Metallic materials, one of the **keys** for the fourth generation

Metallic materials are set to play a crucial part in the development of fourth-generation nuclear systems. Whether it be for fuel cladding, vessel construction, or the fabrication of other structural components for gas-cooled, high-temperature and/or fast-spectrum reactors, innovations are called for.

P.Stroppa/CEA

Setting up the static traction apparatus used to carry out traction or compression tests on irradiated materials, due to form part of the equipment of a mechanical test cell (viewed from the rear area), at the Irradiated Fuels Investigation Laboratory at CEA/Saclay, under the aegis of the PELECI program (Projet d'équipement du Laboratoire d'étude des combustibles irradiés: Equipment Program for the Irradiated Fuels Investigation Laboratory), which has led to the coming on stream, in 2005, of a new line of shielded cells.

Under the aegis of programs concerned with the systems of the future, numerous investigations are being carried out to develop novel materials, combining at the same time satisfactory mechanical strength in hot conditions, good **irradiation** resistance in a **fast spectrum**, while providing all required guarantees in terms of safety, and security for the reactor (see Focus E, ***The main families of nuclear materials***, p. 76).

ODSs for the hotter, more highly irradiated structures

Of the solutions that may be considered for **fuel cladding**, in particular, metallic materials are those for which the widest operational feedback is available. **Zirconium alloys** are currently employed in **pressurized-water reactors** (**PWRs**), and **austenitic steels** in **fast reactors** (**FRs**). These materials do however carry a number of drawbacks: the former may not be used beyond 400 °C, while use of the latter is not to be contemplated at high **doses**, owing to their **swelling** under irradiation (see Figure 1).

There is one other major steel class for which behavior under irradiation is well known: **ferritic–martensitic steels**. These afford many advantages, with respect to fuel element cladding. Indeed, as may likewise be seen from Figure 1, they exhibit outstanding dimensional stability under irradiation, good corrosion resistance in a variety of environments, and, when strengthened by a **nanometric** oxide dispersion, their mechanical strength is greatly enhanced, even at very high temperature (1,000–1,100 °C). Materials of the latter type are commonly known as oxide-dispersion-strengthened (ODS) materials.

One property currently acting as a limitation to the use of ferritic–martensitic steels of the Fe–9/18Cr type (i.e. iron-base alloys, with 9–18% chromium [Cr], and a few percent additional elements) is their **creep** resistance. To limit creep deformation, and obtain very-high-performance materials, researchers and manu-

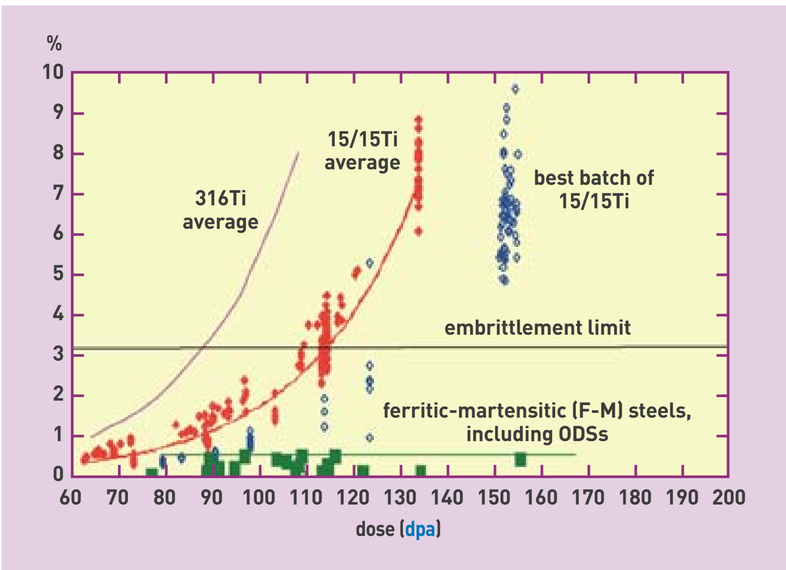

Figure 1.
Dimensional evolution, as a function of the dose found on metallic cladding structures in fuel assemblies used in the core of the Phénix fast reactor. Austenitic steels of the 316Ti type have seen widespread employment, in the past, as cladding and **hexagonal tube** (HT) material. For cladding, they have been supplanted by an austenitic variant, 15/15Ti, exhibiting slightly greater **swelling** resistance, and, for HTs, by EM10, a 9% Cr martensitic steel (from J. L. Séran).

