



Sodium-cooled fast reactors of the future

Using sodium as the coolant in fast reactors affords many advantages, that had already warranted opting for this technology for the first generations in this reactor line. It remains a highly promising option, being the only technology that would allow construction of a prototype fourth-generation reactor within a relatively short timescale. Many innovations are being investigated, to enhance further yet the benefits to be drawn from the technology's advantages, and improve mastery of its drawbacks.

View of the outside of the Phénix power plant, at CEA's Marcoule site. The 34 years' operating time achieved by Phénix, the prototype reactor for the sodium-cooled fast reactor line, amount to a unique operating experience, and testify to the technological maturity of this reactor line.



A. Gonnin/CEA

In any survey of **sodium-cooled fast reactors** (SFRs), it would be inconceivable not to begin by recalling the considerable experience gained with this **technology line** around the world, particularly in France (see Focus A, *The components of a nuclear system*, p. 10; Focus B, *Reactor lines, generations, and neutron spectra*, p. 14). Currently, six such reactors are operating, five of these being coupled to the power grid, delivering an aggregate power of over 1,000 **MWe**. Three further reactors are under construction (see Table 1). In France, the Phénix reactor,

at CEA's Marcoule site, has been in operation since 1973. It has been coupled to the grid for more than 100,000 hours, testifying to the technological maturity of this reactor line.

On the other hand, the experience of the European Superphénix reactor, at the Creys-Malville site (Isère département, southeastern France), does suggest a different view. This 1,200-MWe prototype reactor

Table 1.
Sodium-cooled fast reactors in operation, and under construction around the world.

reactors in operation	power (MWth)	power (MWe)	date of initial criticality
BOR-60 (Russia)	55	12	1968
Phénix (France)	563	250	1973
Joyo (Japan)	50-75/100	-	1977
BN-600 (Russia)	1,470	600	1980
FBTR (India)	40	13.2	1985
Monju (Japan)	714	280	1994
reactors under construction			
CEFR (China)	65	25	2009
PFBR (India)	1,250	500	2010
BN-800 (Russia)	2,100	880	2012



Kurchatov

The BN-600 sodium-cooled fast reactor, sited at Beloyarsk (Russia), with an output power of 600 MWe.



The Chinese CEFR (China Experimental Fast Reactor) sodium-cooled fast reactor, undergoing construction. This experimental reactor will be of the integral type, featuring two secondary sodium loops, and will use a mixed uranium-plutonium oxide as fuel. It will have an output power of 25 MWe. Initial criticality is scheduled for 2009.

was shut down in 1998 by the French government, being deemed too expensive, and overcomplex, after 12 years' operation, during which it had only generated electricity for an equivalent one year's operation at full power. As an industrial-scale prototype, Superphénix unavoidably required a development phase, ultimately demonstrating satisfactory continuous operation in 1997. Concurrently, European manufacturers had been developing an optimized 1,500-MWe version, the EFR (European Fast reactor) project.

The reason why, currently, most countries with a nuclear capability are carrying through, or reinvigorating, a wide-ranging research effort on this technology line, is primarily, on the one hand, that of meeting future uranium resource management requirements, and, on the other, to bring about a reduction in the amount of ultimate radioactive waste. Indeed, only a fast neutron spectrum affords the ability to burn all of the uranium, by generating plutonium, thus multiplying usable natural resources by a factor of around 70. The selfsame fast neutrons further allow the incineration of all long-lived radioactive waste, such as americium, neptunium, and curium, owing to the excess number of neutrons generated by each fission.

The goals assigned to such reactors have been revised, over the past few years, to take on board changes in regulations, the performance of competing reactor lines, and the best technology that will be becoming available over the longer term.

New goals

The overall goals set for fourth-generation nuclear systems have been more precisely specified for sodium-cooled fast reactors (see Box, *The six concepts selected by the Gen IV Forum*, p. 6).

Advances in terms of installation safety are required, in order to enable this reactor line to achieve a level at least equivalent to that of third-generation reactors. On the one hand, sodium-related risks will have to be minimized, whether it be fire hazards, or reactions with water. And, on the other hand, it will prove essential to curb potential core accident initiators, and limit the risk of energy release, in the event of a

severe accident. Finally, the system will have to exhibit enhanced resistance to outside aggressions.

System economic competitiveness must achieve further advances yet, if a possibly higher cost, as compared to current, or proposed, reactors, is to prove acceptable to the operator. It is essential that capital costs be brought down, by simplifying the system, and performance enhanced, whether it be in the area of energy conversion efficiency, or of fuel burnup. Further, installation availability and reliability must be maximized, and greater ease of operation achieved, through the development of more effective in-service inspection, maintenance and repair techniques. Lifetime should be equal to 60 years.

To restrict proliferation risks, designers will be looking to, initially, zero-breeding gain systems, involving no fertile blankets, in other words systems in which the plutonium generated will remain inside the fissile fuel, being recycled as part of a group management of actinides, making any partitioning of it difficult. The potential for long-lived radioac-



tive waste transmutation is to be demonstrated in the context of such homogeneous-path management. However, solutions involving minor actinide-bearing blankets – hence blankets likewise presenting low proliferation risks – will also be tested. Better societal acceptance will be conditional on the mitigation of proliferation risks, and of sodium-related risks; resource sustainability; adequate waste management; and a convincing demonstration of system safety. Overall, environmental impact will have to prove as low as possible. Reducing that impact will involve, aside from the long-lived radioactive waste transmutation capability, downsizing the amounts of liquid or gaseous effluents rejected.

