IRSN
INSTITUT
DE RADIOPROTECTION
ET DE SÛRETÉ NUCLÉAIRE

Generation-four (GEN-IV) reactors

Summary report

Version dated 21 November 2007

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IRSN

//Short view

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The Institute exercises specialist and research assignments in the following fields:

- nuclear safety;
- safety in transporting radioactive and fissile materials;
- protection of man and the environment against ionising radiation;
- protection and control of nuclear materials;
- physical protection of facilities and transport of radioactive and fissile materials.

Doctrine & synthèse

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FOREWORD

Preparation for the fourth generation (GEN-IV) of nuclear reactors

At the December 20th, 2006 meeting of the Council of Ministers, the ministers in charge of research and industry gave a presentation on the fourth generation (GEN-IV) of nuclear reactors.

In the words of the press release, "*France has decided to commit itself resolutely to participate in the design of the fourth generation of nuclear reactors, the industrialization of which could start from the year 2040.*

The industrial design of these reactors must satisfy several requirements, fixed by the Government, relative to the third generation (GEN-III) reactors:

- *reducing the volume and radiotoxicity of waste produced;*
- *generating the same quantity of energy while using a lower amount of uranium;*
- *further improving the safety and security of reactors;*
- *reducing the proliferation risks.*

... (The research to be performed over the coming years) will focus on the sodium - or gas-cooled - fast reactor concepts ...

A milestone has been set for 2012 to decide on the technological choices from among the different technological options and to undertake the construction in France of a prototype of a GEN-IV reactor, with a view to commissioning in 2020, in accordance with the decision taken by the President of the French Republic in January 2006".

It is part to the IRSN to give technical advice on the protection of persons, property and the environment and to carry out research programmes aimed at maintaining and developing the knowledge which is necessary for the expertise of industrial projects in the fields of nuclear security in its widest meaning (safety, radiological protection, non-proliferation, resistance to acts of malicious damage).

In this framework, the IRSN initiated internal reflections on safety, radiological protection and security-related issues affecting the six reactor concepts selected, at the international level, by the "Generation

IV International Forum (GIF)" created in 2000 at the initiative of the American Department Of Energy (DOE). This think-tank, which started in 2004-2005, was upgraded, following the announcement by the President of the French Republic in early 2006, to engage the country in the construction of a GEN-IV prototype by 2020. In addition to power reactor concepts, the IRSN's analysis has been extended to the associated fuel cycles.

Taking into account the government orientations, the IRSN decided to put forward the present state of its thinking on the six reactor concepts mentioned above, it being understood that the subsequent work of the Institute would mainly focus on fast reactor safety. The high temperature and very high temperature reactors, not retained due to their "open" cycle (where actinides are not recycled), will for their part be the subject of an active watch in conjunction with industrial partners interested in their development.

The fast reactors so far selected call on two different concepts:

- The sodium-cooled fast reactors: this concept has already been built up to an industrial scale in France with Rapsodie and Phénix as well as the Creys-Malville power plant; subsequent projects (RNR 1500, EFR) underwent preliminary analysis either within a formal national framework (RNR 1500) or within an international informal framework (EFR). It is clear that some improvements of the concept, likely relevant ones, will be proposed by designers with the aim of improving its competitiveness, safety, security and waste management performance.

- The gas-cooled fast reactors: this concept has not yet been worked on either in France or abroad. Its selection in the framework of GIF is linked to its features for actinide transmutation and its high thermodynamic efficiency (the target temperatures allow considering the production of heat for industrial processes). Future developments of the concept will first require the construction of an experimental reactor (the current REDT project of the French Atomic Energy Commission - CEA- is a 50 MWth reactor; the decision for its construction, if any, would not be taken by 2012).

Concerning the sodium-cooled fast reactors, the text herewith attached emphasizes a number of specific issues, which require in-depth examination between now and 2012. The main concerns are presently the features of the reactor core as well as its behaviour in both incidental and accidental situations, in relation to the choice of cooling devices, and the in-service monitoring of the various components (internal structures, steam generators). It is clear that the design improvements proposed by designers should, in due time, be the subject of in-depth discussions (e.g. giving up the intermediate sodium circuits). Nevertheless, the actual state of knowledge appears sufficient for a brainstorming to be launched in 2007 on the main safety issues to be addressed for an industrial size reactor, as it was done for the 3rd PWR reactor generation at the end of the 1980s (the studies, conducted jointly by IPSN[1] and GRS, led, in mid-1993, to a common statement by the French and German Safety Authorities on the safety objectives to be searched for the EPR reactor project). The proposed think-tank will take into account the results of the analysis carried out for the RNR 1500 and EPR projects. It will include discussions with other expertise assessment bodies at the international level. Moreover, it will contribute to defining the research work in support of the future expertise of a reactor project. It is worth stressing that, if experiments in "large" research facility were to be carried out, they should be defined and scheduled in the coming years in order to match the 2020 milestone.