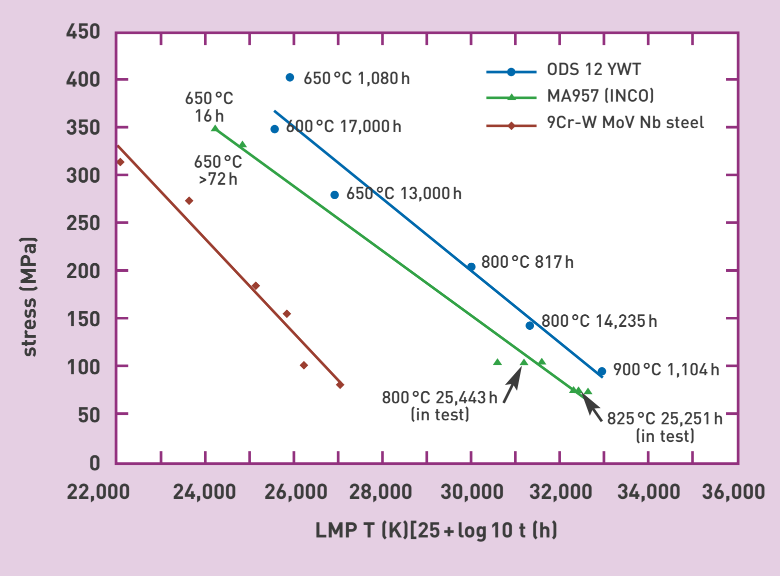

Figure 2.
Creep rupture stress, as a function of time and temperature, for two ODSs and one conventional martensitic steel. These curves were obtained by **ORNL**, using in particular the MA957 grade supplied by CEA, under the aegis of an **INERI** collaboration (with thanks to D. T. Hoelzer, from ORNL).

facturers have succeeded in reinforcing metallic matrices with finely dispersed oxides, highly stable at high temperature.
Illustrating this point, Figure 2 shows a comparison of the creep properties for two ODSs – 12YWT steel (Fe–13Cr 3W Ti Y_2O_3), and MA957 steel (Fe–14Cr 0.3Mo Ti Y_2O_3) – with those of a conventional Fe–9Cr W Mo V Nb steel. The graph, of the Larson–Miller type, chosen for this comparison allows computation of a so-called LMP parameter, which depends on both time and temperature, and allows creep rupture stress to be ascertained. This graph clearly shows the advantage afforded by ODSs, compared to a conventional steel grade. It will be seen, for instance, that, creep rupture stress after 17,000 hours at 600 °C (corresponding to an LMP value of 26,391), for the ODSs lies in the 300–350 **MPa** range as against 150 MPa for a conventional grade. In ODSs, the fine oxides block the displacement of **dislocations**, which are responsible for the material's deformation.
It was in the 1970s that the first commercial ODSs appeared, the grades offered being nickel or iron based. Conventional metal making techniques, involving ingot melting and casting do not allow a material with a dispersion of fine oxides to be obtained. When the oxides are introduced during the fusion step, they either react with the furnace crucible, and are no longer to be found after casting, or they coalesce(1) within the liquid bath, forming oxides that are too large to result in a reinforcement effect. It is thus necessary to go for mechanical synthesis, a process taken from powder metallurgy, and commonly known as mechanical alloying.

Towards controlled nanoprecipitation to limit high-temperature creep

Figure 3 sets out the schematic for iron-base ODS production. Metal, or prealloyed powders, a few tens of **micrometers** in diameter, are mixed with a fine yttrium oxide powder, with particle sizes ranging from about 10 nanometers to around 10 micrometers, oxide volume accounting for some 1% of total material volume. These powders are then co-ground, through a succession of mechanical shocks, inside a mill holding grinding balls of various sizes. During this delicate step, which, to a large extent, determines material ultimate quality, the powders are successively work hardened,(2) welded together, fractured again, welded again… The oxides may become **amorphized**, ending up, after grinding over several tens of hours, in a state where they are as dissolved in the metal powder. The ground powder is then recovered, and placed in a mild steel container, prior to carrying out hot extrusion forming,(3) or **hot isostatic pressing**. This operation allows consolidation of the material to take place, i.e. it makes it possible to obtain, due to the pressure and temperature involved, a material that is dense, free of porosity, and exhibiting a density equivalent to that of a fusion-obtained material.
During material consolidation, the dissolved oxides may precipitate again, forming a fine, homogeneous dispersion in the matrix. Depending on consolidation parameters, quality of grinding, and powder chemical composition, part of the precipitates will be of nanometer size.

(1) Coalescence: the process whereby two identical, but separate objects (e.g. two droplets) tend to merge together.

(2) Work hardening: the plastic deformation of a metal, at a temperature lower than its recrystallization temperature. This operation results in increased hardness characteristics, and reduced ductility characteristics.

(3) Extrusion: a method used to shape metals, made ductile through hot forging, by passing them through a die.

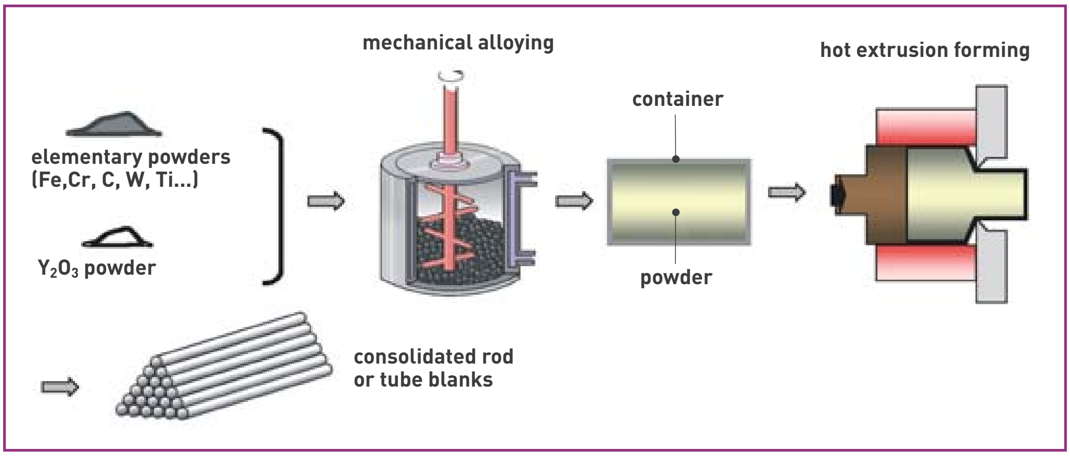

Figure 3.
Principle schematic of ODS alloy fabrication.

