Multiscale, multiphysics simulation tools validated for **reactors**...

At the outset of the design phase for the new generation of nuclear systems, researchers are finally looking to the possibility of combining all of their knowledge, from a number of diverse disciplines, and at various spatial scales, in high-performance, multipurpose simulation tools.

Reactor design and safety rely on numerical **simulation tools** that must be validated against experiments. Simulation and validation activities have a dual character, in that they comprise both transversal activities common to all types of reactors as well as issues which are specific for each type of reactor system.

Besides performing specific experiments and developing specific simulation models, essential activities are **code** *verification* (is the code solving the mathematical model correctly?), code *validation* (does the model represent reality?), and code *qualification*, certifying that a proposed methodology satisfies all performance and safety specifications.

Code validation and qualification can be attained only by selected benchmarking, taking into account systematically all sources of *uncertainties*, computational or experimental. The development of software modules for the validation and verification of simulation programs, including global sensitivity and uncertainty analysis, is a generic issue of fundamental importance for the safe operation of all types of reactors while reducing uncertainty design margins.

Due to **neutron irradiation**, the materials in nuclear installations undergo degradation processes which must be taken into account both in the initial design and for the operation of the installations. The databases of material properties (yield stress, fracture toughness, corrosion crack propagation rate, etc.) were established during long-term irradiation programs in experimental reactors, and during mechanical and corrosion testing in specialized **hot cells**. However, the number of test reactors continues to decrease, while costs and regulatory constraints continue to increase. Therefore, experiments must be complemented by post-irradiation research programs for decommissioned reactors. Likewise, numerical modeling tools must be developed to simulate the effects of irradiation on the mechanical and corrosion behavior of nuclear materials. The aim is to reduce uncertainties and establish safety margins, as a rule in empirical manner. As depicted in Figure 1, the prediction of the combined effects of irradiation and corrosion on reactor internals and/or cladding materials requires the development and qualification of multiscale simulation tools that bridge atomic (**nanometer**), mesoscopic (micrometer), and macroscopic (centimeter) scales, along with the experimental validation of these tools (see Focus D, ***What is multiphysics, multiscale modeling?*** p. 70). Important advances remain to be achieved for that purpose, such as coupling of the combined corrosion and irradiation effects in **PWRs** and **BWRs**; and sensitivity and uncertainty analysis using combined deterministic and statistical methods. Advances must also be made in experimental validation and model qualification, using, as far as feasible, all of the industrial plant data or results from recent experimental irradiations (including non-European Union sources). And advances must be made in benchmarking and code qualification on standard problems. The goal is to contribute to the creation of a European pole of excellence in numerical simulation to improve installation lifetime prediction. Successful accomplishment of these tasks would enable the European nuclear industry to take a leading position in this area, and would help decisionmaking, regarding both existing and future reactors.

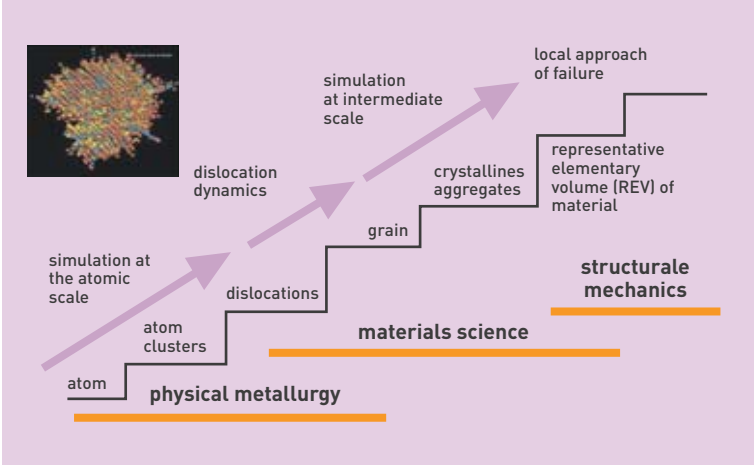

Figure 1. Multiscale modeling of a reactor-core **ferritic steel**.