Concerning the gas-cooled fast reactors, the concept design –even for an experimental reactor- is not achieved enough to adopt a similar approach in the same time frame. It is essential for designers to perform preliminary feasibility studies with particular attention to the safety questions already identified (the main concern being the core cooling in accidental situations).

In the meantime, the IRSN will also conduct a brainstorming on security-related issues (non-proliferation and protection against acts of malicious damage) for both concepts; these problems must be stressed from the very beginning of the design phase so that the overall plant design accounts at best for all the nuclear security-related aspects.

[1] The IPSN (*Institut de Protection et de Sûreté Nucléaire*) became the IRSN in 2002.

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1/ Introduction

Created in 2000 at the initiative of the American Department Of Energy (DOE), the "Generation-Four International Forum" (GIF) today associates twelve countries, including France and the community of states which signed the Euratom Treaty, for cooperation in the development of new nuclear systems (reactors and fuel cycle facilities) allowing the production of energy beyond the life of reactors in operation or under construction (including EPR).

In this regard, the current plans of *Electricité de France* (EDF) include the development of new reactor concepts which could be industrially deployed around 2040 (Generation-Four, GEN-IV), after extending the operation of current reactors of GEN-II (current PWRs) and commissioning the "evolutionary" reactors of GEN-III (EPR); 2040 also matches the planned schedule for the renewal of the fuel cycle reprocessing plants. This implies, for EDF, commitment to an industrial prototype of the Generation-Four around 2015-2020.

The first phase of the work of GIF resulted, in 2002, in the publication of a development plan for those technologies considered as the most promising. Six reactor concepts were selected, which should allow for significant advances in economic competitiveness, safety, savings in uranium resources, reduced radioactive waste production, especially high activity long-lived ones, and resistance to proliferation and acts of malicious damage. These reactor concepts are as follows:

- Helium-cooled High Temperature and Very High Temperature Reactors (HTR/VHTRs),
- Sodium-cooled Fast Reactors (SFRs),
- Gas-cooled Fast Reactors (GFRs),
- Lead or Lead-bismuth cooled Fast Reactors (LFRs),
- Supercritical Water cooled Reactors (SCWRs),
- Molten Salt Reactors (MSRs).

The USA, Japan and France are more involved in the gas-cooled (VHTR & GFR) and sodium-cooled (SFR) reactor concepts. Coordination of GIF-led actions is taken on by:

- France for HTR/VHTRs,
- Japan for SFRs,
- the USA for GFRs.

In France, the company AREVA is carrying out the project of an industrial VHTR with a thermal power of about 600 MW, named ANTARES (Areva's New Technology and Advanced gas-cooled Reactor for Energy Supply), and the French Atomic Energy Commission (CEA) is carrying out the GFR and SFR concept studies. AREVA intends to bid for a possible call for tender by the DOE for a VHTR, the "Next Generation Nuclear Plant" (NGNP), which should be built, between 2015 and 2020, as a demonstrator, on the site of the Idaho National Laboratory in the USA. On its side, several years ago, the CEA announced intention to commission, by 2015, an Experimental Reactor for Technology Development (*REDT* in French) of the GFR type; this reactor should generate a power of 50 MWth.

Moreover, the following activities are carried out with a contribution from the European Commission:

- a VHTR integrated project, named RAPHAEL ((ReActor for Process heat, Hydrogen And ELectricity generation),
- a wide research and development activity in support to the GFR concept design (known as GCFR project).