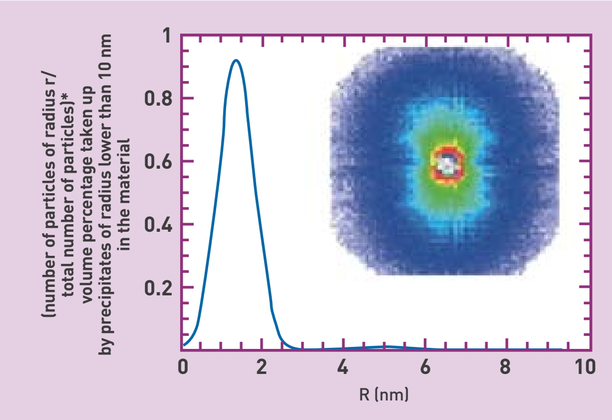

Figure 4.
Normalized distribution of nanocluster radii (i.e. of the radii of clusters of atoms of a metal, forming nanoparticles) in 12YWT steel (Fe–12Cr 3W Ti Y_2O_3), obtained by small-angle neutron scattering (SANS), at the Léon-Brillouin Laboratory, Saclay (France) (from Marie-Hélène Mathon *et al.*). Inset, the raw 2-D scattered intensity spectrum, as obtained at the SANS detector.

Figure 4, obtained by small-angle neutron scattering[(4)] in a 12YWT alloy (Fe–12Cr 3W Ti Y_2O_3), shows a distribution of sizes smaller than 10 nm, for the nanoclusters liable to be formed during consolidation. These observations have been corroborated, by measurements using tomographic atom probes,[(5)] or transmission electron microscopy (see Figure 5). The crystallography, and chemical composition of the nano-oxides formed during consolidation may be quite different from those of the initial oxides. Exhibiting an average radius close to 1 nm, and high thermal stability, the oxide phases coalesce at temperatures higher than 1,000 °C, such reinforcements forming highly effective barriers to material deformation. It should be noted that other, much larger oxides are also present in the material, after consolidation. These oxides, which may be of micron dimension, appear to play no particular role with respect to creep resistance.

Depending on the type of reactor considered, and the stresses involved, iron-base ODSs exhibiting a ferritic structure may be contemplated for nominal reactor operating temperatures higher than 1,000 °C. This advantage, together with an outstanding swelling resistance, means that ODSs currently stand as one of the best candidate materials, to achieve very significantly enhanced **SFR** fuel element performance.

The melting temperature for iron-base ODSs lies at slightly more than 1,400 °C: these may not, therefore, stand as substitutes for refractory metals, or ceramics. Employing ODSs for fuel cladding, in reactors such as **GFRs**, may only be contemplated provided temperature in accident conditions can be restricted to around 1,500 °C, over a few hours.

(4) Small-angle neutron scattering (SANS): a technique for the exploration of solid-state matter, based on X-ray (SAXS) or neutron scattering, at angles that may be as low as 0.005°. This allows the investigation of heterogeneities having sizes ranging from a few angstroms to several billion angstroms.

(5) Tomographic atom probe: a technique allowing e.g. the spatial configuration to be arrived at, of the various constituent atoms of an alloy, by stripping, one at a time, ions from a metal tip cut in this alloy, accelerating them in an electric field, and measuring flight time to a detector.

Oxide-dispersion-strengthened materials may involve a number of drawbacks, which should be properly appraised. Owing to the fabrication technique involved, they exhibit a marked microstructural **anisotropy**, resulting in anisotropy with respect to mechanical properties. For instance, to take the case of a cladding tube, **grains** will be stretched in the direction along the tube axis, inducing lower mechanical strength in the radial, and tangential directions, which correspond to the major stress modes, in operating conditions. This issue will require to properly control all the industrial fabrication product range, to minimize this characteristic.

These materials suffer from poor weldability. If their mechanical properties are to be preserved, after welding, the size and distribution of oxide nanoreinforcements must remain unchanged. Only a few welding processes (inertial friction welding, diffusion welding…) are able to meet this constraint.

Grades presenting a low Cr content (lower than 12%), exhibiting a martensitic structure have a limited potential for high-temperature use, and may raise issues of physical–chemical compatibility with the environment – be it on the outside (sodium), or the inside (fuel) – of the **fissile**-material cladding.

As for high-Cr ferritic grades, as envisaged above, experiments must be carried out, to verify neutron irradiation, at temperatures higher than 400 °C, does not cause unacceptable embrittlement.