The various avenues for innovation

If the goals that have been set are to be achieved, much innovation-directed research and development effort needs must be expended.

A novel, high-performance, safe core

The core of a fast reactor, operating as it does with no moderator, is not normally operated in its most highly reactive configuration. Depletion, or loss of

Construction site of the Indian PFBR (Prototype Fast Breeder Reactor) sodium-cooled fast reactor. This reactor will be of the integral type, featuring two secondary sodium loops, and will use a mixed uranium-plutonium oxide as fuel. It will have an output power of 500 MWe. Initial criticality is scheduled for 2010.



The Superphénix power plant, at the Creys-Malville site (Isère département, southeastern France). Development work for the sodium-cooled fast reactors of the future draws on the experience gained with this European reactor, in the areas of design, technology, operating techniques, and safety. The operational feedback from Superphénix further evidenced the full mastery achieved, as regards sodium handling, and handling of the components of this type of reactor.



Gilles Larvor/EDF

coolant may trigger a significant power surge. This is the case, for instance, if the sodium boils within the core, thus ceasing to ensure the modest level of neutron slowing down associated to its presence in the liquid form. This process is known as the **sodium**

void effect. Its impact on **reactivity** is expressed in terms of a multiple of the **fraction of delayed neutrons** yielded by a fission process, the unit used being the **dollar (\$)**.

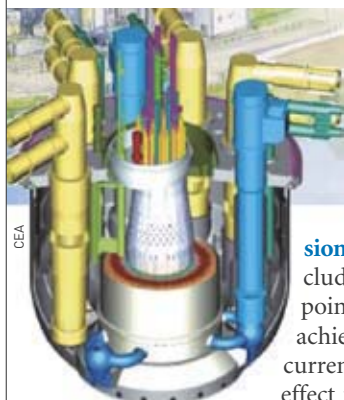
The large cores investigated, or built in the past typically exhibited a **sodium void coefficient** of around 7–8 \$, liable to entail, in the event of incidents, a large **power excursion**.

However, these reactors were designed to preclude the coolant ever being brought to boiling point. Despite the outstanding levels of prevention achieved with respect to this process, designers are currently investigating ways of attenuating this void effect in the sodium-cooled reactors of the future, so that these may benefit from higher safety levels. A number of avenues are available, however these mostly conflict with core performance, useful volume, and plutonium **breeding gain**.

Investigations are also ongoing on the sodium void effect being optimally counterbalanced by the core's natural behavior, through emulation of the various **feedback reactions**. Mention should be made of the **Doppler effect**, whereby ^{238}U **absorption cross-sections** increase as temperature rises, and geometrical effects, such as structural expansion, which lowers fissile **nucleus** concentration. A number of avenues

have been identified: reducing the amounts of sodium inside the core; positioning a large volume of sodium at the outlet (**sodium plenum**) to enhance **neutron leakage** (should that area come to the boil); the possibility of inserting a neutron moderator to soften the spectrum, i.e. to achieve lower average neutron energy; increasing neutron leakage, by acting on core shape...

The issue will be how to maintain, at the same time, plutonium **breeding** within the core, without relying at startup on fertile uranium blankets surrounding it. This points to a reduction in the proportion of (fissile) plutonium within the core, while increasing that of fertile uranium. In order to sustain adequate fissile material concentration to ensure core functioning, the proportion of fuel to coolant mass must be raised, whether by cutting down on coolant, or by using denser fuel (see *What fuel for SFRs?* p. 32). A number of parameters may thus be worked on, to design a core that will meet specifications: plutonium enrichment, the type of fuel used, fuel-to-coolant ratios, power level per unit volume, neutron **spectrum hardness**... The avenues being investigated currently concern enlarging fuel **rod** diameter, to raise the percentage for fuel inside the core while lowering the percentage of sodium; bringing down **volumetric power**, to maintain adequate thermic conditions within such larger-diameter rods; going for lower plutonium enrichment, to allow zero breeding gain; and comparing **oxide** and **carbide**, and even metallic, **fuel** performance.

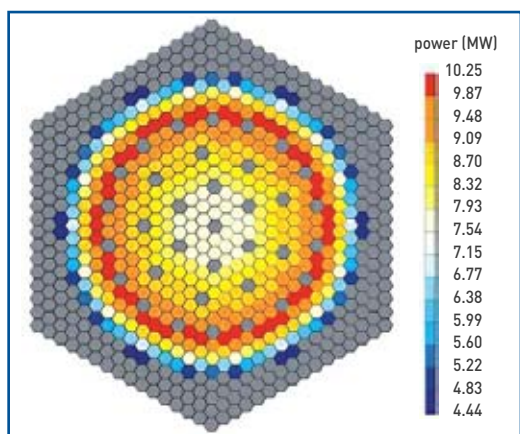


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The EFR (European Fast Reactor) project is a 1,500-MWe sodium-cooled fast reactor, designed by a consortium of European organizations, including **Areva**, **EDF**, and **CEA**, taking on board the experience gained with Superphénix. The EFR program was launched in 1988, then frozen in 1998. Core performance for future SFRs is to be compared with that for EFR, standing as it does as a European reference.