In January 2006, the President of the French Republic set the deadline of 2020 for commissioning a GEN-IV reactor prototype. Moreover, article no 3 of the June 28th 2006 law on the radioactive material and waste sustainable management indicates that *"the studies and research* (on the separation and transmutation of long-lived radioactive elements) *are conducted in conjunction with those carried out on the new generation of nuclear reactors mentioned in article 5 of programme law no 2005-781 of July 13*th *2005 establishing the orientations of French energy policy, as well as on waste transmutation dedicated accelerator-driven systems, in order to have an assessment of the industrial perspectives of such reactor concepts by 2012 and to put a prototype into operation before 31 December 2020".*

In this context, this report gives an overview of the main features of the reactor concepts selected by the GIF and presents the main safety, radiological protection and security challenges raised by concepts which are most likely to be built in the indicated time frame, due either to their industrial maturity, or to a significant interest in their capacity to match the above-listed objectives (savings in uranium resources, etc.). Lastly, it describes the R&D actions undertaken or planned by the IRSN, identified in terms of safety, radiological protection and security.

2/ Features and industrial maturity of the six Generation-Four (GEN-IV) reactor concepts

2/1 Helium-cooled High Temperature and Very High Temperature Reactors (HTR/VHTRs)

HTR/VHTRs are graphite-moderated, thermal reactors; the heat generated in the core is extracted by a gas (helium) under pressure. Several thermodynamics cycles for electricity generation are proposed which would allow very high efficiencies (nearly 50 %, compared to about 35 % for PWR reactors).

The VHTR concept is an evolutionary stage of HTR reactors, which were built in the 1970s and 1980s, mainly in the USA and Germany. We can mention:

- Research reactors from 15 to 40 MWe, such as the Peach Bottom reactor (USA) and the AVR reactor (Germany),
- Reactors of about 300 MWe, such as the Fort Saint Vrain reactor (USA) and THTR reactor (Germany).

The main significant safety-related events which have affected the HTR operation are as follows:

- In the Fort Saint Vrain reactor, problems of neutronic instability induced by the movements of graphite blocks in the core and infiltration of water in the primary circuit at the level of the moto-blowers,
- In the THTR, ruptures of thermal insulation fixing elements on a core outlet pipe.

The objective for VHTRs is reaching an average coolant temperature of about 1000°C at core outlet, compared to 750°C -850°C for HTRs. VHTRs could thus be a supplier of heat for industries with high energy consumption, in particular regarding the production of hydrogen through advanced processes, some of which, such as the high temperature electrolysis, and the iodine/sulphur cycle, are still under development.

Two geometric designs are proposed for the fuel elements: the pebbles and the rod « compacts ».

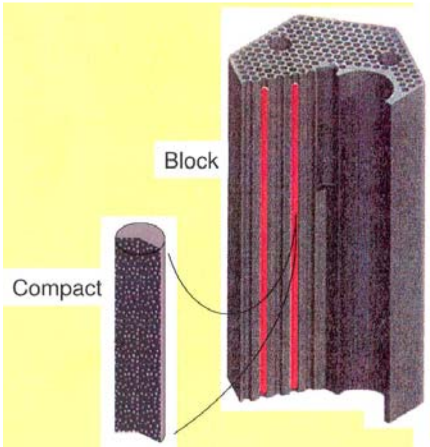
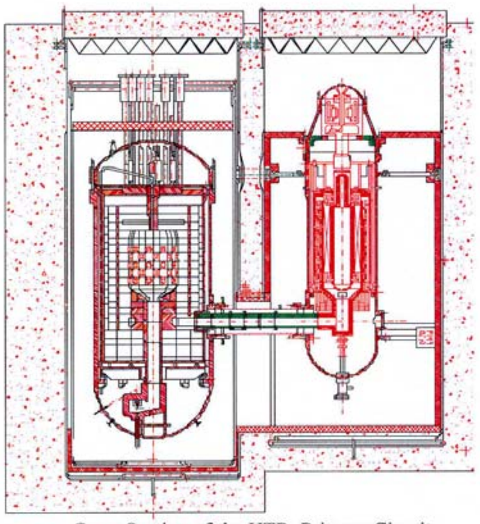


Figure 1:
Left: a reactor pebble.
Right: “compact” and fuel block for high temperature or very high temperature reactors.

However, in both cases, the nuclear material is confined in a spherical micro-particle named TRISO (TRistructural ISOtropic), made of a central kernel, coated with three layers of refractory material.

Figure 2:
A TRISO particle.
Refracting layers are shown in blue.

More recently, two experimental reactors have been commissioned: the Chinese 10 MWe HTR 10 reactor, built near Beijing, which went critical in 1998, and the Japanese VHTR 30 MWe HTTR reactor, built on the Oarai site, which achieved criticality in 2001. The HTTR is equipped with a gas turbine; a hydrogen production process facility was installed in 2005 and is at present undergoing testing.