All of these points are addressed in the context of the major R&D program being conducted by CEA. Indeed, the fabrication and assembly processes must be totally controlled to obtain the correct material microstructure, on which the aging mechanisms depend, both out and under irradiation. In 2007, a number of new ODS grades have been produced, in collaboration with the Aubert & Duval, and Plansee companies. These grades are due to be irradiated in Phénix (MATRIX 2 irradiation), to investigate their microstructural evolution, and mechanical behavior, in the 400–550 °C temperature range.

A number of R&D programs are currently being submitted in the European context (**FP7**), and at the national level, in France (**ANR** project). A wide-ranging international collaboration has also been set up, with regard to ODSs, bringing in, in particular, Japanese and US teams, to meet the challenge set by qualification of ODS alloys, as solutions for fast-reactor fuel cladding.

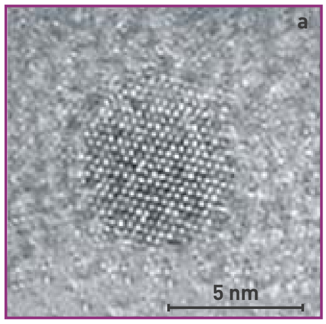

Figure 5.
High-resolution transmission electron microscopy image of a $Y_2Ti_2O_7$ particle, in a Fe–9Cr W ODS (From Klimiankou *et al.*).

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A batch of samples of materials intended for nuclear systems of the future, readied for the MATRIX experimental irradiation program, in the Phénix reactor.

Chromium ferritic–martensitic steels: materials with many applications

Ferritic–martensitic steels (F–M steels, for short) with 9–12% of chromium content are materials affording many advantages (see Focus E, ***The main families of nuclear materials***, p. 76). These are stainless steels, exhibiting high mechanical strength, and combining low thermal expansion with high **heat conductivity**. At the same time, their creep resistance at high temperature has been enhanced over many years of research activity, for instance through addition of such elements as niobium (Nb), and vanadium (V) – these forming a fine carbide precipitation – and addition of molybdenum (Mo) and tungsten (W) in solid solution, these inducing strengthening of the material.
Such steels, as e.g. industrial grades T91 (9Cr 1Mo V Nb), T92 (9Cr 0.5 Mo 2W V Nb), or T122 (12Cr 0.5 Mo 2W V Nb Cu), are being considered as **primary** and **secondary circuit** materials for the future SFR, and for steam generator tubes, as a substitute for the austenitic steels employed for Phénix, and Superphénix. This would make for improved economic competitiveness for the SFR, since production costs for F–M steels are markedly lower than for austenitic steels. Further, owing to their outstanding heat conductivity and thermal expansion characteristics, it then becomes feasible to shorten piping ducts, resulting in a major reduction in the volume of steel required.

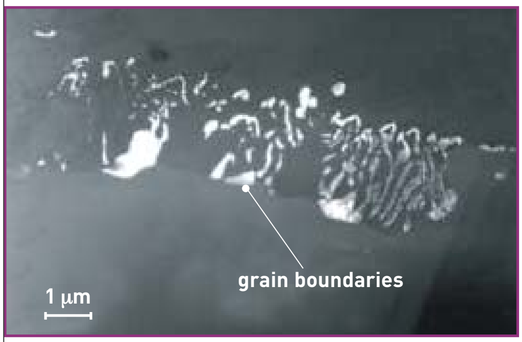

Figure 6. Precipitation of $M_{23}C_6$ carbides at the grain boundaries. – Alloy 617 after 1,000 h aging at 850 °C (from L. Guetaz).

F–M steels further exhibit outstanding resistance with respect to certain neutron irradiation-induced effects. As seen in Figure 1, their dimensional stability, and thus their swelling and creep resistance under irradiation, is excellent. Moreover, at temperatures higher than 400 °C, the mechanical properties of 9% Cr F–M steels, in particular their **toughness**, and brittle–ductile transition temperature, are very little affected by irradiation. It is on these grounds that, as a follow-up to the current reference, EM10, T92 steel is being proposed as **hexagonal tube** material, for the SFR **fuel assembly**. It will be remembered that, for fuel element cladding the candidate material is an oxide-dispersion-strengthened (ODS) F-M steel, since T92 mechanical strength in hot conditions proves inadequate.
9% Cr F–M steels are likewise the materials currently selected for the construction of high-temperature, gas-cooled reactor vessels. By contrast to what pertains to sodium-cooled reactor circuits, the issue, in this case, is to produce components exhibiting very high thickness (up to some 200 mm). Now, there is to date but limited experience, as regards fabrication of thick 9% Cr steel plates. It will prove indispensable to evaluate, as a function of thickness and heat treatment, the degree of microstructural heterogeneity, and consequently the variation in mechanical properties across the thickness, particularly with regard to creep. Another R&D direction of great importance, for vessel construction, is concerned with the assembly process. Indeed, while the TIG[6] welding process – currently inescapable as it is, if industrial development is sought – yields satisfactory results, when medium-thickness plates (a few centimeters thick) are welded, microcracking has been found to occur with thick products (the so-called "hot cracking" phenomenon). A research program is currently ongoing, to understand the mechanisms involved. This has not only made it possible to show it is essential to restrict contents of such embrittling elements as phosphorus, or sulfur, but it has also allowed, through use of thermodynamic computation tools, optimization of the chemical composition for the filler metal, to achieve improved resistance to hot cracking.
Bearing in mind the extremely extended operating lives being planned for fourth-generation reactors (60 years), the issue must be considered, of the in-service evolution of the properties of structural materials, in particular for the vessel material, thermal aging over very long intervals, under low neutron irradiation conditions must be investigated. To date, even though there is a need for caution, as regards extrapolations to highly extended aging times, the data culled from the open literature, or gained at CEA indicate high stability for 9% Cr steels under thermal aging conditions. Indeed, at temperatures under 450 °C (this being the vessel operating domain), a demixing process[7] of the "α–α'" type is liable to occur, resulting in the emergence of α' phase precipitates, of nanometer size, within the ferritic matrix (α phase). The volume fraction for the precipitate being low, this induces a slight hardening of the material, while not impairing its toughness.