core options	EFR type	SFR MOX	SFR MOX + moderator	SFR carbide	SFR carbide + moderator
rod diameter (mm)	6.9	9.5	9.5	9.5	9.5
Pu mass (t)	8.8	10.5	11.8	8.5	9.5
internal breeding gain	- 0.13	+ 0.04	- 0.06	+ 0.11	+ 0.02
power density (MW/m ³)	303	230	230	290	290
burnup (GW · d/t)	128	106	119	91	105
linear power (W/m)	440	550	555	640	660
sodium void coefficient (\$)	6–8	4.5	3.5	4.2	3.3

Table 2.
Performance comparison for various fourth-generation SFR core configurations, and for the EFR project core.



Map of the assemblies in a reference core, in a 3,600-MWth (1,500-MWe) SFR, shown as a function of the power they release. This is a zero-breeding gain, blanket-free core, featuring enhanced safety parameters, compared to EFR values.

The initial results achieved should be assessed in the light of the core performance for the EFR project, the European reference for sodium-cooled fast reactors in the 1990s. While meeting the goals for zero breeding gain (positive **internal breeding gain**), while not incurring excessive degradation with respect to other performance points (immobilized plutonium mass, core size, as measured by **power density**, and refueling rate, as yielded by burnup), these confirm that the sodium void coefficient, standing as it does at about 7 \$ for the EFR core, may effectively be brought down by several dollars yet (see Table 2).



The MASURCA reactor, sited at Cadarache, viewed from below. The avenues for innovation being suggested for the design of the SFR cores of the future will involve experimental R&D programs, to be carried out in critical mockups, including MASURCA.

At the same time, there are a number of configurations that allow the Doppler effect to counterbalance, to some extent, the sodium void effect. This finding thus warrants a degree of flexibility as regards overall safety optimization for future cores. Indeed, such optimization must also take on board the entire range of “risk situations,” such as loss of flow, loss of heat sink, power **transients** in the hypothetical event of further failure of the protection system involving the dropdown of **shutdown rods**.

Suggested technical solutions for the design of such cores (novel materials, arrangements for neutron leakage...) will require empirical corroboration, involving use of **critical mockups**.

Reducing severe accident risk

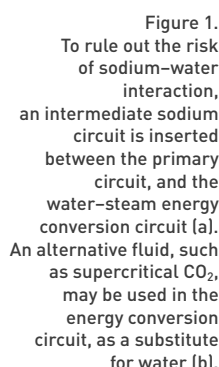
The reactor will have to be designed in such a way as to render altogether hypothetical the occurrence of total **core meltdown**. Nevertheless, as part of in-depth defense precautions, and to verify the absence of accident risks involving unacceptable consequences, core meltdown accidents are to be investigated, and their controllable character must be demonstrated. For that purpose, one of the goals set for the designers of the core, and of the **fuel assembly** will be to preclude the risk of any significant release of mechanical energy in the event of such an accident occurring. This target induces constraints on core geometrical design, so as to preclude the molten core reaching a configuration liable to result in mechanical energy release. The new sodium-cooled core concepts will be subjected to systematic analysis of the possible severe accident scenarios that might lead to core meltdown. These scenarios will include equally those considered for Superphénix (pump stoppage with no dropdown of shutdown rods, and no kicking in of the backup diesels), and those for EFR (total, instant blockage of a fuel assembly, with delayed detection).

The design of core assemblies will have to be revised, and modified, to include e.g. features favoring molten fuel dispersion, and **neutron absorbent** materials, which, by mixing with the molten fuel, will prevent it reverting to **criticality**.

To complement this, the capabilities of the molten core catcher are to be enhanced, on the basis in particular of the knowledge gained from the **corium** catcher development programs carried out for third-generation pressurized-water reactors (**EPR**). Thus, one avenue for investigation being considered is the use of various layers of sacrificial materials (i.e. materials the properties of which are only intended to come into play one single time, in particular to mitigate the consequences of an accident) – some for the purposes, as in **PWRs**, of favoring the spreading out of the corium, to attain a “coolable” configuration, and preclude the catcher being pierced, the others, of the **neutron absorber** type, to bar any possible high-plutonium-content fuel-related corium **recriticality** occurrence.

Suppressing the sodium-water risk

Liquid sodium reacts with water in highly energetic fashion, with concomitant release of **hydrogen**. The **steam generator** is thus a peculiarly sensitive component in the system. The possibility of such interac-



Supercritical CO₂ may equally be used for energy conversion (see Figure 1b). The critical point, for CO₂, lies at 31 °C and 7.3 MPa. Around this point, it behaves like a liquid, and compression work can be minimized. Above it, it exhibits the properties of a highly compressible gas, and lends itself to highly efficient work. With a so-called *split-flow* cycle, whereby part of the fluid exiting the low-temperature recuperator is bled off to be reinjected, by way of a compressor, into the high-temperature recuperator inlet, thus bypassing the cold sink, calculations indicate that an installation efficiency of 45%, for a core outlet temperature of 550 °C, may be achieved. Efficiency rises to 50% for a core outlet temperature of 650 °C. This fluid thus appears highly attractive. Nevertheless, it does raise a number of issues, including its chemical reactivity with sodium, which requires proper

evaluation; and the lack of an extant, associated industrial technology.