Cross Section of the HTR. Primary Circuit

The HTR-10 Main Design Parameters

Reactor thermal power	MW	10
Active core volume	m^3	5
Average power density	MW/m^3	2
Primary helium pressure	MPa	3
Helium inlet temperature	°C	250 / 300
Helium outlet temperature	°C	700 / 900
Helium mass flow rate	kg/s	4.3 / 3.2
Fuel		UO_2
U-235 enrichment of fresh fuel elements	%	17
Diameter of spherical fuel elements	mm	60
Number of spherical fuel elements		27,000
Refuelling mode		multi-pass, continuous
Average discharge burnup	MWd/t	80,000

Figure 3:
The Chinese HTR10 reactor using pebbles.

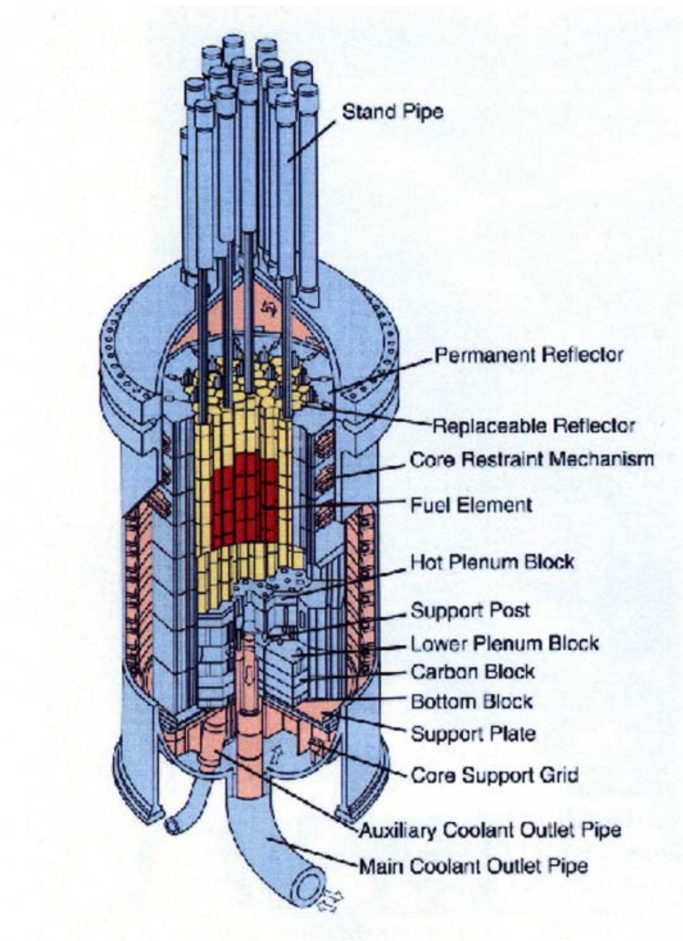


Figure 4:
The Japanese HTTR reactor.

The fuel cycle of the HTR/VHTR concept is "open", but preliminary studies for "closing" it by recycling the fuel in the reactor have been performed and are now still underway within a European R&D framework.

The HTR/VHTR concept has reached a stage of maturity allowing the construction, by 2020, of an industrial prototype operated through an "open" cycle. South Africa is already developing a pebble bed reactor project (PBMR) of about 100 Megawatts of electrical power, with a helium turbine; its announced commissioning objective is around 2010.

2/2

Sodium-cooled Fast Reactors (SFRs)

SFRs are reactors with a fast neutron spectrum, which therefore contain no moderator; the heat generated in the core is extracted by a liquid metal (sodium). The fast neutron spectrum enables the efficient transformation of natural uranium (U238), the “fertile” material, into “fissile” plutonium with a “conversion ratio” (number of plutonium nuclei generated per fission) close to 1 (that’s the SFR breeding-mode) or, contrariwise, a consumption of plutonium (the SFR incinerating mode); it also enables a transmutation of very long-lived actinides. The temperature reached by the sodium (in the range of 550°C) allows a quite high thermodynamic efficiency (about 40 %).