Nickel alloys for very-high-temperature gas-cooled reactors

Reactors of the **VHTR** or GFR types will be operating with **coolant** helium circulating in the primary circuit at temperatures ranging from 850 °C to 950 °C. The class of material selected for that circuit, and the primary–secondary heat exchanger is that of Ni-base superalloys strengthened by the solid-solution mechanism, and by carbide precipitation. Two industrial grades are currently being evaluated: alloy 617 (Ni–Cr–Co–Mo), and alloy 230 (Ni–Cr–W). These materials are already seeing widespread employment in the chemical industry, however they have to be qualified for a nuclear reactor-type application, which entails, in particular, gathering data guaranteeing very long lifetimes (from 20 years, for the heat exchanger, to 60 years for the primary circuit). Ongoing investigations at CEA mainly concern thermal stability, corrosion resistance (at Saclay), and creep resistance (at Grenoble). At such high temperatures, indeed, the carbides present in the material coalesce, and newly formed carbides precipitate close to the

(6) TIG (tungsten–inert gas): an arc welding process, using a refractory electrode in inert atmosphere conditions, employed when very high weld quality must be achieved (pressure apparatus).

(7) Demixing: a microsegregation process, occurring between phases of diverse compositions in an alloy, when one constituent of the mixture that stands in solution exceeds its solubility limit.

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Eurofer–copper ITER TBM (test blanket module) mockup (prior to hot isostatic pressing), representative of part of a fusion reactor tritium-breeding blanket.

grain boundaries, causing a loss of ductility in the materials (see Figure 6). With regard to corrosion, the presence of impurities in the helium results in risks of oxidation, and carburization–decarburization. Finally, the extrapolation of creep data gained from tests of a few thousand hours, to predict behavior over 100,000 hours entails that the deformation mechanisms involved be perfectly ascertained, and understood, as indeed the microstructural evolutions related to the thermal and mechanical loads.

Synergies between fission and fusion reactor materials

Compared to **fission** reactors, fusion reactors will be subjected to far greater neutron stress. Indeed, the **deuterium–tritium** reaction yields a very fast, 14-**MeV** neutron, this inducing **transmutations** inside the material, generating, in particular, **helium** (10 **appm**/**dpa**[8]), and **hydrogen** (40 appm/dpa). At the same time, the neutron flux will be inducing up to 30 dpa per year in structural materials. This may possibly cause embrittlement of the material, due to a rising transition temperature, from the ductile[9] to the brittle state, which may not be contemplated in a structural material.
Development of adequate materials thus stands as a crucial technological barrier, with respect to the design of electricity generating fusion reactors. For the coming project, **ITER**, neutron **fluences** will remain very low (< 3 dpa), allowing use of "conventional" austenitic steels. On the other hand, the subsequent stages will require a material with the ability to withstand the above-mentioned very high irradiation stresses. Ongoing developments rely on the research work carried out for Generation IV fission reactors, on the basis of low-**activation** ferritic–martensitic steels, whether of the oxide-dispersion-strengthened (ODS) type, or otherwise. After a laboratory-scale development stage, the European controlled fusion program has already resulted in the industrial production of one such (non-ODS) steel, known as Eurofer, which is being targeted by numerous investigations, for the purposes of characterization (such as the determination of mechanical properties), the drawing up of a database, for the purposes of codification, along with investigations with respect to weldability, and irradiation resistance.
Concurrently, a number of laboratories are investigating the feasibility of an ODS Eurofer grade: the aim is to incorporate nanometer-size Y_2O_3 oxides, allowing enhanced performance, thus making it possible to raise maximum allowable temperature by 100 °C. A number of ODS variants of Eurofer have been fabricated in various European laboratories, including at CEA, where the process route selected is based on mechanical alloying, followed by hot isostatic compaction – yielding a material for which a typical microstructure is shown in Figure 7. The materials obtained to date all exhibit much reduced **impact strength** values, compared to conventional Eurofer. Investigations are currently ongoing, to understand the cause of this loss of toughness, and optimize the fabrication process.
Eurofer is to be tested in ITER, and, most crucially, in the **IFMIF** (International Fusion Materials Irradiation Facility) source, which should produce sufficiently large fluxes of high-energy-spectrum, fusion-type neutrons to investigate, in accelerated fashion, aging of such steels: an agreement between the European Union and Japan, parallel to that concluded for the ITER program itself, covers the design of such a reactor: a project team is starting work as of now at the Rokkasho-Mura site (Japan). This effort goes hand in hand with multiscale ***ab-initio*** modeling development work, targeting the material's behavior under irradiation. This materials program lies on the critical path for the DEMO demonstration reactor, the goal for which will be to demonstrate the ability to build an electricity production fusion reactor.

