Sodium-cooled reactors: towards new advances in **monitoring** and **inspection** techniques

The characteristics exhibited by sodium have led the designers of sodium-cooled reactors to devise specific monitoring and inspection techniques. Reactors of the new generation will call for novel developments.

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The MIR inspection robot, used to test main vessel welds in the Superphénix reactor, seen here in the space extending between that vessel and the safety vessel.

Monitoring nuclear power plants during operations – this being a stringent requirement from operators and safety authorities – covers both the monitoring, on a continuous basis, of the reactor while it is functioning, and thoroughgoing in-service inspection, during scheduled shutdown periods.
Now, the sodium-**cooled fast reactor** (**FR**) concept (**SFR**), as selected by the **Generation IV International Forum**, owing the presence of hot sodium (180 °C during shutdown), opaque as it is, and not readily "drainable," makes for a highly specific environment, making such operations difficult. Consequently, CEA embarked, as early as the 1990s, in collaboration with its partners **EDF** and **Areva**, on a specific R&D effort on this issue, to go beyond what could be drawn from the experience gained with reactors of the previous generations.

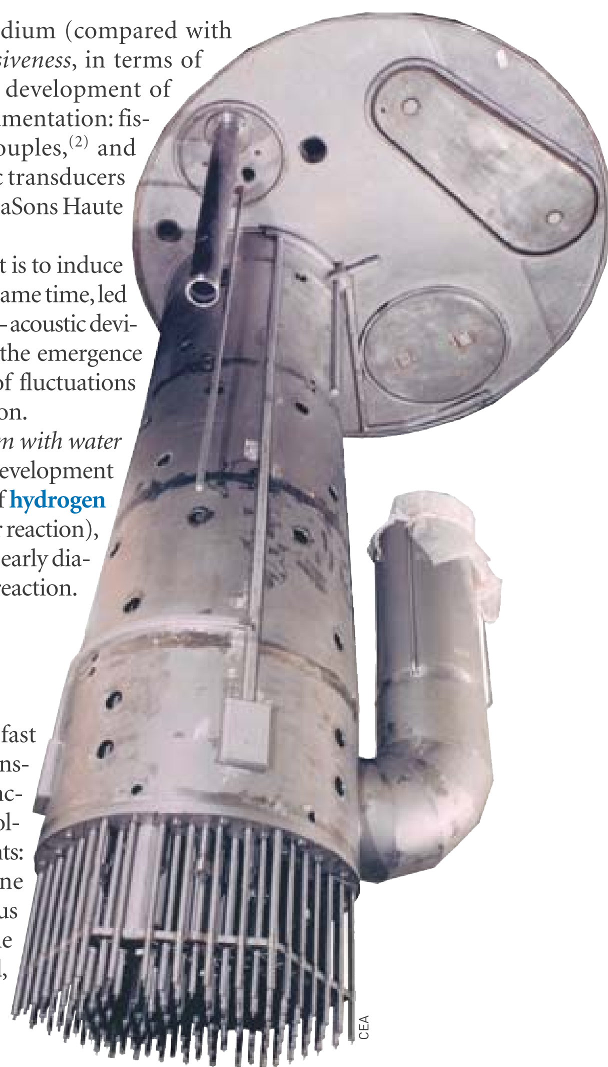
Specific sodium characteristics

Sodium exhibits specific characteristics, in several respects. Its *opacity*, first of all, imposes the substitution, for light, of **ultrasound** waves, electromagnetic waves, or nuclear radiation, if it is to be probed. Its *incompatibility, in hot conditions, with oxide* **fuel,** secondly – by contrast with the **light water** used in **PWR** reactors – which may cause a mere crack in a **pin cladding** to evolve into outright failure, liable to result in a transfer of **fissile** material to the **primary circuit** sodium. The delayed **neutrons** from fission, appearing as they do within a timescale that is compatible with the time constants relating to sodium circulation, do however allow cladding failure to be detected from outside the core.

The *high temperature* of sodium (compared with light water), and its *aggressiveness*, in terms of corrosion, thirdly, entailed development of specific measurement instrumentation: fission chambers,[1] thermocouples,[2] and high-temperature ultrasonic transducers (TUSHTs, Traducteurs à UltraSons Haute Température).
Its **void coefficient**, liable as it is to induce *positive* **reactivity,** has, at the same time, led to the development of devices – acoustic devices, in particular – to detect the emergence of gas bubbles, making use of fluctuations in ultrasound wave attenuation.
Finally, the *reactivity of sodium with water* (or air) has resulted in the development of systems for the detection of **hydrogen** (as a product of sodium-water reaction), down to trace levels, allowing early diagnosis of the sodium–water reaction.

The requirements of operational monitoring

Operational monitoring of fast reactors has the purpose of constantly checking that their functioning remains safe. This involves two inseparable components: *online monitoring*, on the one hand, which allows continuous monitoring of the state of the reactor, and, on the other hand, *in-service inspection,* covering the entire ensemble of periodical tests for major structu-



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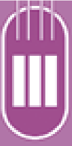
Phénix core cover plug, showing, in the lower region, the forest of sodium sampling tubes for the LRG system, and, at the center of each tube, the thermocouple to measure assembly outlet temperature.

(1) Fission chamber, ionization chamber, ionization fission chamber: a radiation detector, based on the ionization of a gas by particles passing through an enclosure inside which a voltage is applied between two **electrodes**, these collecting the **ions** and **electrons** released. Passage of an ionizing particle triggers an *electron avalanche* inside a counting device.

(2) Thermocouple: a device measuring temperature (more accurately, a temperature differential), based on the *Seebeck effect*, which causes a voltage to arise between two junctions, connecting metallic wires of different kinds, one of which is kept at a known temperature.