The reference fuel is generally mixed uranium and plutonium oxide; the use of mixed carbide or nitride is also planned. Moreover, SFR concepts are characterised by a great thermal inertia due to the large amount of sodium present in the primary circuit, which allows the operators a quite long response-time to intervene in the event of failure to remove the heat, as well as a margin of more than 300°C before the boiling of sodium in normal operation. Moreover, sodium has a high capacity for trapping some fission products in the event of damage to the fuel.

SFRs are operated through a “closed” fuel cycle, which allows in reactor recycling of uranium, plutonium and minor actinides. The industrial feasibility of reprocessing the SFR fuel (mixed uranium and plutonium oxide) with the current PUREX process has already been demonstrated in the reprocessing plant of La Hague. Several tonnes of fuel unloaded from the Phénix reactor diluted with fuel irradiated in the UNGG reactors (natural uranium graphite gas cooled reactors) have been successfully reprocessed in the UP2-400 factories at La Hague. Moreover, the recycling of plutonium and minor actinides in SFRs has been tested in experiments carried out in Superphénix (CAPRA programme) and in Phénix (PAVIX experiment) reactors.

SFRs have the advantage of a large experience, via the operation of several power reactors:

- In France, the experimental Rapsodie reactor (initially 25 MWth, later upgraded to 40 MWth), then the Phénix (250 MWe) and Superphénix (1240 MWe) reactors,

Figure 5:
The Phenix reactor hall is accessible during operation. The reactor in operation since 1973 will be shut down in 2009.

- In Britain, the PFR reactor (135 MWe),
- In the states of the ex-Soviet Union, the BN 350 (135 MWe) and BN 600 (550 MWe) reactors; the construction of the BN 800 reactor, which had been postponed for several years has been put again on the agenda,
- In Japan, the experimental Joyo reactor, the power of which reached 140 MWth, and the 280 MWe Monju reactor, the resumption of operation of which, interrupted after a sodium fire in 1995, is planned for about ten years from now.

Some of the significant events which have affected SFRs (Phénix, Superphénix, PFR, and Monju) are mentioned below in section 4/2. These events have diverse origins, such as inappropriate operator reactions, design errors, inhibited security systems, insufficient performance requirements for some items of equipment, difficulty in mastering performance due to the complexity of the industrial organization. Moreover, it must be emphasized that the Phénix reactor underwent, in 1989 and 1990, automatic shutdowns due to a sudden decrease in reactivity, the origin of which remains unexplained and is still under investigation.

India is currently building a 500 MWe prototype fast breeder reactor (PFBR) while China is building an experimental reactor, the CEFR, with a power of 25 MWe.

In France, in the 1980s and after the commissioning of Superphénix, studies were carried out on a 1500 MWe SFR reactor project (named RNR 1500), with an increased breeding capacity and lower neutronics sensitivity to the loss of coolant, including "fertile" or even non fuel zones in the inner core. Such studies were pursued in the European framework, associating the British, the Germans and the French on an SFR power reactor fleet project (EFR project).

Two different designs are proposed for existing and future SFRs: the "integrated-type" concept (Phénix, Superphénix, PFR, CEFR), the primary circuit of which is confined in the reactor vessel (which contains both the primary pumps and the heat exchangers), and the "loop type" concept (Joyo, Monju) the primary sodium of which circulates in loops linking the main reactor vessel to secondary reactor vessels containing the large components.