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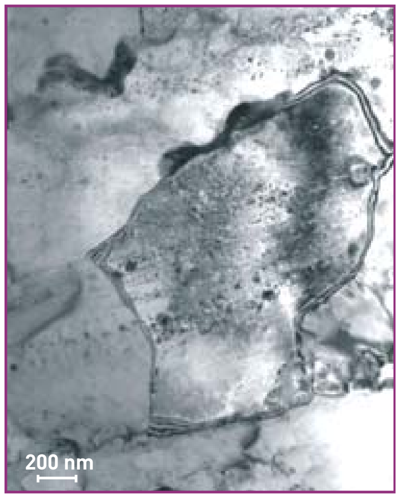


Figure 7.
ODS Eurofer, strengthened with 1% Y_2O_3 (from C. Cayron).

Table
Materials being considered for in-core application, for the six systems being investigated by the Gen IV Forum.

	VHTR	SFR	GFR	SCWR	LFR	MSR
fuel U, U-Pu	SiC-, ZrC-coated particles	MOX metal	carbide nitride	UO_2 MOX	nitride	fluoride (chloride)
fuel TRU		MOX metal	carbide nitride			fluoride (chloride)
core materials	graphite C/C,SiCf/Sic	AISIs AFMSs, ODSs	SiC, TiC other ceramics	Ni alloys		graphite
structural material	AFMSs	AFMSs, ODSs austenitic steels	AFMSs, ODSs Ni alloys	Ni allys		hastelloys

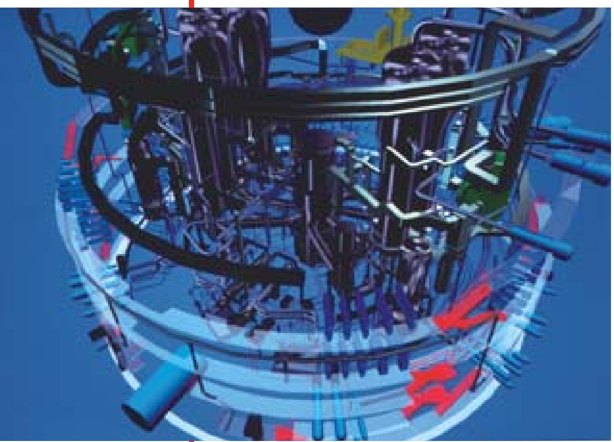
(8) appm: atomic part per million.
(9) Ductile state: the state of a material that undergoes plastic deformation while resisting the propagation of defects induced by its deformation, as opposed to the brittle state, prevailing prior to rupture.

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,



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 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

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

FOCUS (Cond't) B

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

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

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

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

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

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

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

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

FOCUS C

Thermodynamic cycles and energy conversion

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

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

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

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

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

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

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

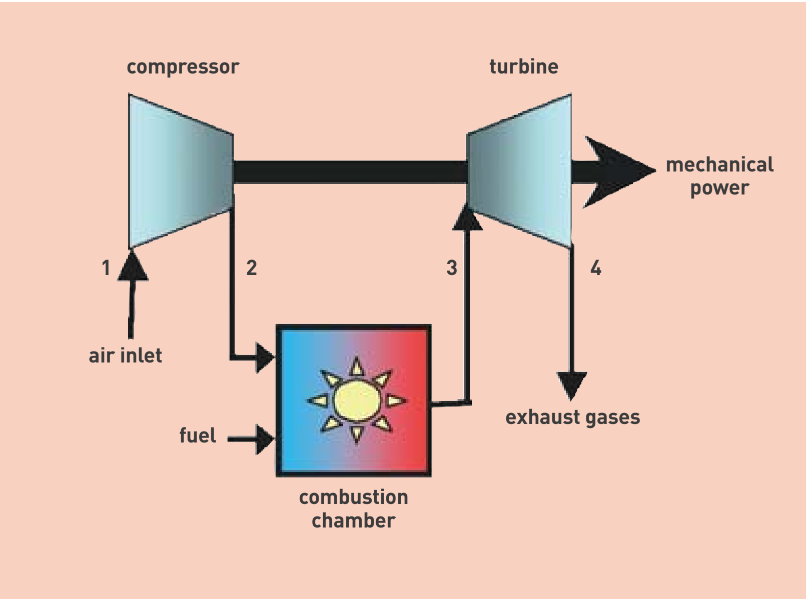


Figure.
Brayton cycle, as implemented in an open-cycle gas turbine.

FOCUS D

What is multiphysics, multiscale modeling?