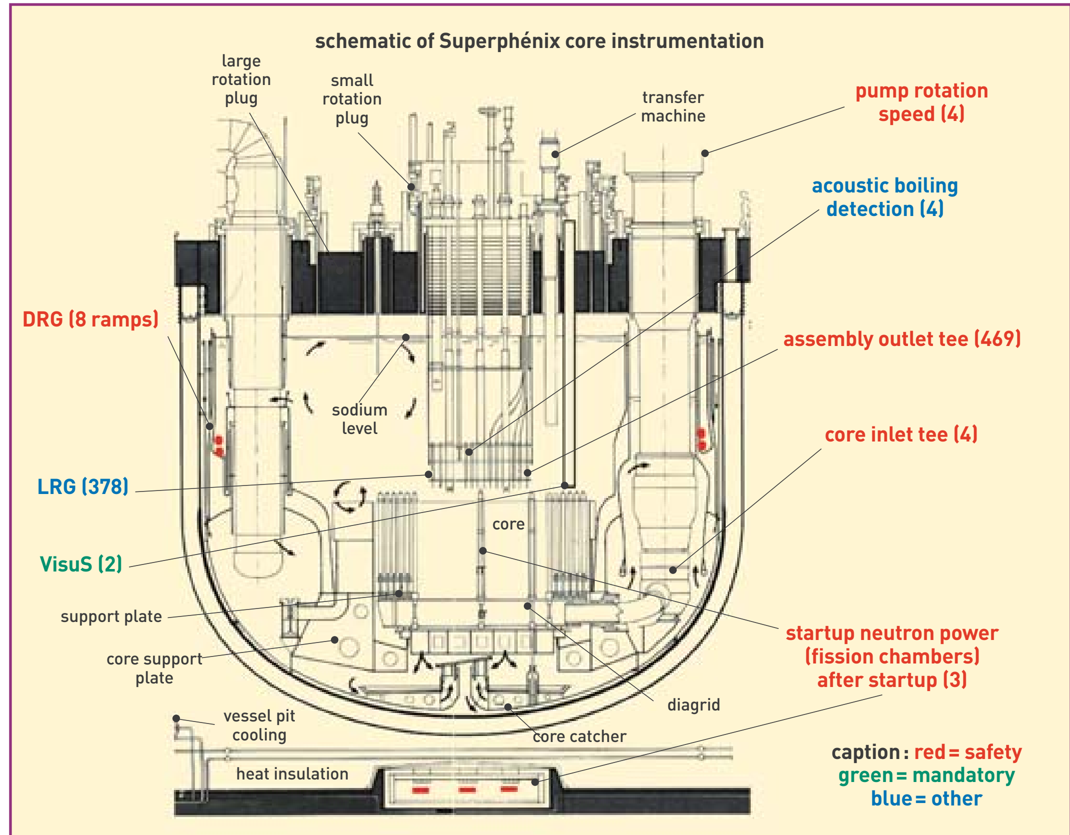


Figure 1. Instrumentation for the core of a sodium-cooled fast reactor (taking the instance of Superphénix).

ral components (chiefly: the reactor block, secondary sodium circuits, and heat exchangers).

Core monitoring

Core monitoring has the chief purpose of ensuring maximum allowable temperatures are complied with, as regards pin claddings, and fuel pellets. In most sodium-cooled reactors, this function is ensured, on the one hand, through global measurements (core power, pump flow rates, or operating speeds, sodium heating as it passes through the core), and, on the other hand, through individual temperature measurements, taken at **assembly** outlets. The latter have the purpose of safeguarding the reactor against local incidents, such as decreased flow rate in an assembly, due to more or less complete blockage; or an unexpected **control rod** extraction.

The measuring devices employed are thermocouples (for the swift processing of core temperatures), electromagnetic flowmeters (making use of Faraday's law,(3) since sodium is an electrical conductor), and neutron chambers, to monitor power. With respect to the latter, temperature and neutron flux sensitivity constraints have led, in the past, to using several types of chamber – ionization, or fission chambers – to cover the full operating dynamics (to measure power levels ranging from 1 **watt** to several thousand megawatts, i.e. over a range of some 10 decades). Such chambers are positioned outside the region where high temperatures prevail inside the core, and specific construction measures are implemented (e.g. "neutron guides," in Superphénix), to ensure an adequate neutron flux.

(3) Faraday's law: a changing magnetic field has the ability to induce an electric field, causing an electric current to flow in a circuit, voltage being a function of the rate of change of the magnetic field.

For the future, the planned use of high-dynamic-range fission chambers, with the ability to withstand high temperatures, is intended to restrict the numbers of such chambers required, while allowing their location inside the reactor vessel.

As regards temperature measurements, there is a need to cater for thermocouple replacement, in the event of failure, or even of sensor drift. The hydraulics above the core should also be taken on board, to guarantee the representative character of the measurements taken.

Cladding failure detection

A further measurement system, known as the "cladding failure detection" (DRG: *détection de rupture de gaine*) system comes into play if a cladding component is damaged. Even a small crack will allow sodium to come into contact with the fuel, and, as a result, carry outside certain **radioactive atoms**, which are liable to emit neutrons as they undergo radioactive **decay**. These atoms will be swept along the entire primary circuit by the sodium, allowing their detection through neutron counting. The sensitivity of this system allows the reactor to be shut down before damage to the fuel results in any significant loss of fissile material. The "conventional" arrangement involves bleeding off a small amount of sodium, at a suitably selected point in the primary circuit hot collector, and channeling it past detectors. One alternative being contemplated envisages immersing fission chambers right in the sodium, to do away with the sampling circuit, and gain on system response time. The same arrangement would allow detection of more severe degradation, whereby fissile material would be liable to be released in significant amounts. Emergency shutdown of the reactor would then be triggered automatically.

Operating assistance systems

All of the above systems play a part in safety. They are linked to the reactor emergency shutdown system. There are, further, systems that do not trigger shutdown, their purpose being to safeguard the investment, or lower operating costs. This applies to the cladding failure localization (LRG: *localisation des ruptures de gaine*) system, making it possible to ascertain which assembly may be exhibiting a crack. The principle is the same one as that used for the DRG system, however in this case a small flow of sodium is drawn off directly at the outlet of each assembly. Another operating assistance system, VisuS (Visualization in Sodium), comes as a simplified version of what is being considered for in-service inspection. This allows, through an ultrasonic pulsed-echo technique, the detection of unintended mechanical bonding between rotating plugs and the core, when they are to be actuated. Operational versions use ultrasonic transducers, positioned at moderate-temperature (40 °C) locations, and liquid waveguides, filled with NaK (a mix of sodium and potassium, which is liquid at 40 °C). The entire setup remains in position, even when the reactor is at full power. CEA has developed high-temperature ultrasonic transducers (TUSHTs), withstanding temperatures higher than 550 °C, which are being considered for use directly in sodium.

The notion of "listening" for sounds emitted by certain incidental events has led to the currently ongoing development of "acoustic detection" systems. These are targeted at sodium boiling within the core; the sodium–water reaction, in the event of a steam-generator tube failure; and mechanical failures resulting in impacts. Further consideration is being given to making use of the attenuation of ultrasounds, caused by the presence of bubbles in the sodium, to detect a gas intrusion.

Finally, a hydrogen detection system provides the ability to detect traces of hydrogen, yielded by sodium–water interaction. The conventional sensor, comprising a nickel membrane separating the sodium from a space in which a device measures the incoming hydrogen flow rate, makes use of the fact that hydrogen readily diffuses across nickel. An electrochemical probe system, currently being developed and tested, may provide an alternative.