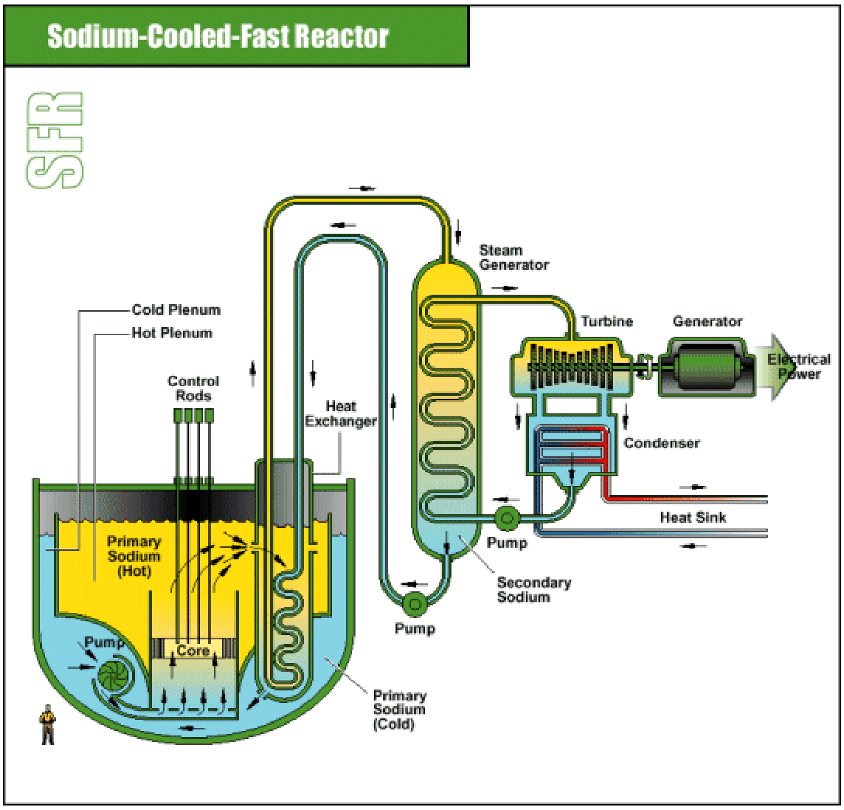


Figure 6:
Schematic diagram of an integrated type sodium-cooled fast reactor.

The search for cost reductions should lead the designers to suggest innovative solutions to simplify the systems and their equipment.

The SFR concept has reached a stage of comfortable maturity, which makes the construction of a new industrial prototype be likely for 2015-2020.

2/3

Gas-cooled Fast Reactors (GFRs)

GFRs are fast neutron spectrum reactors, the heat of which, produced in the core, is extracted using a gas (helium) under pressure. In principle, GFRs enable actinides to be transmuted thanks to their fast neutron spectrum, while remaining by nature insensitive in the event of loss of coolant and ensuring high thermodynamic efficiencies and, possibly, the production of industrial heat (helium temperature of at least 850°C in normal operation).

There is no experience of operating GFR reactors. Nevertheless, the concept is not totally new: studies have been engaged as early as 1962 in the USA (GCFR project) and since 1968 in Europe (GBR project), in which the French Atomic Energy Commission (CEA) has been involved.

The GFR concept evaluated in the framework of GIF is being developed by the CEA (project for potentially 300 to 1200 MWe of power). It includes an entirely refractory core (fuel contained in a ceramic material), able to confine the very high temperature nuclear materials.

The IRSN has no detailed documents describing the design of a GFR and in particular the core design, the barriers and the main equipment. The project is still in the preliminary definition phase and only general descriptions are available. Thus, the nature of the fuel has not already been precisely defined (oxide, carbide or nitride, with a refractory coating), but the oxide solution seems to be excluded due to its too low density in heavy nuclei (for the same power outlet, a GFR needs a greater free volume for the coolant than a SFR, due to the lower heat extraction capacity of gas compared to sodium, thus a fuel with a higher fissile kernel density is necessary).

A few experiments have already been performed in the Phénix reactor, on a nitride based fuel (NIMPHE experiments), and further experiments were loaded in the reactor in 2007 (FUTURIX-MI experiments on structure materials, FUTURIX-CONCEPT experiments on the fissile material and cladding, FTA-nitride experiments).

The GFR fuel cycle is "closed", with the possibility of recycling the uranium, plutonium and minor actinides. It should be emphasized that GFRs were the reference system selected by the CEA in the framework of axis 1 of the 1991 law relative to the research for the management of high activity and long-lived waste (partitioning and transmutation).

The development of the GFR concept would include, in the first phase, the commissioning of a low power experimental reactor. The construction of such a reactor implies the solution of several technological problems, mainly on innovating fuels at present under development, and the assessment of the safety aspects to be addressed. Commissioning by 2020 of a demonstration reactor, such as the *REDT*, currently studied by the French Atomic Energy Commission, seems quite optimistic.

2/4
Lead or lead-bismuth cooled Fast Reactors (LFRs)

LFRs are fast reactors cooled by a molten metal, such as lead or a lead-bismuth alloy.