Research to seek the highest-performance materials

Nowadays, the materials used for sodium technology components and circuits have been fully qualified. In particular, 316L(N) stainless steel – a member of the **austenitic steel** family of materials – which is used for the reactor vessel, but equally for internal structures, and intermediate circuits, is fully mastered with respect to manufacture, and employment (see Focus E, *The main families of nuclear materials*, p. 76). Its only potential limitation lies in its use for structures reaching the temperature of the reactor's hot collector, should temperatures rise above 550 °C, over a dimensioning lifetime set at 60 years for future SFRs. In such conditions, it is mainly **creep** resistance that proves inadequate, for this steel grade. If some of the innovative circuits mentioned above are to be used, a switch in materials may thus be required.

The problem is a difficult one, and more highly alloyed steels (stabilized nickel–chrome steels), or even iron bases, of the Alloy 800 type, or nickel bases, should temperature reach 650 °C, are being considered for construction of the major fixed, hot structures in the reactor block.

For more readily accessible, removable components (core cover plug, heat exchangers, pipes), **ferritic** or **martensitic** materials are being considered. There are, indeed, grades in the 9–12% chromium classes offering satisfactory creep resistance, while exhibiting smaller expansion coefficients, allowing, for circuits subjected to restricted-expansion stresses, construction of shorter circuits; and higher **thermal conductivity**, resulting in lower thermal **stresses** across component thickness.

Concurrently, inside the core, fuel **cladding** temperatures (at nominal power, or during transients) may reach higher levels. At the same time, neutronics optimization entails greater constraints with regard to

geometry, with larger-diameter **pins**, and smaller-diameter spirally wound spacer wire. To preclude pin **swelling**, in **irradiation** conditions, resulting in contact occurring between pins, with the cladding overheating to such an extent as to cause damage to it, use of a material must be considered, that will exhibit lower swelling than the current 15/15Ti ϵ optimized austenitic steel. **Oxide dispersion-strengthened (ODS) ferritic–martensitic** steels, as investigated in particular by Japanese teams, or even advanced austenitics are potential candidates (see *Metallic materials, one of the keys for the fourth generation*, p. 71).

Nuclear waste incineration

Mitigating the harmful properties of nuclear waste inescapably entails the transmutation of long-lived waste in the reactor core. The **elements** involved are the minor actinides – americium, neptunium, and curium. Two processes are being investigated, to take advantage of the fast reactors' minor actinide transmutation capability. The first process consists in blending the minor actinides with the fuel, throughout the reactor core. This is **homogeneous-mode** incineration. The advantage lies in the relative actinide content remaining low, about 1%, in the fuel, while its behavior under irradiation raises no particular issues. The major disadvantage is that the entire fuel fabrication plant must be shielded, making it exceedingly costly. The second process involves incinerating minor actinides in dedicated assemblies, which may be positioned e.g. at the core periphery, as a blanket might be. If the substrate **matrix** for the fuel material is **UO₂**, then such a blanket will yield plutonium even as it incinerates minor actinides, enhancing core performance. The drawback is that minor actinide concentration, in dedicated fuel, is high (25–40%), making for fabrication difficulties, and problematical handling.

A collaboration set up between CEA, **DOE** (United States), and **JAEA** (Japan) has drawn up a specific experimental program, for the fabrication of a minor-actinide charge-bearing fuel assembly, from **reprocessed**



The Monju experimental sodium-cooled fast reactor, in Japan. Currently closed down for alterations, it is due to be recommissioned in 2009. It is planned to irradiate, in this reactor, a minor-actinide charge-bearing fuel assembly in 2020, under the aegis of the GACID program.

MOX, and its irradiation in the Monju experimental sodium-cooled reactor, in Japan. This program, named GACID (Global Actinide Cycle International Demonstration), is to be carried out over the long term, with irradiation scheduled for 2020, since this requires construction of a pilot assembly fabrication workshop, and operational feedback from Monju, which is due to be recommissioned in 2009.

Advanced reactor concepts

The various innovations that have been suggested take on their full significance when seen as part and parcel of a complete reactor concept. Indeed, it is by looking at the overall architecture that the designer can check the consistency of the various options, refine R&D issues, and identify possible gaps, work out performance, in terms both of nuclear materials management and safety, through computation of functioning modes, investigation of transients, and of specific conditions, such as **natural convection**, and in terms of economics. Two main avenues have currently been selected, for an innovative sodium-cooled reactor concept.

Their common feature is to specify a core exhibiting a low void coefficient, to be optimized in each boiler, to allow for margins in the various accidental and incidental transient situations. Moreover, the core will be designed to exhibit satisfactory behavior in the event of severe accident, i.e. to present a low high-energy-release event probability. The reactor will further feature a structure having the capability to act as in-vessel molten material catcher. These cores will likewise draw on innovations in the fuel area, with the ability to take in minor actinides, in accordance with the two strategies outlined above. The boilers in turn will have to feature a significantly enhanced diagnostic capability, by way of instrumentation, improved in-service inspection potential, and repair facilities for the more critical components, to ensure availability (see *Sodium-cooled reactors: towards new advances in monitoring and in-service inspection techniques*, p. 85).