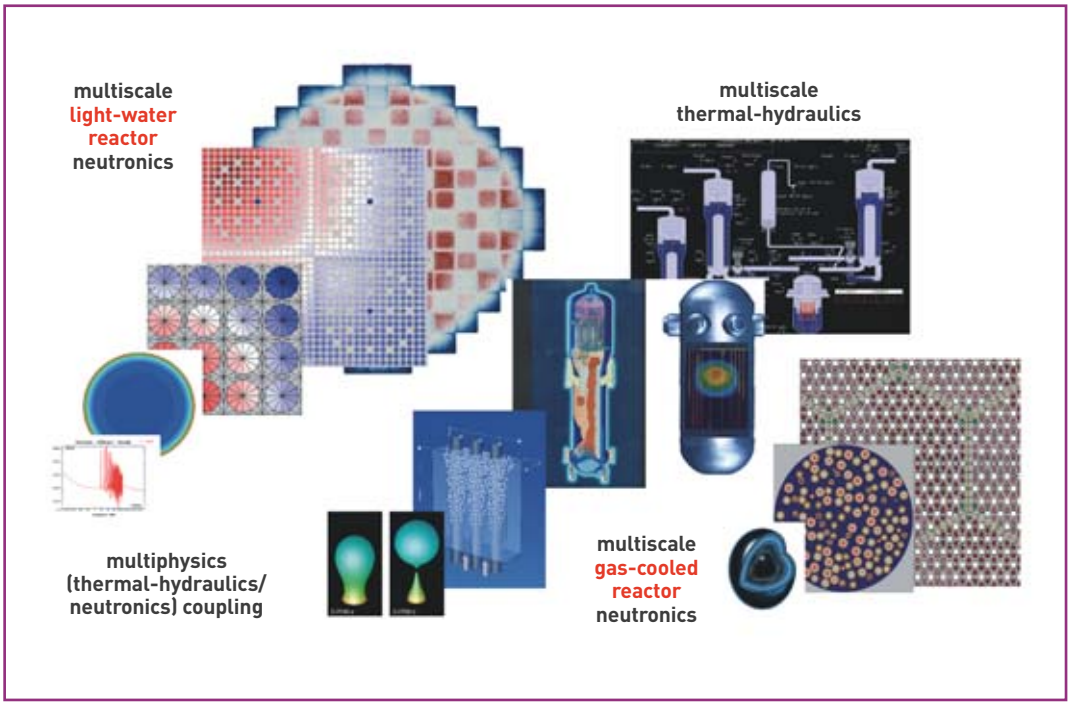

Figure 2.
Multiscale **neutronics** and thermal–hydraulics modeling.

From reactors currently operated to the systems of the future

To complement the prediction of the combined effects of irradiation, and corrosion on structures, the development, qualification, and validation of multiscale, multiphysics simulation tools must be pursued, for the purposes of reactor design and safety studies, whether it be for currently operated **PWRs**, or for reactors of the future (see Figure 2).
At CEA, the development of numerical simulation tools for reactor design, lifetime extension, and safety is facilitated by the use of the same software platform, called SALOME, which facilitates the exchange of solvers and data across sites (see Figure 3).
The continuing development and qualification of simulation tools for core physics, **thermal–hydraulics**, **fuels**, structures, and materials must also emphasize the coupling and validation of these tools, leading to the creation of a European pole of excellence in reactor safety computations for the development of the next generation of experimentally validated "best estimate" tools. Issues to be addressed in the near future include the coupling of core physics and thermal–hydraulics models for reactor safety; the addition of models related to fuel behavior (see below) and structural mechanics; sensitivity and uncertainty analysis using deterministic and statistical methods; continued development is needed to combine the strengths of deterministic and statistical methods. Concurrent, combined development of all of these approaches makes it possible to eliminate as many as possible of the limitations inherent in each of these, when taken separately.
As part of this action, a user group should test the new tools in a number of benchmark exercises with a view to their qualification. The validation and qualification of numerical simulations against experiments requires the inclusion of methods of sensitivity and uncertainty analysis together with data assimilation in the presence of both computational and experimental uncertainties.
Successful completion of these activities would enable Europe, in particular the European Union's nuclear industry, to take a leading position in the field of numerical simulation for nuclear safety evaluations and would help decisionmaking on safety issues regarding both existing and future reactors.
Closely related activities would be the development of "advanced safety assessment methodologies" for best practice guidelines for the implementation of Level-2 probabilistic safety analysis (PSA) methodologies. The aim is to achieve harmonization at the European Union level, leading to a meaningful and practical uncertainty evaluation in such an analysis. Close collaboration with nuclear regulators would be essential in order to use Level-2 PSA methodologies with greater confidence for the further development of advanced severe accident management procedures.

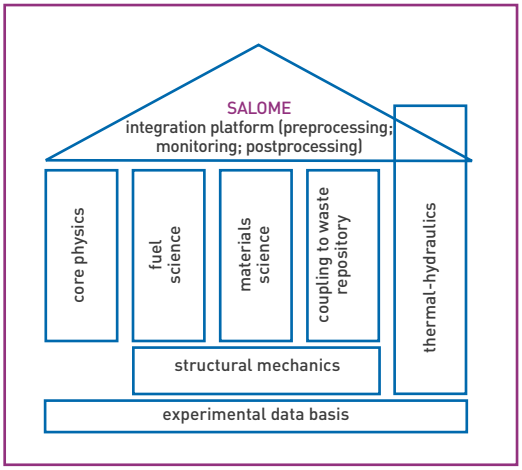

Figure 3.
Integrated approach of the SALOME platform.

...and for nuclear **fuel** behavior **under irradiation**

Modeling of nuclear **fuel** behavior is an activity which was initiated, at CEA, in the 1970s. The goal of investigations, in the early years, was to predict, to the best extent of then-available knowledge, fuel behavior for the **pressurized-water reactor** (**PWR**) and fast-neutron reactor, or **fast reactor** (**FR**) lines. At the beginning of that period, models were empirical, and global on the scale of the fuel element (PWR **rod**, FR **pin**). Such modeling became increasingly mechanistic (description of elementary physical mechanisms, at scales consistent with those of the experiment), due to the collection of experimental findings of an increasingly local character (fuel material grain, at least, this being achieved by means of metallography, microprobes, scanning electron microscopy, SIMS…), and the ever more highly predictive results achieved with numerical simulation. Nowadays, nuclear fuel research and development could not be contemplated without such finescale, detailed physics modeling, and predictive findings, underpinned by global, and local experimental findings, that are as precise as possible.

Varying the investigation framework, depending on the fuel

Aside from work for the PWR and FR reactor lines, particle fuel concepts for high-temperature reactors (**HTRs**), and fuels for experimental reactors (such as RJH) are also being investigated. While the collaborative frameworks, set up with CEA's partners – industrial (**EDF**, **Areva**), academic (**CNRS**), or international (**ANL**) – vary, for each individual reactor line, the required simulation tools, and experimental data are brought together to best advantage in the PLEIADES simulation platform, developed by CEA.