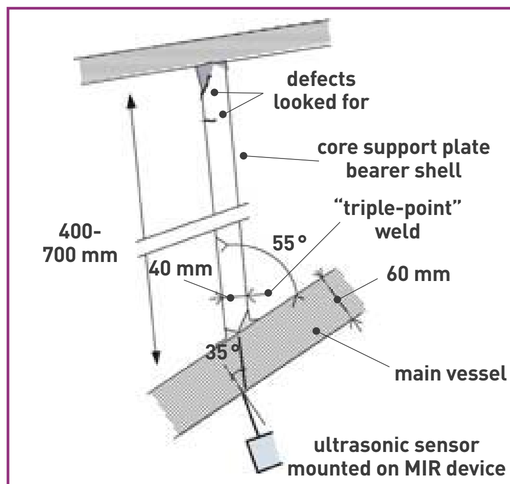


Figure 2.
Testing the Superphénix core support-plate bearer shell.

Criteria for in-service inspection

Carried out during reactor shutdowns, in-service inspection involves global tests (telemetry measurements), and local tests (nondestructive weld testing). Historically

"Integral" and "loop" reactors: equivalent maintenance conditions

One major parameter, determining the ability to carry out maintenance of the reactor block, is structure accessibility. In this respect, is there a major difference to be found, between the two main configurations for SFRs, namely the integral reactor, and the loop reactor?

At first blush, the fact that, in the latter configuration, the primary circuit extends out of the main vessel should make it more readily accessible for maintenance. However, since this circuit channels active sodium, during reactor functioning, and would possibly be contaminated, in the event of an accident, it is imperative it be adequately confined, consequently restricting access to the internal structures. Use of welded joints further exacerbates the issue.

In a loop reactor, the primary circuit structures must, moreover, be subjected to increased monitoring, since failure of these structures would cause more far-reaching consequences than if they were contained inside the vessel. This results, in particular, in a greater length of weld requiring inspection, and likewise in the use of double-wall piping for the primary circuit. Indeed, safety makes it imperative that the outer wall should have the ability to take on the confinement functions ensured by the inner wall; it must therefore, in effect, be perfectly leakproof. The possibilities as regards inspection and maintenance of the inner wall are consequently reduced.

On the other hand, maintenance of primary pumps, and intermediate heat exchangers does seem to be easier, in a loop reactor, as these components are located in auxiliary vessels, smaller than the main vessel used in the integral concept. Provisions to ease maintenance (larger-diameter upper manhole, side access...) are more readily implemented in the case of a smaller, auxiliary vessel.

As for the main vessel, its internals, and components of the energy conversion system, the inspection and maintenance issues are fairly similar, for both types of reactor. Indeed, optimization of the main vessel results in its diameter being reduced as much as feasible, which does not make maintenance any simpler.

To conclude, the maintenance conditions for both concepts prove to be broadly similar: the advantage gained as regards pumps and heat exchangers, in the loop system, is mitigated owing to the existence of double-wall external primary circuit piping, while the primary vessels afford equivalent maintenance opportunities.

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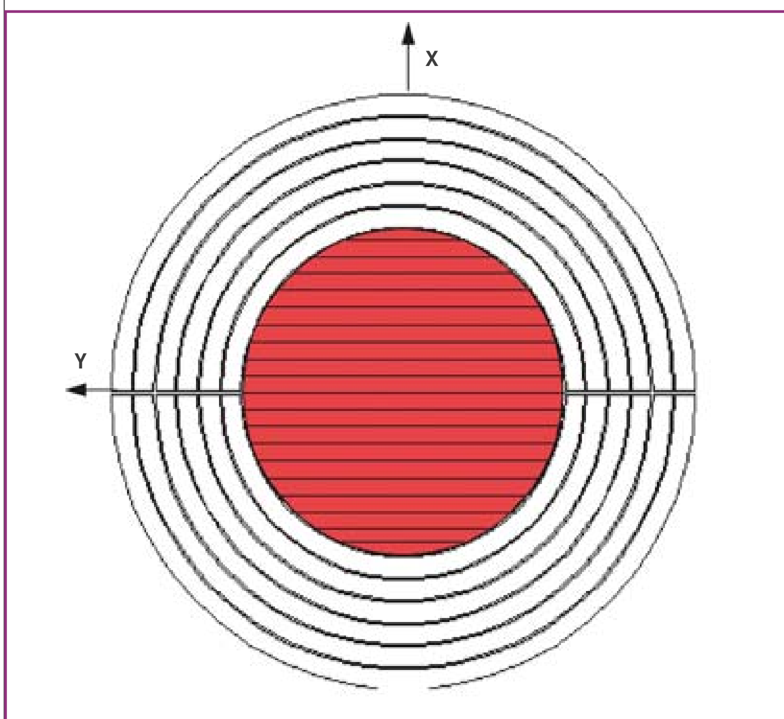
inaugurated with the inspection of welds in the main vessel of Superphénix, in-service inspection of fast reactors has expanded, to cover many other structures in the reactor block (see Figure 1), and outside it.

The procedure (called "risk informed approach") sets out, for every component, the failure modes foreseen (cracking, breaking, gradual deformation, etc.), specifying two particular deformation thresholds: a so-called "*notification*," alarm *threshold*, indicating a non-critical failure, and a so-called "*acceptability*," maximum threshold, related to level A of the French (AFCEN) Design and construction rules for mechanical components of FR nuclear islands (RCC–MR: *Règles de conception et de construction des matériels mécaniques des îlots nucléaires RNR*). For each structure, the computed potential damage is then associated to the severity of the failure risk, with regard to its safety function. The reactor's crucial sensitive components may thus be identified, for which R&D must demonstrate the feasibility of processes that, once industrialized, and implemented on site, will enable measurement of notification and acceptability thresholds.

Structural inspection through nondestructive testing

Techniques had to be developed, for the volume inspection (i.e., probing inside the material) of internal structures immersed in sodium. A robot was thus developed, at the time, to test, from the intervessel space – i.e., from outside of the sodium environment – welds in the main vessel of Superphénix. The ultrasonic sensors used (known as *transducers*), made from a composite material, featured direct focusing, by shaping a piezoelectric pellet. The performance these transducers exhibited, in terms of detection and dimensioning, depended on sensitivity, and damping. A finding analysis aid tool served to model echoes, as related to weld profiles, evaluate the influence of local geometry on fault response, and take into account the **anisotropic**, heterogeneous structure of welds.

Figure 3.
Multi-element transducer for the testing of main vessel butt welds.



Towards *in situ* testing of vessel internal structures

As regards the vessel's internal structures, plunged in liquid sodium as they are, no *in situ* nondestructive test is available as of yet. Development of such tests forms the next stage: the issue, in particular, is that of finding materials that prove suitable both for sodium, and ultrasonic volume inspection.