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

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

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

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

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

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

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

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

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

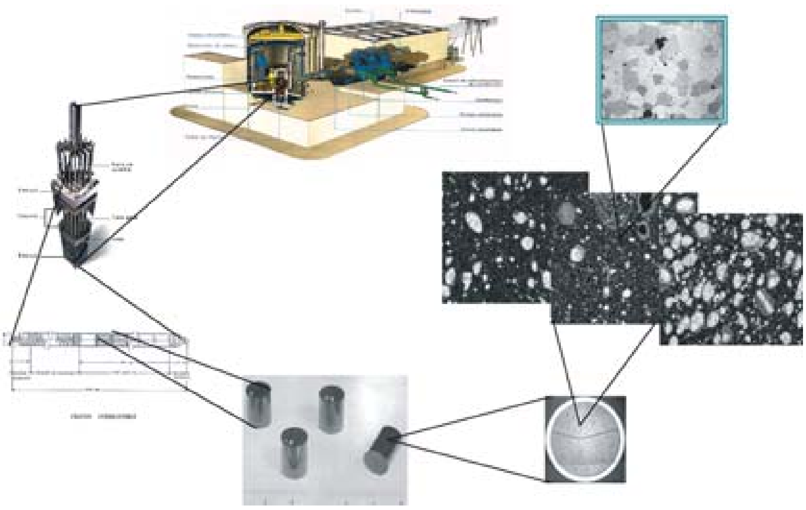


Figure.
Improving nuclear fuel reliability, and cost-effectiveness calls for finescale modeling of that fuel, through a multiscale approach, from reactor to fuel microstructure (in this instance, MOX fuel). Microstructural characteristics (porosity, cluster size and distribution, grain size…) have a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

FOCUS E

The main families of nuclear materials

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

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

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

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

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

(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.

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 $\cdot$ cm^{-2}, for neutrons of energies higher than 1 **MeV**.

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

Areva NP

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

FOCUS (Cond't) E

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

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

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

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

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

Oxide-dispersion-strengthened (ODS) **ferritic and martensitic steels** were developed to combine the swelling resistance exhibited by ferritic steels, with a creep resistance in hot conditions at least equal to that of austenitic steels. They currently provide the reference solution for fuel cladding, for future sodium-cooled reactors.

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

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

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

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

In particle fuels, uranium and plutonium oxides are coated with several layers of insulating **pyrocarbons**, and/or silicon **carbide** (**SiC**), possibly in fibrous form (**SiCf**). These are known as coated particles (CPs). While SiC-coated UO_2, or **MOX** balls stand as the reference, ZrC coatings might afford an alternative.

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

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

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

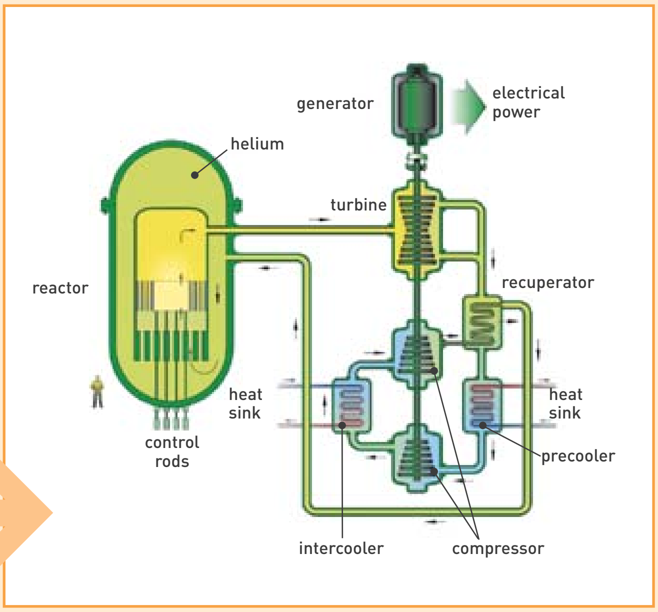
The six concepts selected by the Gen IV Forum

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

GFR

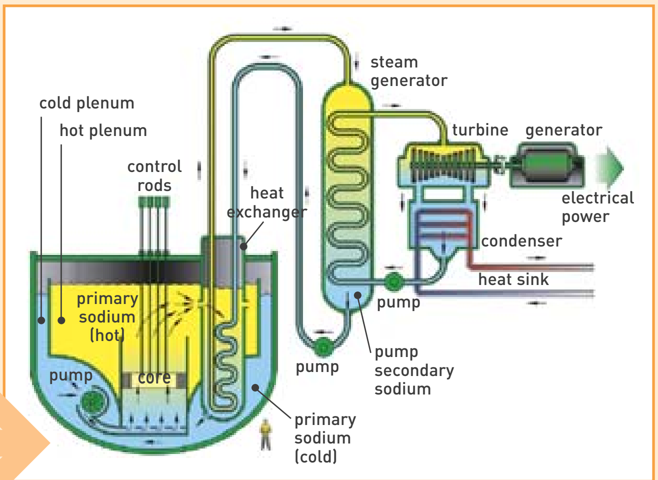
The gas-cooled fast reactor system (GFR) is a high-temperature, gas-cooled (helium-cooled as a rule), fast-neutron reactor allowing actinide recycle (homogeneous, or heterogeneous), while sustaining a breeding capability greater than unity. The reference concept is a helium-cooled, direct- or indirect-cycle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