For instance, with respect to PWR fuels, the model drawn up in collaboration with industrial partners EDF and Areva is integrated into the PLEIADES platform, jointly developed with EDF, and multi-dimensional – 1-D, 2-D, and 3-D – simulation tools are developed within this computing resource framework.

As regards HTR particle fuels, the model is arrived at in partnership with Areva, with support under **FP**5 and FP6, and a design tool (ATLAS) is being developed, within PLEIADES, to evaluate performance, and the potential under irradiation, for that reactor line.

From the atom to the fuel element

Modeling the behavior of nuclear fuels under irradiation entails taking into account mechanisms from a large number of physical domains, as will be seen below. Since basic processes originate at the deepest level of matter, modeling scales needs must range ever further, away from the rod, the pellet, plate, or particle (macroscopic scale), to the **atom** (atomic scale), through **nanometer** (cavities, fission gas bubbles), and mesoscopic (fuel material grain size) scales (see Figure 4).

The core of **fuel element** simulation involves modeling the *thermal* and *mechanical* behavior of these

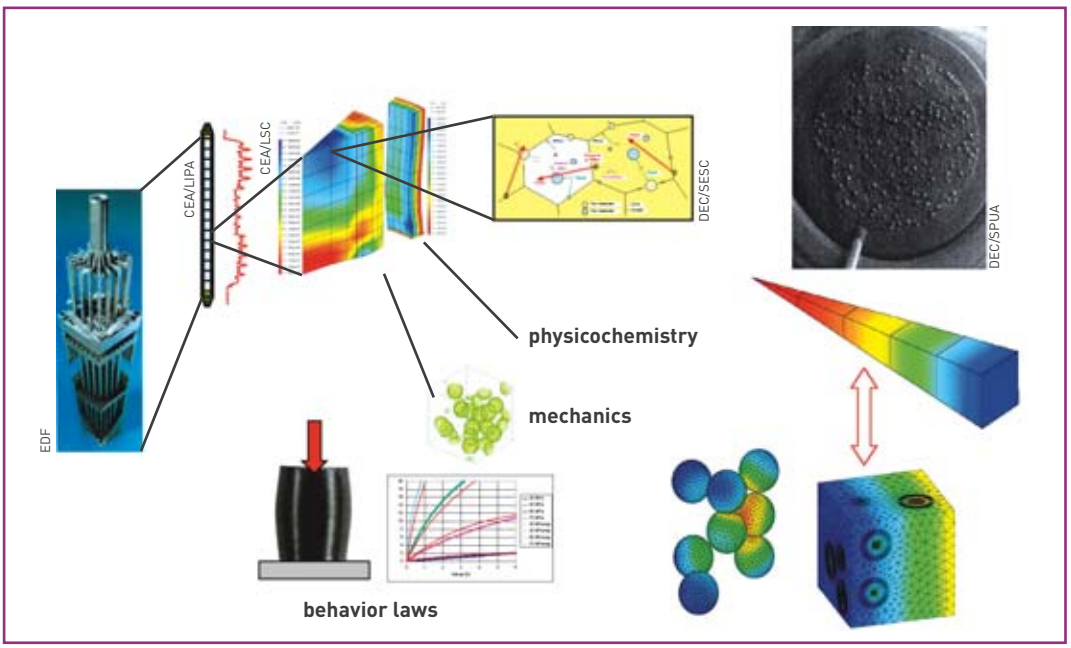

Figure 4. Homogenization/localization techniques are being developed to investigate fuels for a variety of reactor concepts (generation-II, -III, and -IV PWRs and GFRs). Taking into account the mechanical aspects, coupled to processes of physicochemical origin, requires development of multiphysics, multiscale numerical tools, validated through in-reactor and laboratory tests.

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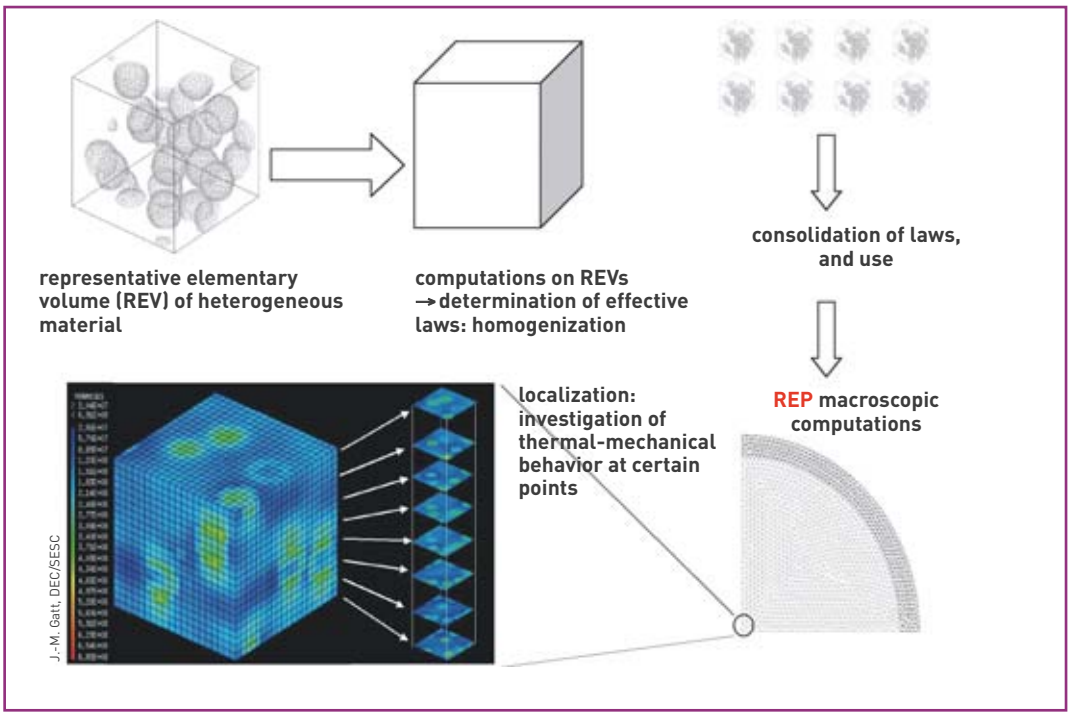

