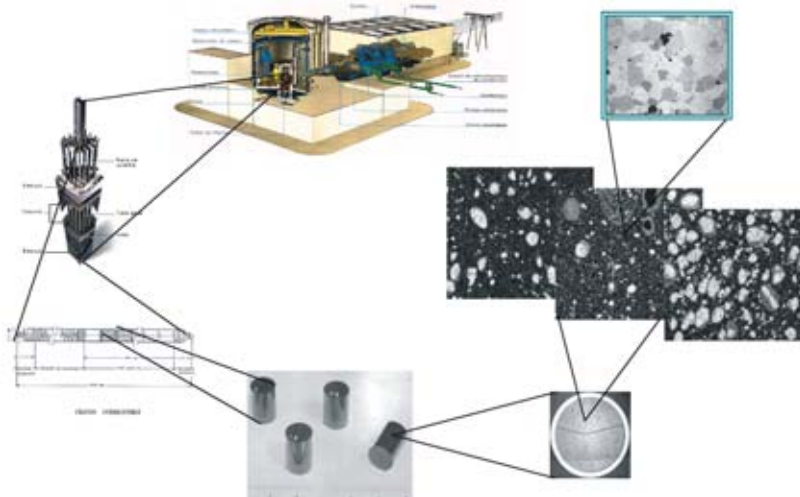


Figure.

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

The main families of nuclear materials

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

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

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

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

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

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

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

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



Areva NP

Pressure-vessel nozzle shell for EDF's Flammanville 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_E, 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.

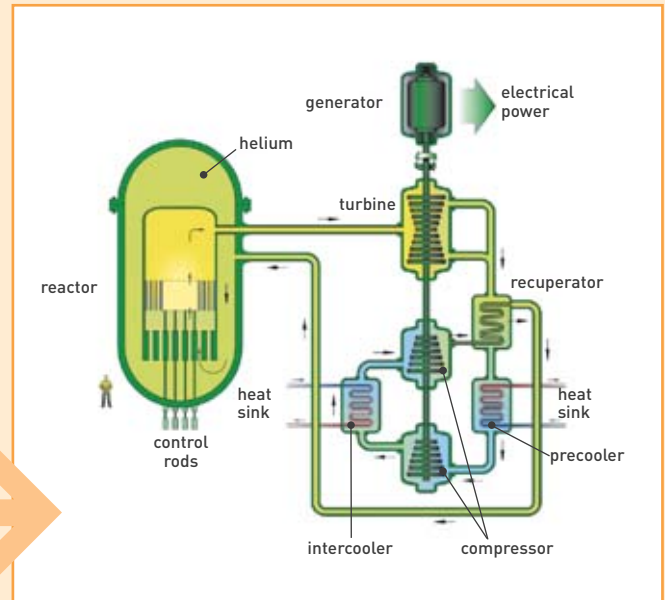
The six concepts selected by the Gen IV Forum

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

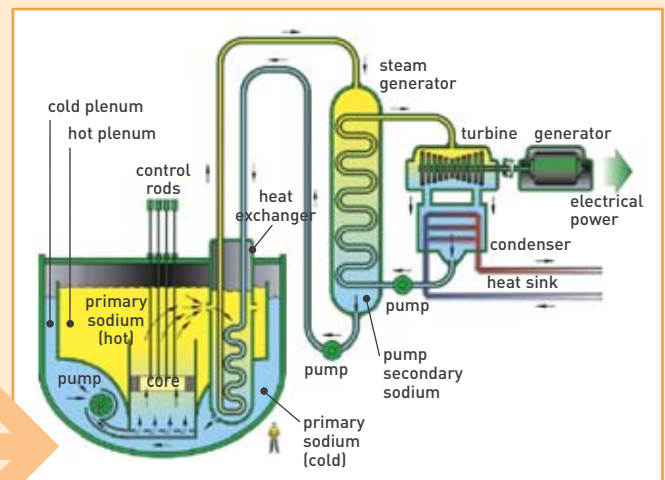
The gas-cooled fast reactor system (**GFR**) is a high-temperature, gas-cooled (helium-cooled as a rule), fast-neutron reactor allowing **actinide recycle** (**homogeneous**, or **heterogeneous**), while sustaining a **breeding** capability greater than unity. The reference concept is a **helium-cooled, 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.