The first avenue is based on the integral concept, in which the reactor's primary circuit is wholly enclosed within the primary vessel, as in Phénix, and Superphénix (see Figure 2a). The reactor block benefits from relative compactness, involving as it does a lower steel and concrete mass requirement for its

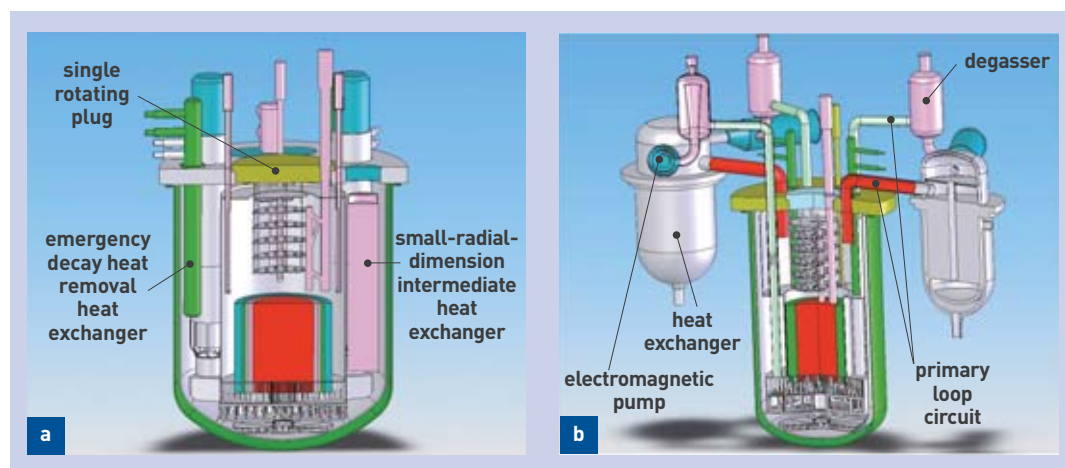
construction, through the use of compact heat exchangers. In this innovative integral concept, the energy conversion circuits are of the water–steam type, with redesigned steam generators, compared to the options for Superphénix and EFR. The intermediate circuits, few in numbers, hold a fluid exhibiting little or no reactivity with water, and sodium compatibility, such as a lead alloy, or a molten salt, eliminating risks of sodium–water interaction. These circuits may be constructed from **ferritic–martensitic** chromium steel, to make them more compact; here too, gains in construction costs are anticipated.

It is in the second concept, of the so-called loop type, that the more ambitious innovations come together (see Figure 2b). This type of architecture allows the intermediate sodium circuit to be done away with, while a gas energy conversion circuit is used. There is thus no remaining risk of sodium–water interaction. This circuit employs nitrogen, or – a better option – supercritical CO₂. On the other hand, design of protections to counter the risk of gas flowing into the core must be provided for, and will prove decisive, for concept viability. To ensure adequate efficiency, core outlet temperature may be raised by 50–100 °C, particularly if a conventional gas is used. The materials for the core, components, and circuits must have the ability to withstand such a temperature over a lifetime of 60 years. The materials surveyed above are to be systematically used in this respect, by going for, at first blush, ferritic–martensitic steels, whether for fuel cladding – in an oxide dispersion-strengthened version – or for removable high-temperature components, and circuits (see *Metallic materials, one of the keys for the fourth generation*, p. 71).

While affording safety levels that are higher still, for performance levels equivalent to, or better than, those of the preceding concept, this type of reactor does require, obviously, a greater development effort. In particular, the loop design is forcing French teams to reconsider the reactor block in its entirety, and understand the benefits, and hard points of this solution. Thoroughgoing exchanges with Japanese researchers, past masters of this technology, are already ongoing, to gain from their knowhow.

Aside from the circuits, and major components, fuel assembly handling will exert a major influence on the architecture. Prior to any handling, an interval must be allowed to elapse, for the assembly's **resi-**

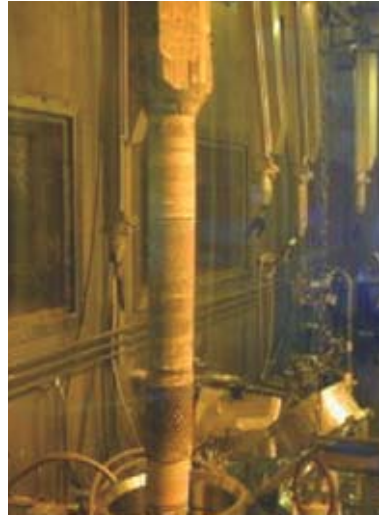
Figure 2.
Shown at (a),
the innovative
integral concept.
At (b), the advanced
loop concept.





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Development work for the sodium-cooled fast reactor line has relied on an ensemble of experimental sodium, and liquid-metal rigs. General view of the Mininaret facility, initially designed as a mockup of the wells used for the water wash of Phénix and Superphénix components such as fuel pins.