Testing a metallic wall immersed in liquid sodium is feasible as of now, using the metal plate subjected to inspection as an ultrasonic waveguide (see Figure 2). Testing of the conical shell in Phénix, in 2000, could thus be carried out successfully.

A major avenue for advances is provided by the use of multi-element transducers to test vessel welds, allowing beam focusing parameters to be altered according to the geometry, or thickness being tested. Such optimization of the transducer for each configuration makes for improved fault characterization, and dimensioning, while reducing perturbation effects due to the geometry. A transducer has been designed, and constructed on this basis (see Figure 3): by adjusting the delay laws for each one of its 32 elements, different angles of incidence (45° or 70°) may be obtained, along with a variable depth of focus.

Other methods are being considered, such as a main vessel test method based on the detection of **radioisotope** ^{24}Na – which is present in the primary circuit sodium – by gamma spectrometry.

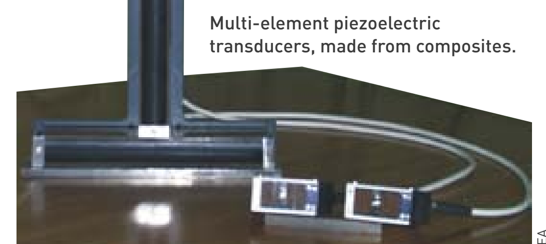
Sodium environment inspection techniques

R&D investigations have been initiated, to develop the resources required for the volume inspection of internal structures, in a sodium environment.

Two nondestructive test methods have been looked into. The first one, an electromagnetic method, uses electromagnetic acoustic transducers (EMATs), for the purposes of generating ultrasound in sodium by means of eddy currents.[4] The second method uses piezoelectric transducers, made from composite materials. The aim is to develop transducers that are chemically, and acoustically compatible with sodium. A further method, making use of multi-element transducers, is also being developed. The aim? To develop a method allowing the entire volume subjected to inspection to be probed, when the possibilities, as regards transducer translation motions, are restricted.

Ultrasound visualization in sodium

Visualization in sodium meets a threefold requirement: to assist in the positioning of an inspection instrument carrier inside the reactor; ascertain whether an object is missing (and look for it); and carry out remote measurements.



Multi-element piezoelectric transducers, made from composites.

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For the purposes of "viewing the invisible," ultrasounds (with frequencies of around 1 MHz) prove highly suitable, involving an attenuation coefficient of about 0.1 dB/m. In the simplest version, an active sonar[(5)] features a single transducer, which emits a wave, receives it after it has undergone reflection on structures, and ultimately yields an electrical signal. "Reflection" on a metallic structure actually involves three components, their sum forming the signal received (see Figure 4). The components are three kinds of echo, taken in decreasing order of amplitude. The *specular echo*, delayed with respect to emission, is dependent, as a first approximation, only on the distance between the transducer, and the closest point in the structure. The *edge of structure diffraction echo* allows – as its name implies – the structure's edges to be located. The *backscatter echo*, finally, is generated by every point in the structure, provided local roughness is not too small, compared with wavelength.

For ultrasound vision, the aim is to use the backscattered acoustic energy to localize an object, and view it in three dimensions. (In Phénix and Superphénix, the VisuS systems were already operating, using the specular echo: see above.)

A sodium vision system has been designed, featuring two perpendicular "antennas," one being a transmitting, the other a receiving antenna (see Figure 5). Their focus zones approximate to two lines, respectively horizontal, and vertical, intersecting at the point in space the system is probing. In order to move this point across the three spatial dimensions, the antennas are segmented into elements, each one being fitted with electronics allowing dynamic focusing, and the delay laws are varied, across the elements.

Computations of the order of magnitude for the ratio of backscatter echo, to specular echo, for this type of system, and anticipated roughness heights of around 10 µm, have shown that discriminating between these echoes is feasible, though not easily achieved. For corroboration on this point, and to ascertain whether weak scattered echoes "emerge" from the electronic noise, trials in water were carried out in the Navire ultrasound tank (see Figure 6). Transmission is effected by a large, single-element antenna, allowing generation of a focus line at a typical distance of 2 m. Figure 7 shows the "B-scan" image yielded by the system, in the case of 700-mm square flat plate, positioned at that distance from the antennas.

The transmitting and receiving antennas will have to be adapted to meet sodium-related environmental constraints, and the dynamically focusing transmission and reception electronics adjusted for the "orthogonal" configuration selected. Image processing will also be amenable to improvement, to reduce faults due to speckling, thus assisting object recognition, and measurement.

(4) Eddy currents: induced currents, arising e.g. in a conductor moving through a constant magnetic field, or in a static metallic solid body, subjected to a changing magnetic field. Among other applications, such currents are used for the nondestructive testing of conducting surfaces, allowing in particular the evaluation of discontinuities occurring on or beneath the surface, or the density and thickness of a wall.

(5) Sonar: the acronym for "sound navigation and ranging;" a device for the localization of objects, using the propagation of sound waves, and their reflection on these objects.