Figure 5.
Homogenization methods are being developed, to build up macroscopic behavior laws, in the domains of mechanics, and thermics. These laws, introduced into simulation tools, allow global fuel element behavior to be evaluated. For the subsequent evaluation of local stress and temperature fields, a localization technique, derived from the homogenization approach, is used.

objects, in order to compute temperatures, stresses, and deformations, at different points in the fuel material and cladding, in a variety of situations, such as basic irradiation (nominal conditions), or postulated accident situations. The aim, in the PWR case, is to compute the resistance exhibited by the first containment barrier (**cladding** tube), and thus determine operating limits, or paths for improvement, allowing performance to be enhanced. As regards other novel concepts, the aim will be to design an element having the capability to meet the specifications drawn up by the designer of the reactor involved (HTR, materials testing reactor, FR).

Such modeling involves, on the one hand, the use of thermal, and mechanical behavior laws, which in itself is a major scientific challenge to be met, considering the complex changes (steep thermal and **stress** gradients, and **transients**), but equally the multidimensional (1-D, 2-D, or 3-D) schematizations, involving fracturing and cracking of the material, for which behavior is to be simulated. At the same time, it becomes necessary to implement *homogenization techniques*, in order to take into account the highly heterogeneous character of this material, and the range of scales involved (from the mesoscopic to the macroscopic scale) (see Figure 5).

Setting the boundary conditions for such thermal–mechanical computations entails use of a **thermal–hydraulic** (or gas thermal–hydraulic) *model*, to allow heat exchanges to be computed, between the fuel element, and the coolant fluid, and consequently cladding temperature (cladding tube, for PWRs, or **boiling-water reactors** [**BWRs**]).

For similar reasons, in the PWR case, for instance, as the power generated within the pellet exhibits radial, and axial profiles, it becomes necessary to implant into simulation tools (e.g. METEOR, for PWRs) a *neutronic model*, in order to compute, for every point in the fuel material, the power generated during irradiation, and its variation over time. At this point, it must be stressed that two paths are available, to arrive at these boundary conditions: the first one uses simplified neutronics and thermal–hydraulics models, allowing self-standing calculations to be carried out; the second path involves couplings (linkage…) with dedicated **codes** (e.g. APOLLO, for neutronics; CATHARE for thermal–hydraulics).

The thermal and mechanical properties of fuel materials, and claddings vary in the course of irradiation, as materials sustain *irradiation-induced damage*, and solid, or gaseous **fission product**s are formed. Modeling must thus draw on *solid-state physics*, to simulate the formation, and evolution of point defects, and *thermodynamics*, to determine the physical–chemical states of the fission products generated.

Fission product behavior

The behavior of gaseous fission products is one of the main rubrics of fuel modeling. Indeed, some species play a major role in cladding material *corrosion* (iodine, in the PWR case; tellurium for fast reactors; or silver, in HTRs). At the same time, very particular attention is paid to the behavior of **fission gases**, since these contribute, through **swelling** effects, and release of such gases into free spaces, to the loading experienced by the cladding material, in nominal conditions, or accident situations.

Modeling of fission gas behavior is the major exemplar of multiscale modeling, in terms both of numerical simulation, and the physical approach involved. Such modeling must moreover necessarily be grounded on *multiscale, integral and separate-effect*


EDF

Pressurized-water reactor fuel assembly, with control cluster inserted

characterizations, and experiments, as will be seen from the example of PWRs.

Pressure inside a rod (4 **meters** in length) is determined through computation of gas release, for every point in the fuel material. For that purpose, the rod is partitioned into a finite number of cells, in **numerical simulation** tools (METEOR, or PLEIADES).

Before they can reach the edge of the pellet (about 1 **centimeter** in diameter), the gas's atoms must migrate within the grain (dimension: 10 **micrometers**), moving from an intragranular to an intergranular location, in order to arrive at the **grain boundary**.

Rare gases (xenon, krypton) are nor soluble in the **crystal** lattice, and, in the course of their migration, precipitate into bubbles. The size of such bubbles, typically, is of the order of a **nanometer**, and they stand in a dynamic equilibrium: they grow through the effects of vacancy and atom migration; they may be destroyed by fission spikes. At high temperature, and high **burnup**, the size of these bubbles increases by coalescence, and can reach the micrometer scale.

Two predictive simulation approaches

In order to achieve predictive simulation at these diverse scales, two approaches are currently implemented. One is based on exploiting macroscopic, and microscopic findings from R&D programs, as a rule supported by CEA's industrial partners, Areva and EDF; these consists, firstly, in global measurements (volume of gas released, diametral profiles of PWR rods), and observations at the micrometer scale, carried out subsequent to irradiation (using metallography, microprobes, SEM, SIMS). These data, of great importance, allow advances to be made, as regards understanding of the fuel element's global behavior, however they do not allow mechanisms to be quantified at the nanometer scale. Such global findings do not make it possi-

How is multiphysics, multiscale modeling carried out?