CEA

Figure 3. Foot of an irradiated Superphénix assembly, before (left), and after (right) washing. For future reactors, investigations will address improvements to washing techniques, to bring down overall campaign time.

dual power to decay sufficiently. This can take place at the core periphery, or in a separate sodium tank. Conventionally, handling of the assembly above the core is effected, as is current practice, by means of a manipulator arm, requiring actuation of two rotating plugs in the covering slab, if the core surface is to be traversed in its entirety. Use of an articulated pantograph arm would make it possible to do without one of these plugs. This is probably the solution that would prove imperative for the loop concept, owing to the small vessel size it allows.

Towards high availability

Operational feedback from Phénix and Superphénix has clearly shown that sodium handling, and the handling of the components of these reactors were fully mastered. This corresponds to technical mastery, in other words the processes are fully developed, and may be industrially deployed. However, the selfsame operational feedback also shows this technology to be relatively unwieldy, and costly to operate. Indeed, handling sodium entails systematic safety constraints, to preclude any contact with air, or humidity; and components that have been immersed in sodium require a special wash, in inert atmo-

sphere conditions. In particular, handling of irradiated fuel assemblies, e.g. for the purposes of discharging them from the core, and placing them into storage in a water pool, is a process that has a strong impact on availability, calling for improved washing techniques (see Figure 3). R&D will address washing at higher temperature, i.e. at higher assembly residual power (decay heat). Ultimately, savings in terms of overall campaign time may be anticipated.

Vouchsafing good equipment condition

This point, crucial as it is for safety, and to conserve capital investment, is peculiarly acute, with respect to the inspection of fixed structures immersed in sodium, the opacity of the liquid metal making this an arduous operation (see *Sodium-cooled reactors: towards new advances in monitoring and in-service inspection techniques*, p. 85). Teams are working on development of **ultrasound** sensors for a sodium environment, for a variety of applications, the specifications, particularly as regards measurement precision, being based on the identification, both in qualitative and quantitative terms, of the faults to be detected: assembly head displacement, remote measurement (measuring differential displacements, from one inspection to the next), long- and close-range visualization, nondestructive volume control (both large and small defects).

Accumulated experience, and innovations

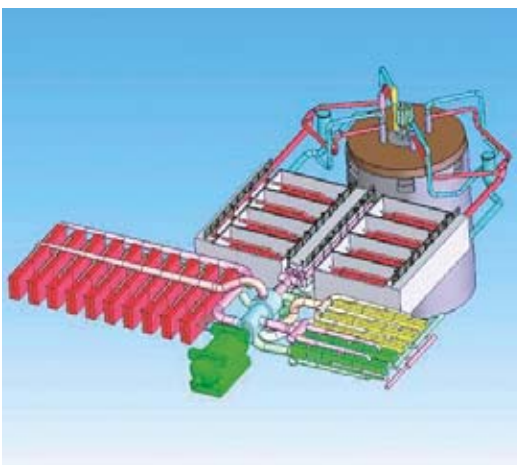
The sodium-cooled fast reactors of the future will be different from those presently known to us. While drawing on the extensive experience gained from operating the latter, in France, and around the world, these new concepts will feature significant innovations, further enhancing their safety, economics, and availability. These advanced concepts will afford a high capability to achieve savings in terms of uranium resources, and a reduction of ultimate radioactive waste.

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CEA

Modular concept for a 1,200-MWth (500-MWe) SFR using gas-turbine energy conversion, featuring an optimization of handling, heat exchanger dimensioning, and power removal systems.

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

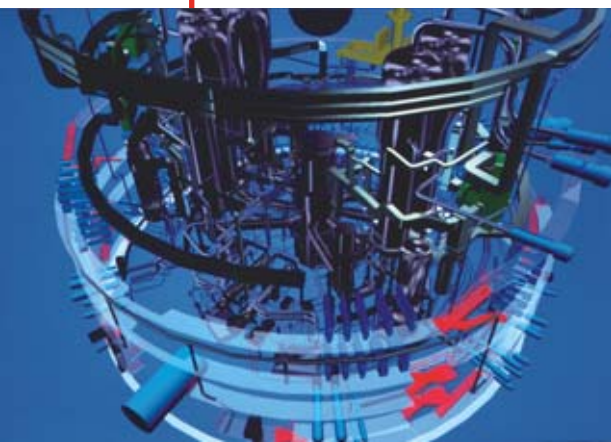
population to be regulated and, in the process, by acting on its **reactivity**, to hold reactor power at the desired level, or even to quench the chain reaction. The rods, held integral and moving as one unit (known as a **cluster**) are inserted more or less deeply into the core. Poisons, on the other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