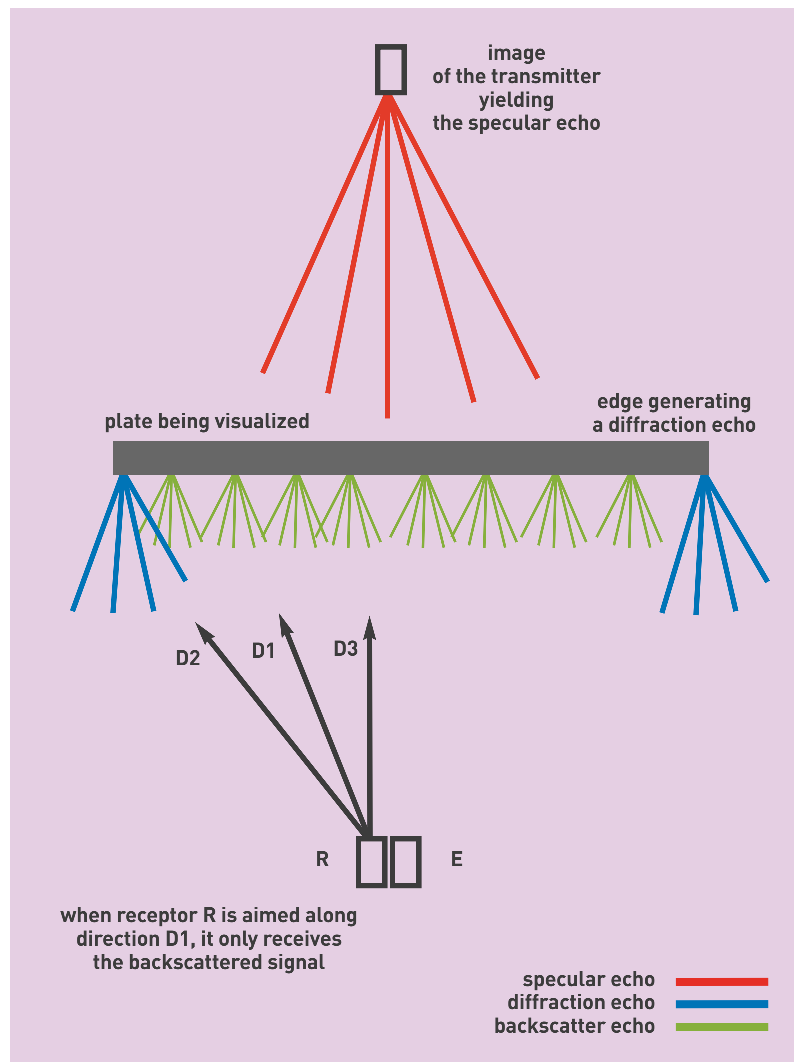


Figure 4.
Ultrasound echoes. "Reflection" on a metallic surface actually involves three components, their sum forming the signal received.

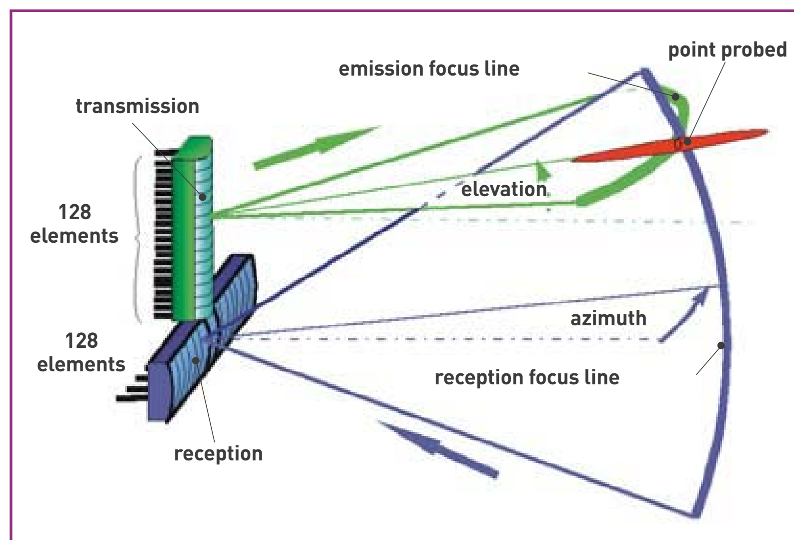
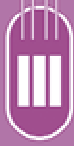
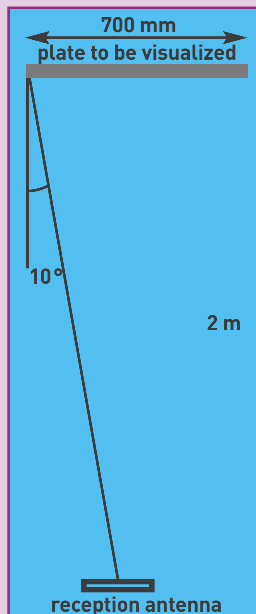


Figure 5.
Orthogonal ultrasound imaging. The characteristics of such a "camera" might be as follows: L = 500 mm; number of elements per antenna = 128; image aperture = 30°, i.e. 1,020 mm at a distance of 2 m; frequency 1.6 MHz (λ = 1.5 mm); longitudinal resolution Δr = 4 mm, angular resolution $\Delta\alpha$ = 0.33° (i.e. Δy = 11 mm at a distance of 2 m); acquisition time for a single sector image (= a single slice) (analog solution) Δt = 65 ms; full image (= 64 sector images) 64 Δt = 4 s.





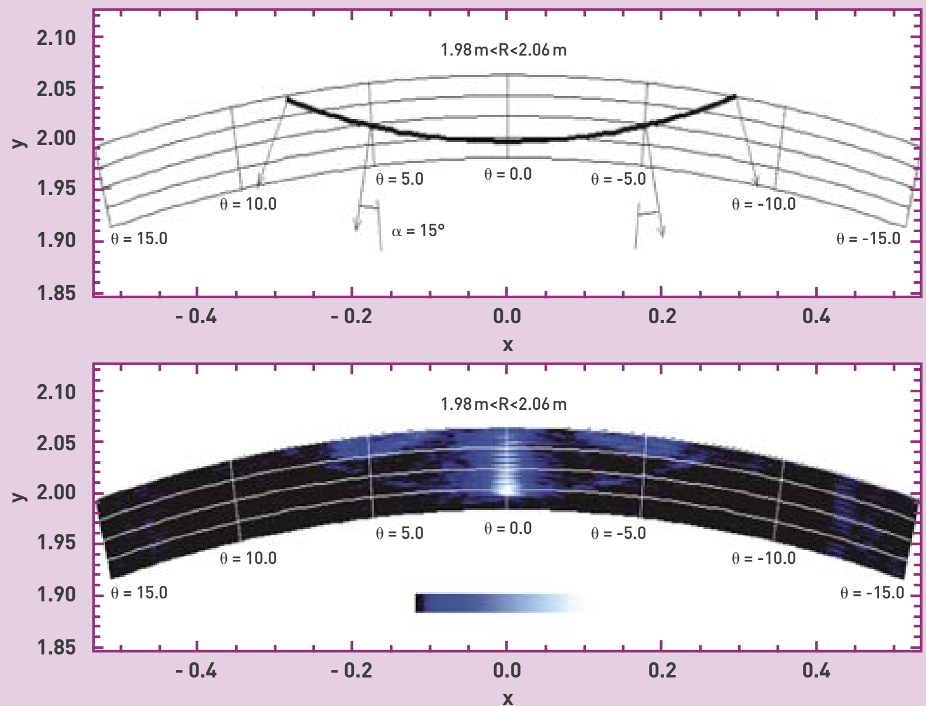


Figure 6.
Left. The Navire ultrasound tank.

Figure 7.
Right. B-scan image obtained in water, in Navire. All points in the plate are visible, with glare due to the specular echo at the point closest to the antennas (in this instance, the center of the plate), and excess intensity at the edge of the plate, due to edge diffraction echoes.

Cover gas space inspection by laser telemetry

At locations where liquid sodium is covered by a neutral gas (argon, as a rule), tests can be carried out. A number of procedures may be used, to carry out measurement of distances in fast reactor cover gas spaces,(6) during shutdowns. Laser telemetry provides, at first blush, the most accurate, and simplest technology to use. The uncertainties associated to such measurement do need, however, to be evaluated, and an inspection methodology must be specified. Indeed, a number of processes may disturb light beam propagation: thermal gradients, and fluctuations, on the one hand, and sodium deposits, and aerosols,(7) on the other.