In reactor safety, the analysis and **numerical simulation** of **transient** scenarios requires multiphysics simulation tools which couple core **neutron** kinetics , as a function of time, with plant **thermal–hydraulics**, since changes in fuel temperature, **moderator** density, neutron **poison** concentration, and **control rod** position all affect the neutron macroscopic **cross-sections**. Originally, such multiphysics simulation tools were provided only with models that were independent of the direction in space, called point kinetics models. In other words, the spatial scale of *point kinetics models* is the entire reactor **core**. However, the stronger the cross-section variations, the less accurate the point kinetics model is. Hence, models are needed that take into account, in more precise detail, core geometry, to avoid unacceptably large errors in the computation of the neutron flux. Thus, to extend the applicability and improve the accuracy of coupled neutronics/thermal–hydraulics simulation tools, the point kinetics models were supplemented with at first one-dimensional, then three-dimensional neutron kinetics models. For the one-dimensional neutron kinetics models, the spatial scales involve radially and azimuthally averaged "slices" of the reactor core, while the three-dimensional models involve the smallest spatial scales, at the level of individual fuel **assemblies**. Incorporating three-dimensional neutron kinetics models, though, brings about a very high computational overhead that prohibits their routine use. On the other hand, most of the postulated accident scenarios are characterized by relatively large time intervals during which a point kinetics or a one-dimensional model would suffice. Thus, an efficient multiphysics, multiscale simulation tool would activate the three-dimensional model only when called for by the physical processes occurring in the respective portions of the transient. At all other times, it would use a lower-dimensional model. This goal would be well worth pursuing by all future-generation multiscale (and multiphysics) simulation tools, since their efficiency would be tremendously increased by including an automatic, dimensionally adaptive, **algorithm** which switches by itself, using the underlying physical processes, between three-, one-, and zero-dimensional models. Such a setup is illustrated in Figures 1–3.

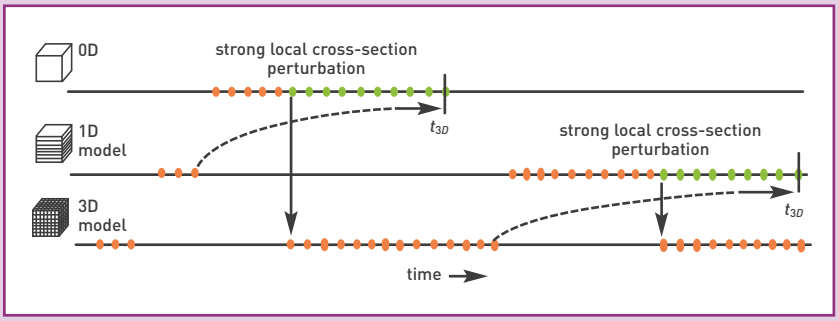

Figure 1.
Automatic switching between neutron kinetics models of varying numbers of dimensions.

Application to a control rod ejection

The rapid ejection of a **control rod** is accompanied by very strong local perturbations of the neutron flux shape. Following the ejection, the power increase in the core induces a global, nonuniform increase in fuel temperature that in turn changes the cross-sections through the **Doppler effect**. These features were studied through a set of benchmark problems released by the **NEA** (**OECD**) Nuclear Science Committee (NSC). They serve as an appropriate test bed for the dimensionally adaptive algorithm described above. The specific problem depicted in Figures 2 and 3 below prescribes the ejection of a fully inserted central control assembly from a **critical** reactor core at hot zero-power conditions. Staring from a steady-state solution (generated by a numerical simulation carried up to 100 seconds or so), the central control assembly is ejected from the core between 101.0 and 101.1 seconds. The simulation is continued to 111.0 seconds, when the state variables reach the asymptotic state.

The total core power and maximum nodal power peaking factor for calculations using solely the 3 D, 1-D and 0-D (point kinetics) models, respectively, are compared in Figure 2 with results from the dimensionally adaptive calculation.

ble, either, to discriminate between the basic mechanisms involved, e.g. in the destruction of gas bubbles by fission spikes, or to arrive at actual diffusion coefficients. The other approach relies on using the knowledge and data gained through the fundamental research carried out on the fuels considered, allowing – through separate-effect experiments, in particular (such as e.g. ion irradiation of plates transparent to UO_2 ions) – the basic solid-state physics data to be arrived at, such as location of the gas, in atomic form, in the crystal lattice (***ab-initio*** computations), migration energies, diffusion coefficients (heat treatments, and ion irradiations), or the healing of irradiation damage.
It is nowadays apparent that, if advances are to be made, in our understanding, and as regards the predictive character of numerical simulations, the behavior of fuel materials must be modeled at scales ranging from the meter, to the atomic scale. Two major scientific challenges currently arise. The first one is that of gaining knowledge of processes at scales in the 1–100 nm range, knowledge that is lacking, at the present time, owing to the dearth of good-quality experimental findings at such scales; the second is that of systematically working out the relationships prevailing between the various scales involved in simulation.
Such major advances will only become feasible once numerical methods, and, most crucially, the performance of computing resources available reach processing speeds much higher than current levels (i.e., higher than the petaflops(1)), and storage capacities allow effective linkage of all of the scales simulated.

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(1) Petaflops = 10^{15} floating operations per second.