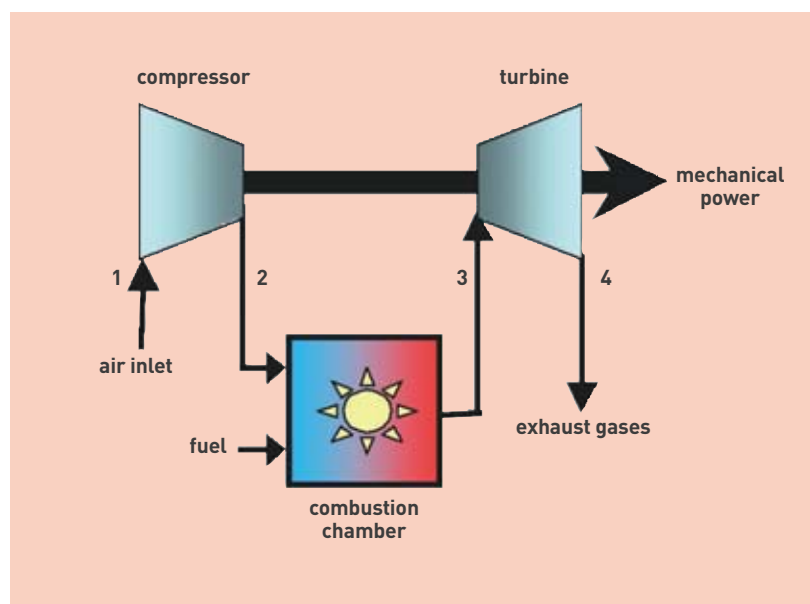


Figure.
Brayton cycle, as implemented in an open-cycle gas turbine.

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.

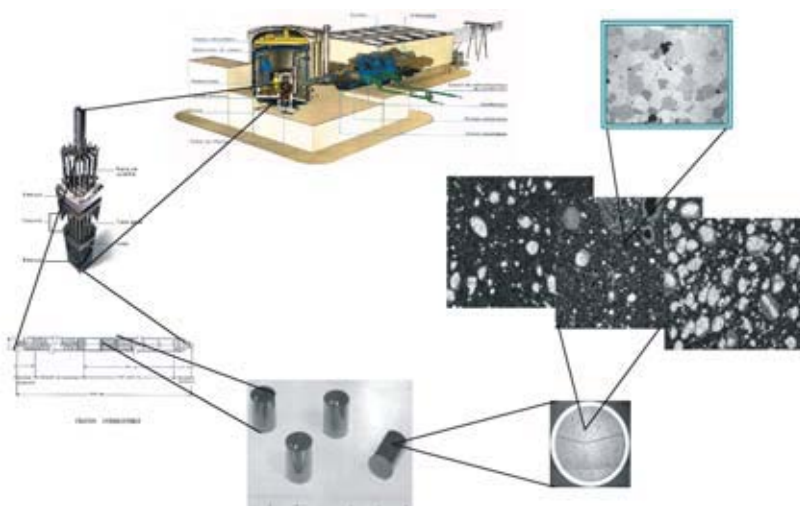


Figure.

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

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

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

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

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

Crystal structures

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

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

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

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

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

Steel families

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

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

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

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



Areva NP

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

The six concepts selected by the Gen IV Forum

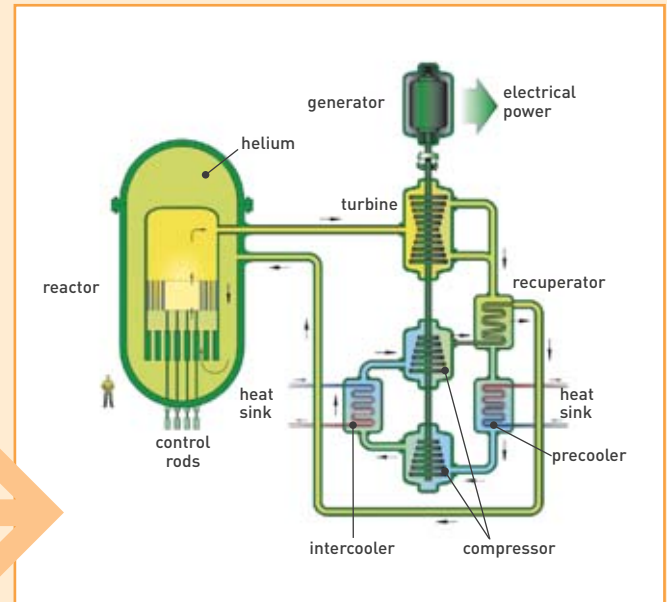
Of the six concepts selected by the **Generation IV International Forum** for their ability to meet the criteria outlined, three – and ultimately four – make use of **fast neutrons**, while three (ultimately two) use **thermal neutrons**. At the same time, two of the six concepts use gas as a coolant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

GFR

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

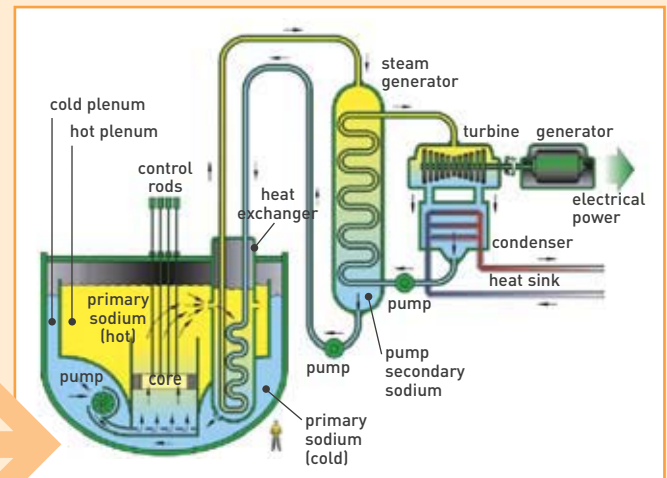
(1) GT-MHR: Gas-Turbine Modular Helium Reactor.

(2) PBMR: Pebble-Bed Modular Reactor.