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



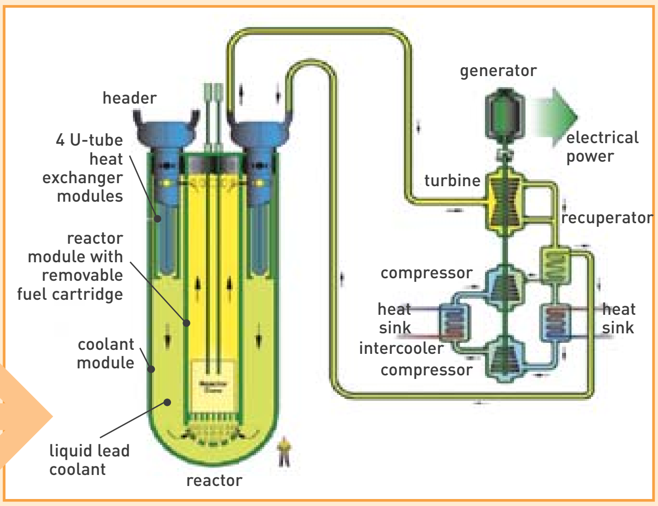
Le SFR

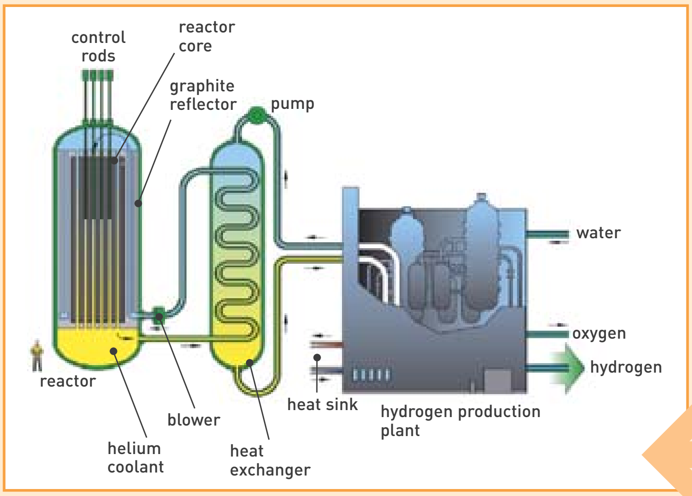
The sodium-cooled fast reactor system (SFR) is a liquid-sodium-cooled, fast-neutron reactor, associated to a closed cycle, allowing full actinide recycle, and plutonium breeding. Owing to its breeding of fissile material, this type of reactor may operate for highly extended periods without requiring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



LFR

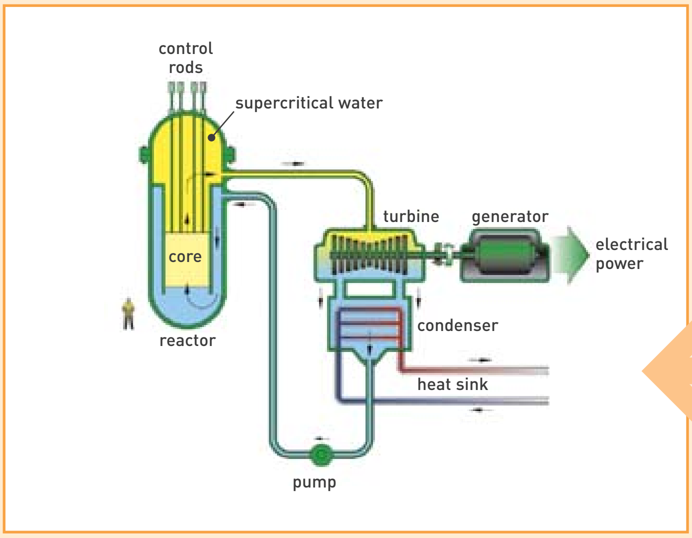
The lead-cooled fast reactor system (LFR) is a lead- (or lead–bismuth alloy-) cooled, fast-neutron reactor, associated to a closed fuel cycle, allowing optimum uranium utilization. A number of reference systems have been selected. Unit power ranges from the 50–100 MWe bracket, for so-called battery concepts,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





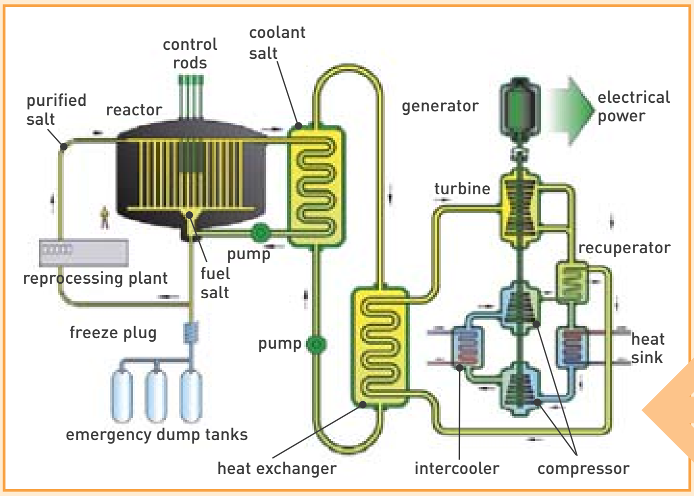
VHTR

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



SCWR

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



MSR

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