It will be seen that the algorithm models the transient quite well. By contrast, the calculations which use only the one-dimensional, and point kinetics models do not predict a prompt power excursion.
The model dimensions chosen by the adaptive algorithm are also shown. It will be seen that first the 1 D and then the point kinetics model are activated at the beginning, during the stationary part of the calculation. The 3-D model is reactivated when the rod is ejected. The algorithm switches back to the lower-dimensional models once the rod is out of the core, but makes frequent returns to the 3 D model until the transient reaches an asymptotic state, when the point kinetics model is activated during most of the remaining time steps.
Figure 3 shows the core power, and the model dimension of the adaptive calculation between 101.0 seconds and 102.0 seconds. When the rod ejection begins, the dimensionally adaptive algorithm uses the point kinetics model. Once the 3-D model is reactivated, the rate of change of the flux shape function is so large that the lower-dimensional models are not reactivated until after the control rod has stopped moving. For the next 0.9 seconds, the lower-dimensional models are activated with brief returns to the 3-D model when the error estimate is too large. It should be noted that the dimensionally adaptive algorithm yields a 70% saving in computation time when compared to the 3-D reference calculation, while preserving an equivalent accuracy.

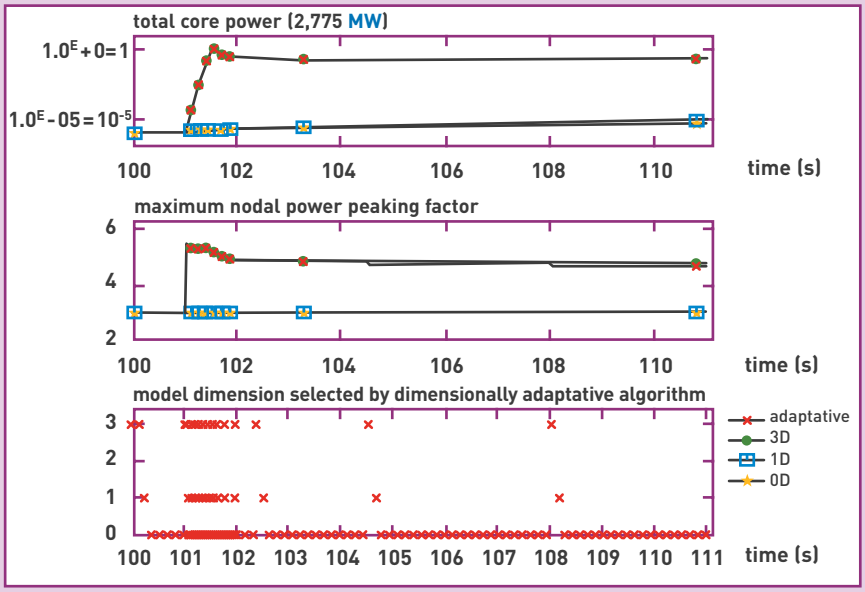


Figure 2. Comparison of computation results for a control rod ejection.

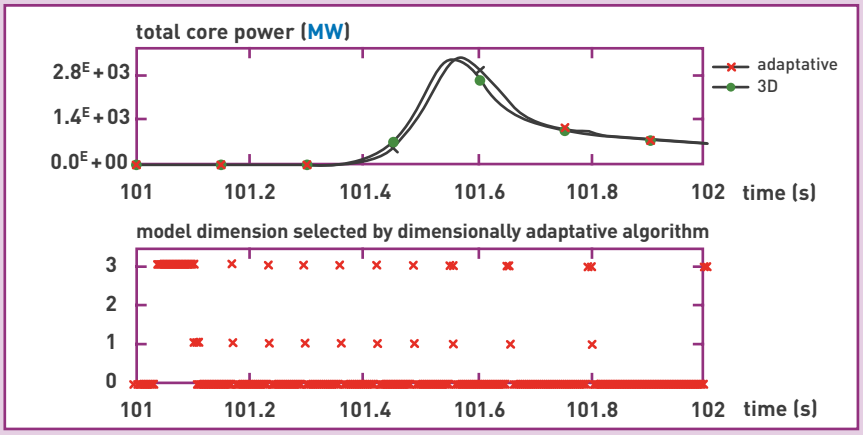


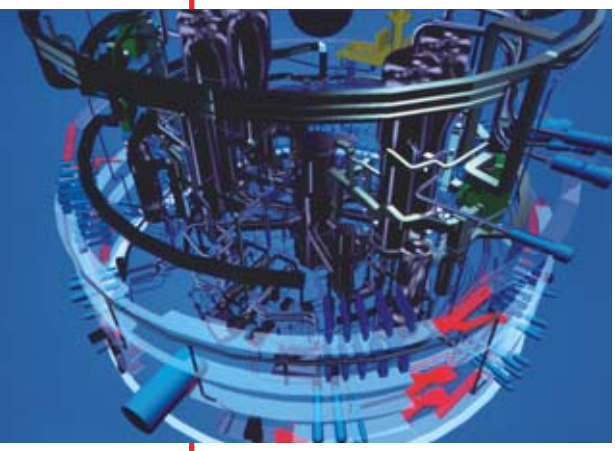
Figure 3. Results for the adaptive algorithm, for times in the t = 101 s to t = 102 s interval.

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

M. Brigaud/ EDF Médiathèque

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

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

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

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

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

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

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

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

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.