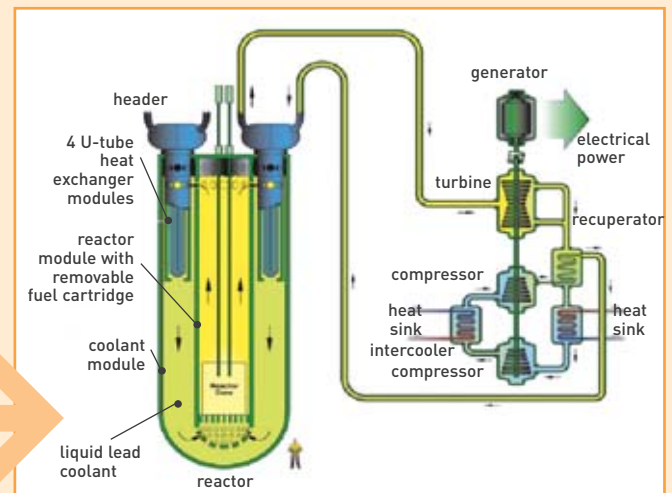
Le SFR

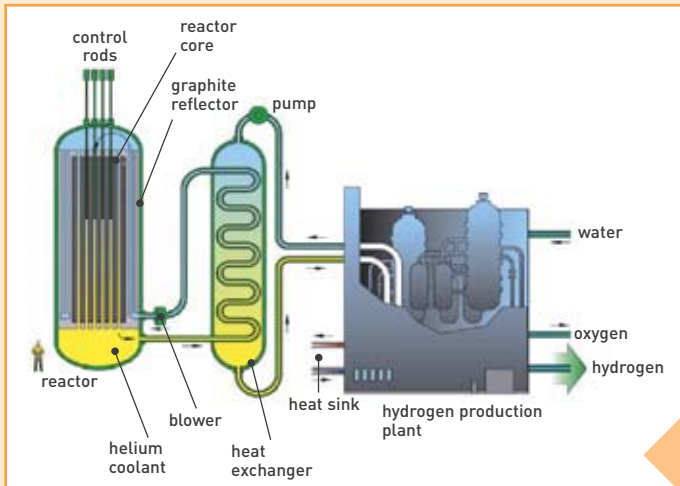
The sodium-cooled fast reactor system (**SFR**) is a liquid-**sodium**-cooled, fast-neutron reactor, associated to a **closed cycle**, allowing full actinide recycle, and **plutonium** breeding. Owing to its breeding of **fissile** material, this type of reactor may operate for highly extended periods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



LFR

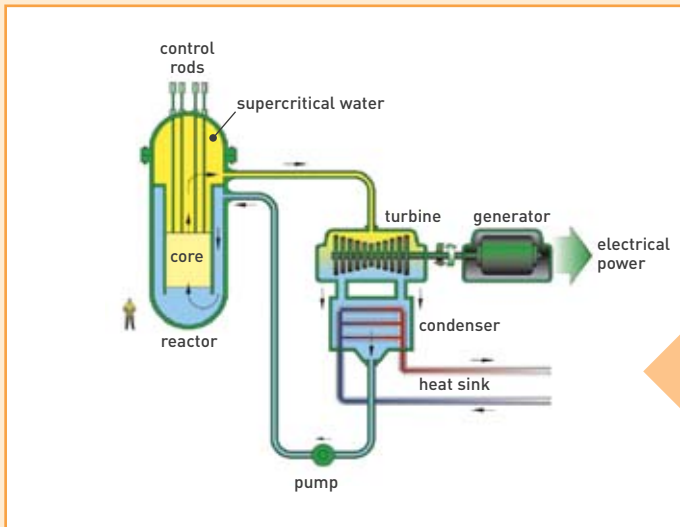
The lead-cooled fast reactor system (**LFR**) is a lead- (or lead-bismuth alloy-) cooled, fast-neutron reactor, associated to a closed fuel cycle, allowing optimum **uranium** utilization. A number of reference systems have been selected. Unit power ranges from the 50–100 MWe bracket, for so-called battery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





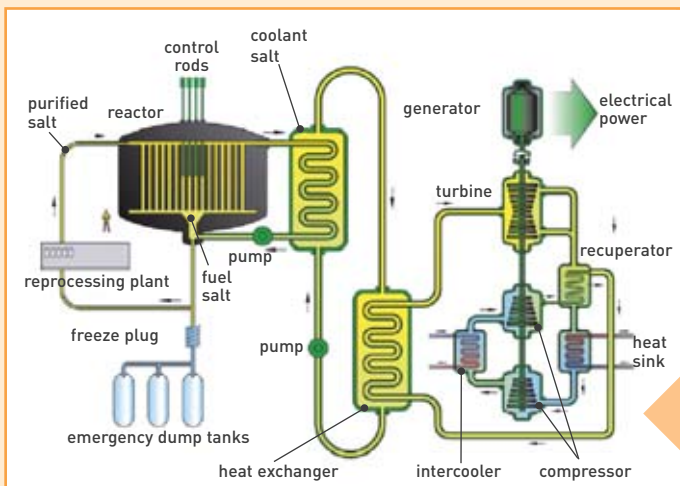
VHTR

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



SCWR

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



MSR

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