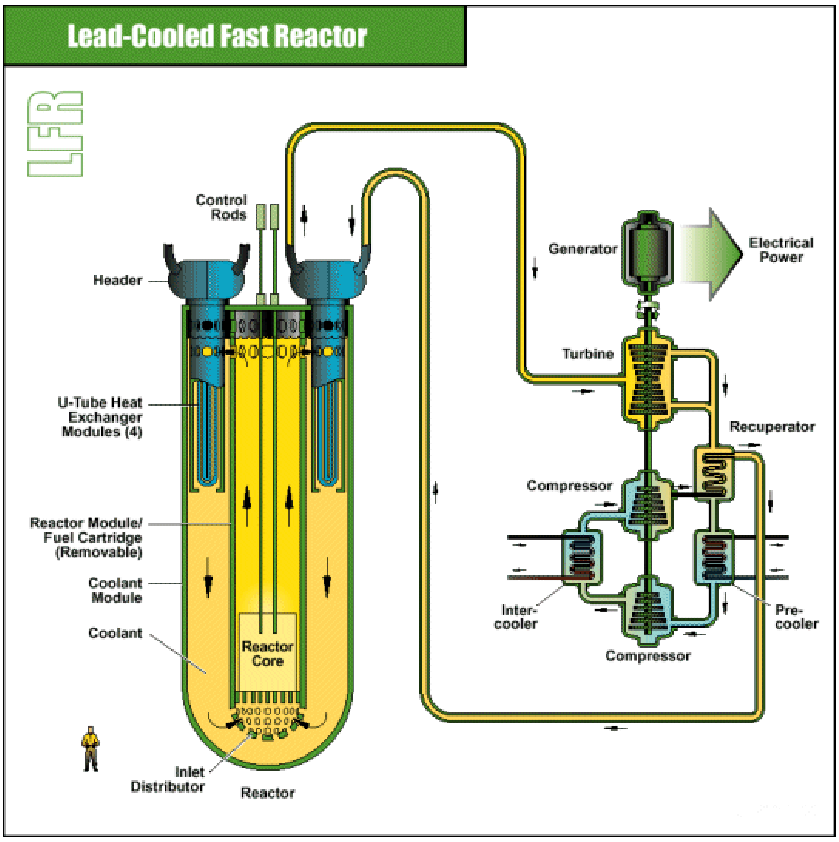


Figure 7:
Schematic diagram of a lead-cooled fast reactor.

The fuel cycle of the concept is “closed”.

One advantage of lead or of a lead-bismuth alloy compared to sodium is the absence of any chemical reaction with water and air.

The Russian BREST 300 (300 MWe) is the most industrially advanced project of a LFR in the world. In this design, the lead is heated up to a temperature of 550°C.

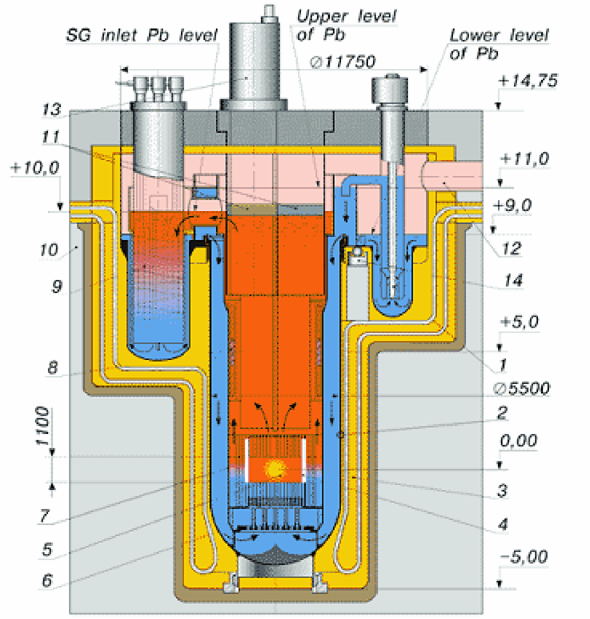


Figure 8:
The BREST project.

This concept is derived from Russian Alpha class submarine reactors, cooled with a lead-bismuth alloy, which were abandoned in favour of PWR reactors. The last submarine of this class was disarmed in 1995.

The submarines equipped with LFRs suffered from huge maintenance problems, especially due to the presence of lead activation products. According to available information, serious failures occurred in two of them.

Several design choices on the BREST 300 project, notably a moderate power density, allow a satisfactory fuel behaviour, even in the event of accident with protection system failure, thanks to the passive cooling (natural convection).