The approach selected involves independently modeling the factors of thermal origin, and the presence of aerosols. Investigations on the influence of sodium deposits on the surfaces targeted involve experiments only. Glovebox tests have made it possible to specify possible measurement conditions for sodium-covered surfaces (droplets, film), according to surface state (roughness), and laser beam angle of incidence.

Tests in the FRUCTIDOR facility, at Cadarache, have allowed observation of the effects of the presence of sodium aerosols on light beam propagation. Putting all these findings together makes it possible to draw up specifications for the telemetry equipment, and specify the operating methodology.

(6) Cover gas space: the upper region of the reactor's primary circuit, extending between the free surface of the sodium and the vessel roof slab, this space – 1–2 m high – being filled with a neutral gas (argon).

(7) Aerosol: a suspension, in a gas (or a gas mixture), of very fine solid or liquid particles.

Two types of high-temperature ultrasonic transducers (TUSHTs).

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H-meter, for the detection of hydrogen. The hydrogen that may be contained in the sodium circulating along the four nickel tubes diffuses across that metal, arriving in the space under vacuum surrounding the tubes, where a spectrometer measures concentration of the gas.

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Prospects for Generation IV R&D

Over the past 15 years, the R&D effort, as regards inspection, has made it possible to meet the concrete requirements, for existing fast reactors. As of now, taking on board "inspectability," from the design stage on, for reactors of the new generation will make it possible to optimize testing, by going for improved accessibility for the zones requiring inspection, and better component "dismountability." Development of ever-higher-performance sensors, and methods augurs well as to the feasibility of inspection being carried out within the liquid sodium coolant itself.

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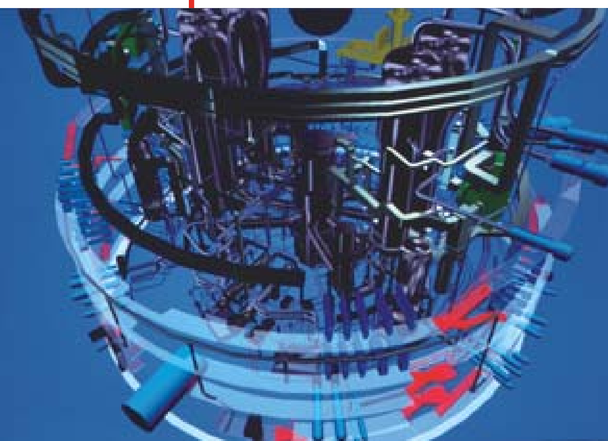
FOCUS A

The components of a nuclear system

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

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



Areva NP

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

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

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

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

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

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

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

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

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

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

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

FOCUS B

Reactor lines, generations, and neutron spectra

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

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

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

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

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

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

M. Brigaud/ EDF Médiathèque

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

FOCUS (Cond't) B

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

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

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

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

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

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

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

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

FOCUS C

Thermodynamic cycles and energy conversion

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

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

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

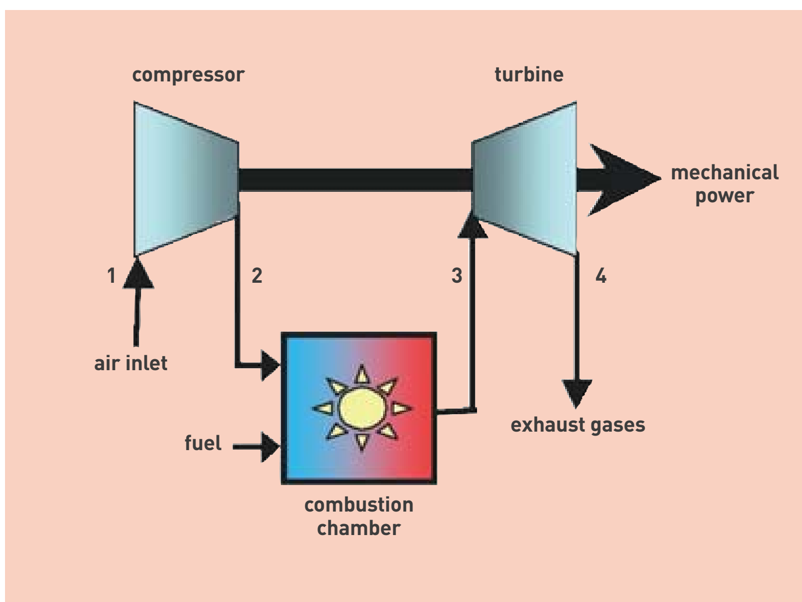


Figure.
Brayton cycle, as implemented in an open-cycle gas turbine.

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

FOCUS D

What is multiphysics, multiscale modeling?

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

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

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

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

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

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

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

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

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

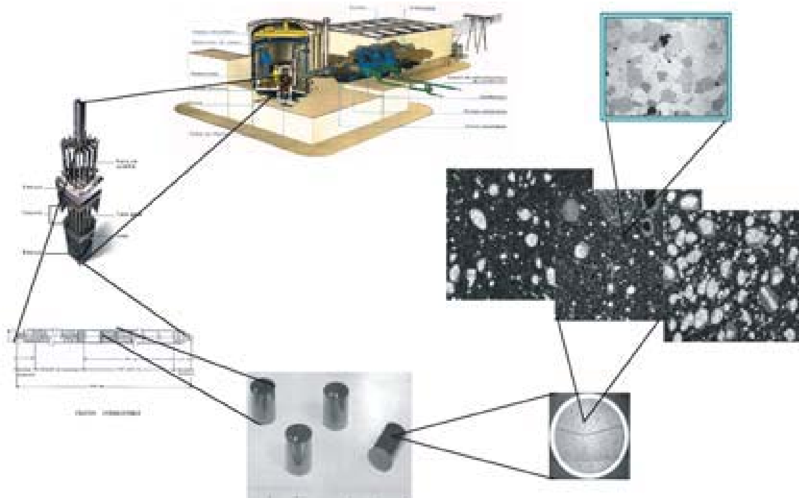


Figure.
Improving nuclear fuel reliability, and cost-effectiveness calls for finescale modeling of that fuel, through a multiscale approach, from reactor to fuel microstructure (in this instance, MOX fuel). Microstructural characteristics (porosity, cluster size and distribution, grain size...) have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