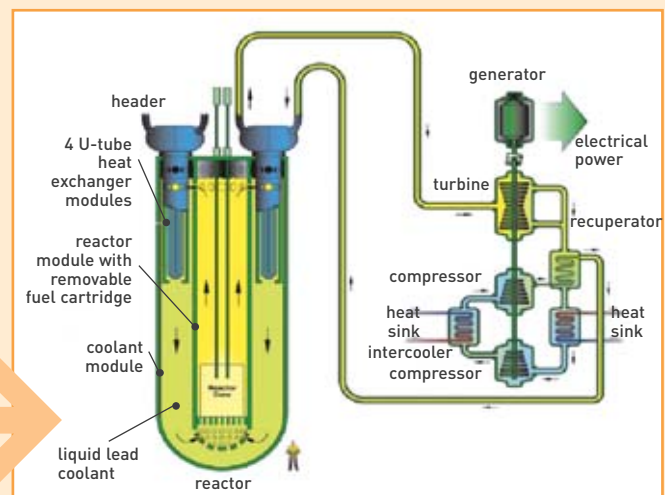
Le SFR

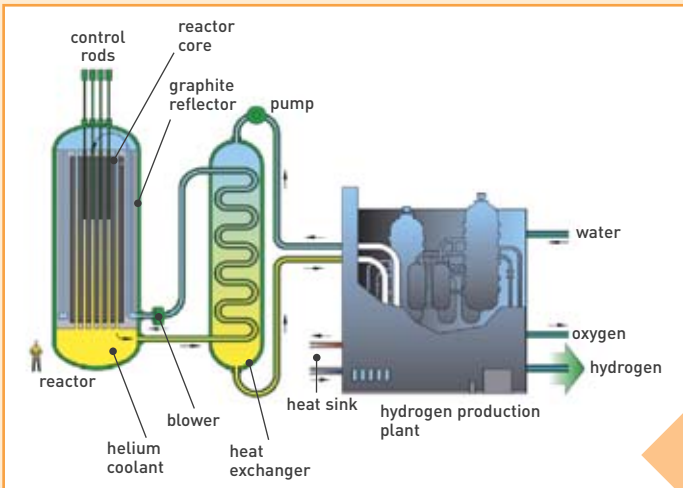
The sodium-cooled fast reactor system (**SFR**) is a liquid-**sodium**-cooled, fast-neutron reactor, associated to a **closed cycle**, allowing full actinide recycle, and **plutonium** breeding. Owing to its breeding of **fissile** material, this type of reactor may operate for highly extended periods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



LFR

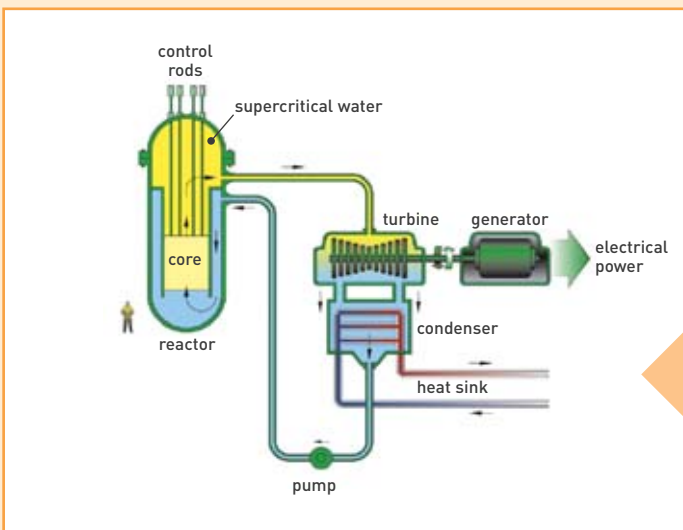
The lead-cooled fast reactor system (**LFR**) is a lead- (or lead-bismuth alloy-) cooled, fast-neutron reactor, associated to a closed fuel cycle, allowing optimum **uranium** utilization. A number of reference systems have been selected. Unit power ranges from the 50–100 MWe bracket, for so-called battery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





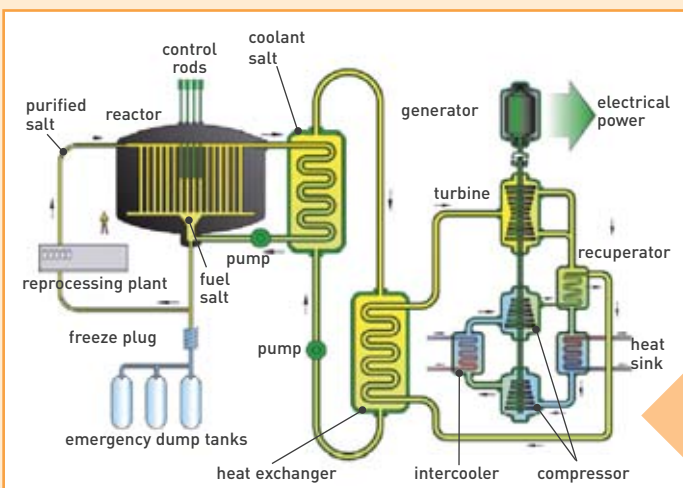
VHTR

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.