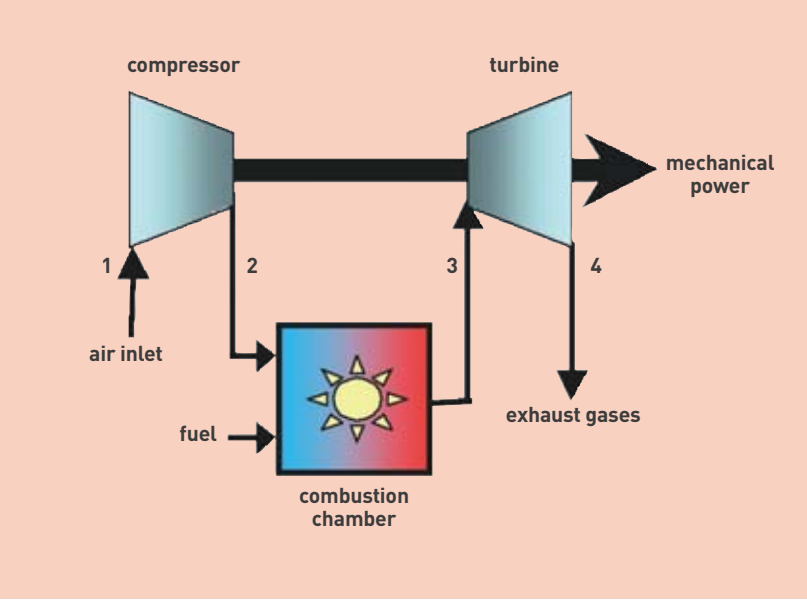


Figure.
Brayton cycle, as implemented in an open-cycle gas turbine.

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 → 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.

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.

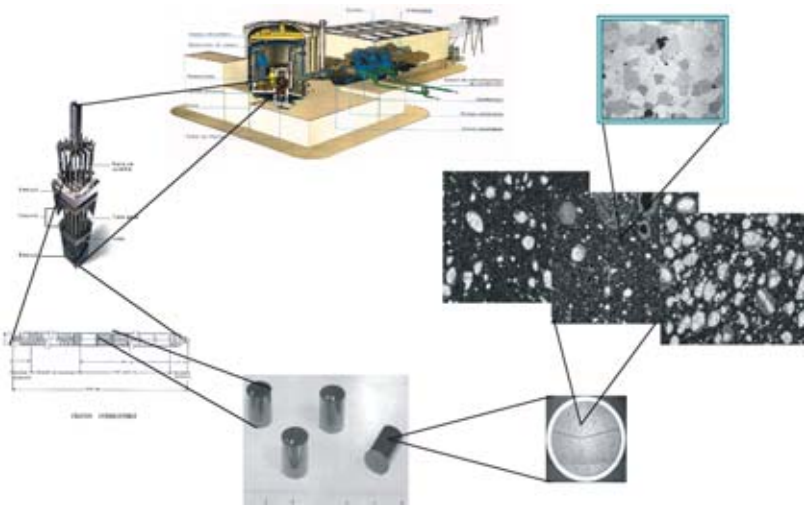


Figure.
Improving nuclear fuel reliability, and cost-effectiveness calls for finescale modeling of that fuel, through a multiscale approach, from reactor to fuel microstructure (in this instance, MOX fuel). Microstructural characteristics (porosity, cluster size and distribution, grain size...) have 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 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.

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.

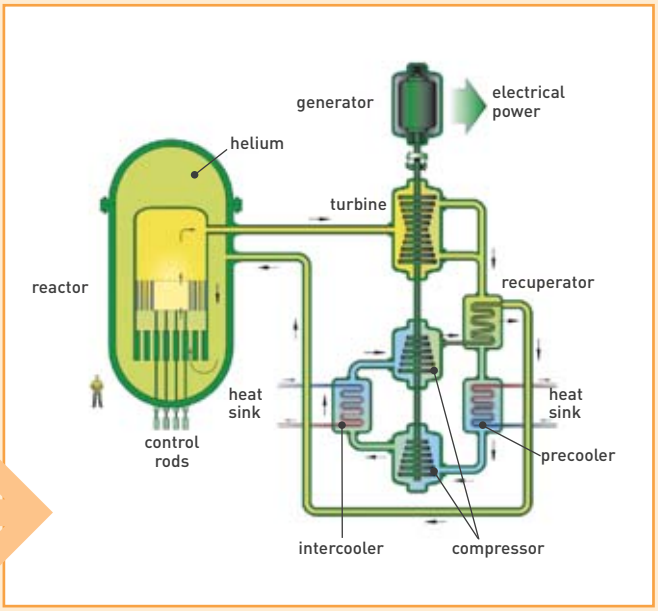
The six concepts selected by the Gen IV Forum

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

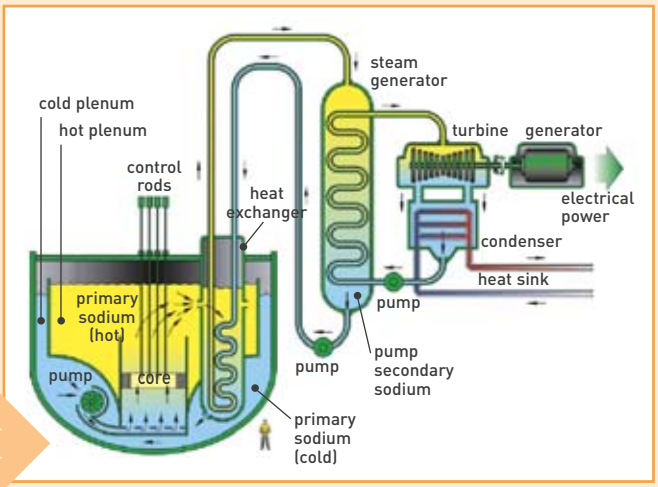
The gas-cooled fast reactor system (GFR) is a high-temperature, gas-cooled (helium-cooled as a rule), fast-neutron reactor allowing actinide recycle (homogeneous, or heterogeneous), while sustaining a breeding capability greater than unity. The reference concept is a helium-cooled, direct- or indirect-cycle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