FOCUS E

The main families of nuclear materials

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

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

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

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

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

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

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

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

Crystal structures

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

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

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

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

Steel families

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

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

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

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

Areva NP

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

FOCUS (Cond't) E

which have been subjected to *annealing*: they become martensitic when quenched, in air or a liquid, after being heated to reach the austenitic domain. They subsequently undergo softening, by means of a heat treatment. They may contain nickel, molybdenum, along with further addition elements. These steels are magnetic, and exhibit high stiffness and strength, however they may prove brittle under impact, particularly at low temperatures. They have gained widespread use in the nuclear industry (fastenings, valves and fittings...), owing to their good corrosion resistance, combined with impressive mechanical characteristics.
Austenitic steels, characterized by a *face-centered cubic* structure, contain some 17–18% chromium, 8–12% nickel (this enhancing corrosion resistance: the greater part, by far, of stainless steels are austenitic steels), little carbon, possibly some molybdenum, titanium, or niobium, and, mainly, iron (the remainder). They exhibit remarkable **ductility**, and **toughness**, a high expansion coefficient, and a lower **heat conductivity** coefficient than found in ferritic–martensitic steels. Of the main grades (coming under US references AISI[2] 301 to 303, 304, 308, 316, 316L, 316LN, 316Ti, 316Cb, 318, 321, 330, 347), 304 and 316 steels proved particularly important for the nuclear industry, before being abandoned owing to their excessive swelling under irradiation. Some derivatives (e.g. 304L, used for internal structures and fuel assembly end-caps, in PWRs; or 316Tiε, employed for claddings) stand as reference materials. In fast reactors, they are employed, in particular, for the fabrication of **hexagonal tubes** (characteristic of reactors of the Phénix type) (316L[N] steel), while 15/15Ti austenitic steel has been optimized for fuel **pins** for this reactor line, providing the new cladding reference for fast reactors.

Austenitic–ferritic steels, containing 0%, 8%, 20%, 32%, or even 50% ferrite, exhibit good corrosion resistance, and satisfactory weldability, resulting in their employment, in molded form, for the ducts connecting vessels and steam generators.
One class of alloys that is of particular importance for the nuclear industry is that of **nickel alloys**, these exhibiting an austenitic structure. Alloy 600 (Inconel 600, made by INCO), a nickel (72%), chromium (16%), and iron (8%) alloy, further containing cobalt and carbon, which was employed for PWR steam generators (along with alloy 620) and vessel head penetrations, was substituted, owing to its poor corrosion resistance under stress, by alloy 690, with a higher chromium content (30%). For certain components, Inconel 706, Inconel 718 (for PWR fuel assembly grids), and Inconel X750 with titanium and aluminum additions have been selected, in view of their swelling resistance, and very high mechanical strength. For steam generators in fast reactors such as Phénix, alloy 800 (35% nickel, 20% chromium, slightly less than 50% iron) was favored. Alloy 617 (Ni–Cr–Co–Mo), and alloy 230 (Ni–Cr–W), widely employed as they are in the chemical industry, are being evaluated for gas-cooled **VHTRs**.
Ferritic–martensitic steels (F–M steels) exhibit a *body-centered cubic* structure. In effect, this category subsumes the martensitic steel and ferritic steel families. These steels combine a low thermal expansion coefficient with high heat conductivity. Martensitic or ferritic steels with chromium contents in the 9–18% range see restricted employment, owing to their lower creep resistance than that of austenitic steels. Fe–9/12Cr martensitic steels (i.e. steels containing 9–12% chromium by mass) may however withstand high temperatures, and are being optimized with respect to creep. For instance, Fe–9Cr 1Mo molybdenum steel might prove suitable for the hexagonal tube in **SFR** fuel assemblies. Under the general designation of AFMSs (advanced ferritic–martensitic steels), they are being more particularly investigated for use in gas-cooled fast reactors.
Oxide-dispersion-strengthened (ODS) **ferritic and martensitic steels** were developed to combine the swelling resistance exhibited by ferritic steels, with a creep resistance in hot conditions at least equal to that of austenitic steels. They currently provide the reference solution for fuel cladding, for future sodium-cooled reactors.
The **cladding material** in light-water reactors, for which stainless steel had been used initially, nowadays consists of a **zirconium alloy**, selected for its "transparency" to neutrons, which exhibits a *compact hexagonal* crystal structure at low temperature, a *face-centered cubic* structure at high temperature. The most widely used zirconium–iron–chromium alloys are tin-containing **Zircaloys** (Zircaloy-4 in PWRs, Zircaloy-2 in BWRs, ZrNb – containing niobium – in the Russian VVER line), owing to their outstanding behavior under radiation, and capacity with respect to creep in hot conditions.
After bringing down tin content, in order to improve corrosion resistance, a zirconium–niobium alloy (M5®) is presently being deployed for such cladding.
Among nuclear energy materials, **graphite** calls for particular mention: along with heavy water, it is associated with reactors that must operate on **natural uranium**; it proves advantageous as a **moderator**, as being a low neutron absorber.
For **GFRs**, novel ceramics, and new alloys must be developed, to the margins of high fluences. Researchers are storing high hopes on refractory materials containing no metals.
In particle fuels, uranium and plutonium oxides are coated with several layers of insulating **pyrocarbons**, and/or silicon **carbide** (**SiC**), possibly in fibrous form (**SiCf**). These are known as coated particles (CPs). While SiC-coated UO_2, or **MOX** balls stand as the reference, ZrC coatings might afford an alternative.
At the same time, conventional **sintered** uranium oxide (and plutonium oxide, in **MOX**) pellets might be supplanted by advanced fuels, whether featuring chromium additions or otherwise, with the aim of seeking to overcome the issues raised by **pellet–cladding interaction**, linked as this is to the ceramic fuel pellet's tendency to swell under irradiation.
Oxides might be supplanted by **nitrides** (compatible with the **Purex** reprocessing process), or **carbides**, in the form e.g. of uranium–plutonium alloys containing 10% zirconium.

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

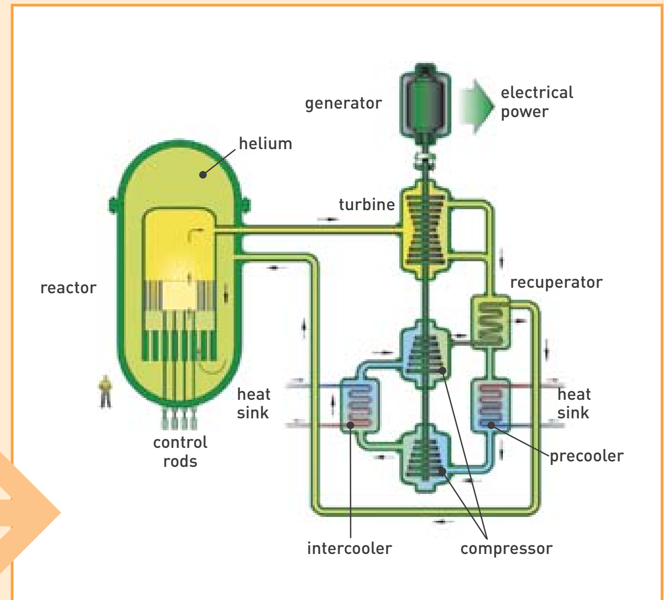
The six concepts selected by the Gen IV Forum