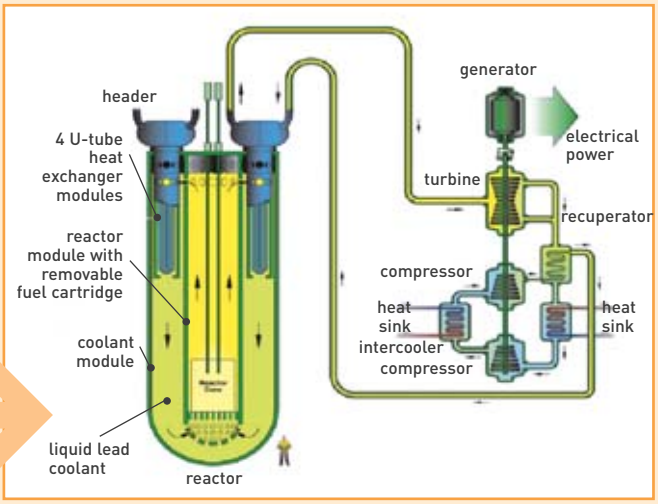
Le SFR

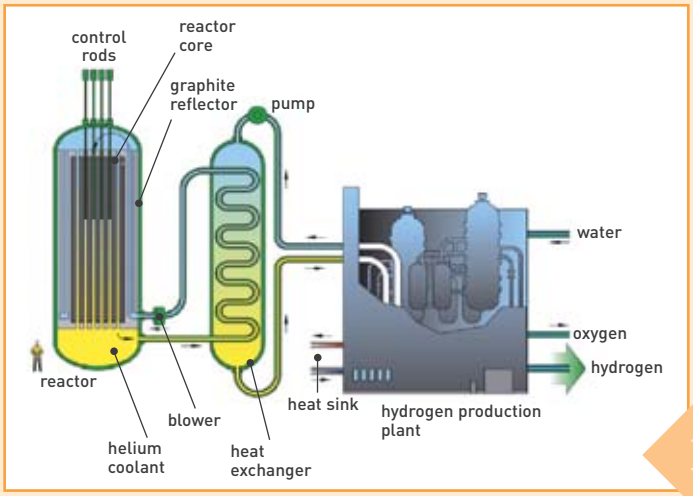
The sodium-cooled fast reactor system (SFR) is a liquid-sodium-cooled, fast-neutron reactor, associated to a closed cycle, allowing full actinide recycle, and plutonium breeding. Owing to its breeding of fissile material, this type of reactor may operate for highly extended periods without requiring 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

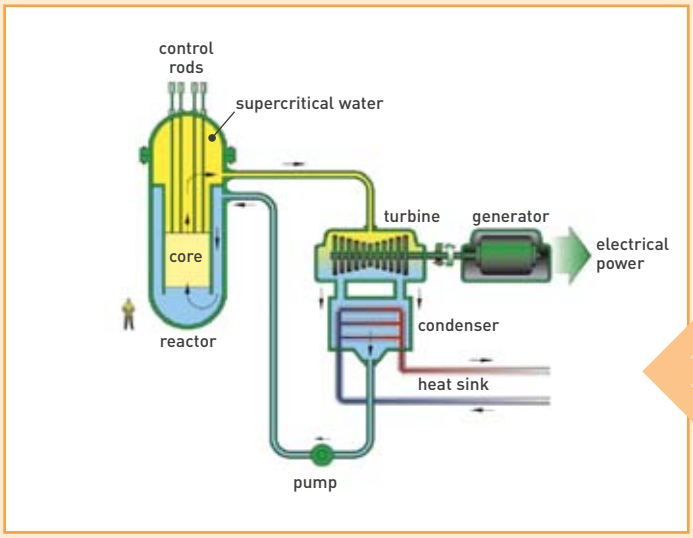
The lead-cooled fast reactor system (LFR) is a lead- (or lead–bismuth alloy-) cooled, fast-neutron reactor, associated to a closed fuel cycle, allowing optimum uranium utilization. A number of reference systems have been selected. Unit power ranges from the 50–100 MWe bracket, for so-called battery concepts,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





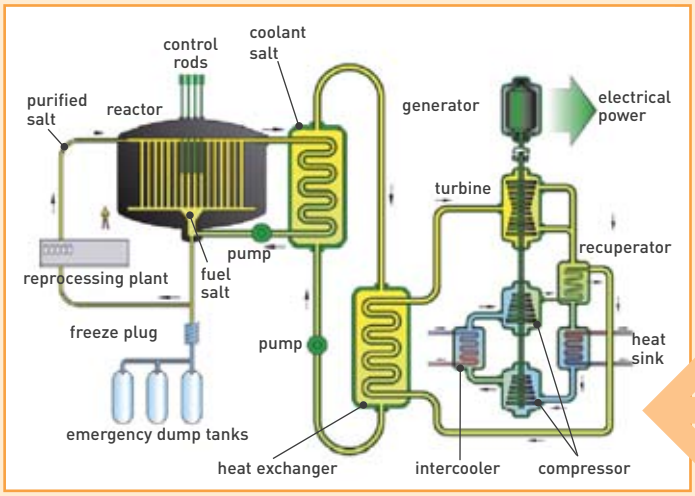
VHTR

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.