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

GFR

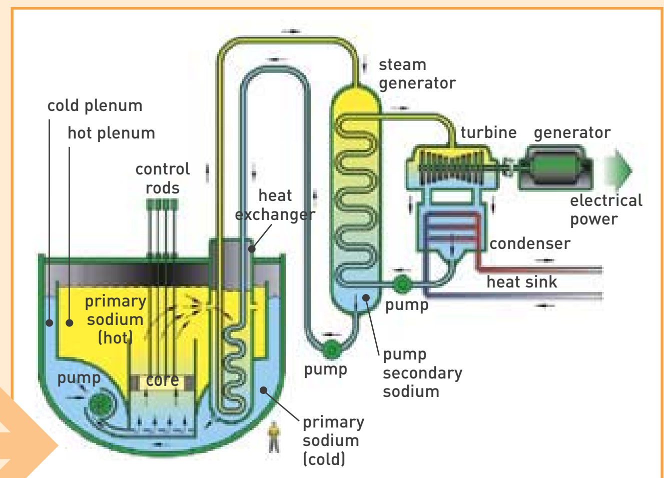
The gas-cooled fast reactor system (GFR) is a high-temperature, gas-cooled (helium-cooled as a rule), fast-neutron reactor allowing actinide recycle (homogeneous, or heterogeneous), while sustaining a breeding capability greater than unity. The reference concept is a helium-cooled, direct- or indirect-cycle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

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



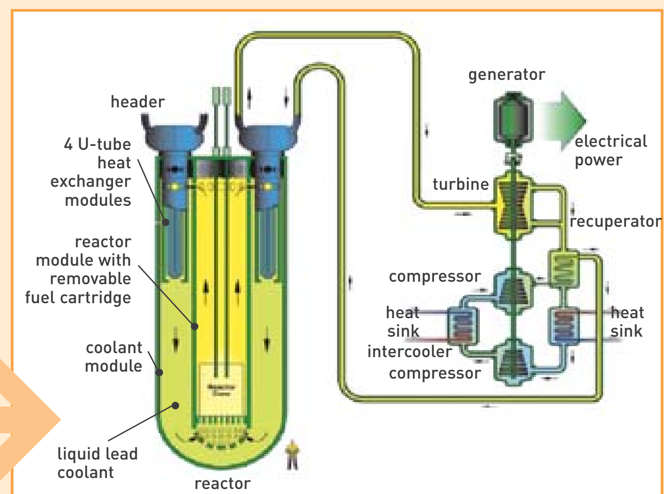
Le SFR

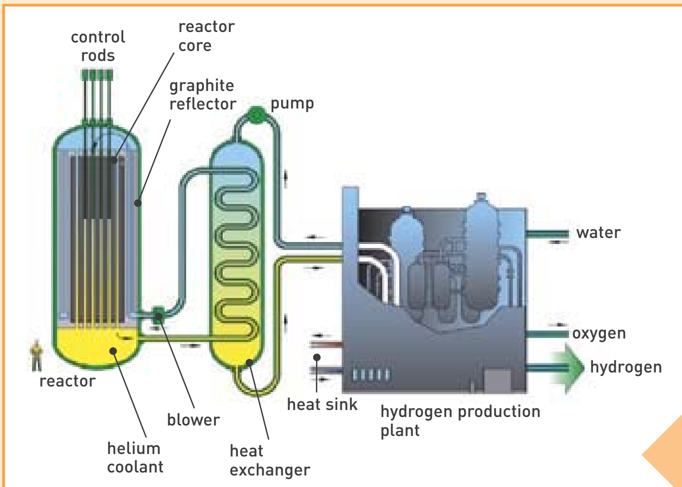
The sodium-cooled fast reactor system (SFR) is a liquid-sodium-cooled, fast-neutron reactor, associated to a closed cycle, allowing full actinide recycle, and plutonium breeding. Owing to its breeding of fissile material, this type of reactor may operate for highly extended periods without requiring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



LFR

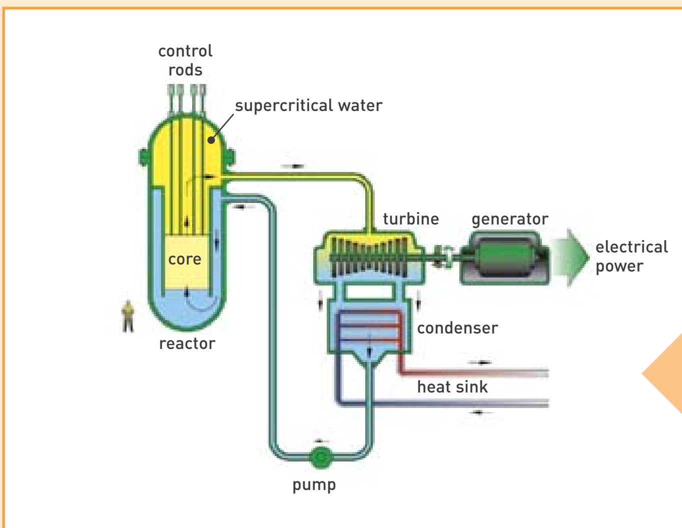
The lead-cooled fast reactor system (LFR) is a lead- (or lead-bismuth alloy-) cooled, fast-neutron reactor, associated to a closed fuel cycle, allowing optimum uranium utilization. A number of reference systems have been selected. Unit power ranges from the 50–100 MWe bracket, for so-called battery concepts,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





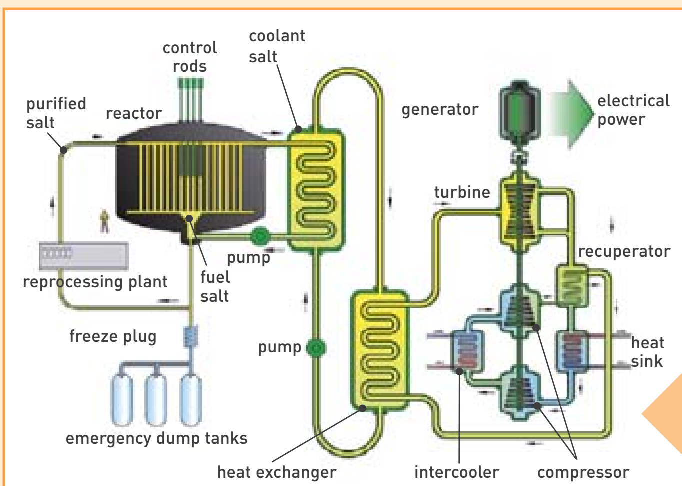
VHTR

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



SCWR

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



MSR

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