The main difficulty of LFRs is due to the highly corrosive nature of the molten lead on the steel structures. The only known preservation method consists in creating, then maintaining, a protective layer of iron oxides on the surface of the steel structures in contact with the lead. That involves injecting oxygen into the lead and scrubbing the lead oxides and the corrosion residues present in it. This method was developed for the operation of submarine reactors, but deviations in its application originated severe failures in two vessels (blocking, assembly melting) and one of them sank.

Moreover, the BREST 300 project energy conversion system has not yet been defined. In the present project, the reactor, which is of the integrated type, has no intermediate circuit (to minimize the costs). The secondary circuit would use supercritical water at a pressure of 250 bar; this would have the disadvantage of positioning equipment under high pressure near the reactor core, resulting in no appreciable gain, in terms of thermodynamic efficiency, compared to a design with an intermediate circuit.

Spent fuel processing methods (nitride or metallic type) as well as the fabrication of fuels containing minor actinides remain to be developed.

Moreover, due to its activation, the lead-bismuth type coolant would contain long-lived alpha emitting radioactive waste.

Thus, in the light of experience feedback and the present state of knowledge, a deployment of LFR reactors on an industrial scale is not likely within the considered time frame. In any case, the construction of the first experimental reactor would require the solution of several technological challenges and the solution of some major safety concerns.

2/5
Supercritical Water Reactors (SCWRs)

The concept design of supercritical water reactor is derived from experience from the thermal electricity production plants which have adopted this technology. The use of water in supercritical state, at a 250 bar pressure and a temperature of about 550° C, increases the plant efficiency significantly, compared to a pressurized water reactor: in a direct thermodynamic cycle, it can be as high as 44%.

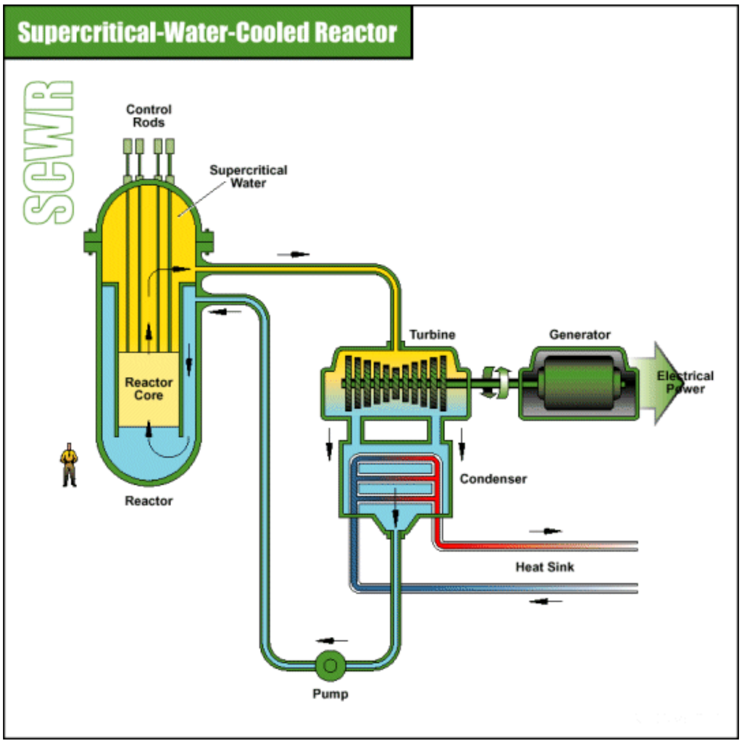


Figure 9:
Schematic diagram of a supercritical water reactor.

The concept developed by Westinghouse and sponsored by the DOE is also minded to the search for lower investment costs than for PWR reactors (ruling out the secondary circuit).

Supercritical water reactors have not passed the stage of not convincing feasibility studies (e.g., the first core calculations performed by Westinghouse were based on an assembly with more than ten enrichment zones).

An industrial deployment of supercritical water reactors is unlikely within the considered time frame and the construction of the first experimental reactor does not seem foreseeable by 2020.

2/6

Molten Salt Reactors (MSRs)

Two different MSR concepts are proposed: those in which the molten salt only acts as a coolant, and those the fuel of which is dissolved in the molten salt. In the latter, the molten salt contains a mixture of thorium (a "fertile" material) and uranium 233 (a "fissile" material). Such reactors allow the fission of uranium 233 produced from thorium, so that an initial load of uranium 233 or plutonium is required at start-up. The fuel cycle of MSRs is "closed": on line extraction of transuranics elements and fission products, then uranium 233 recycling.

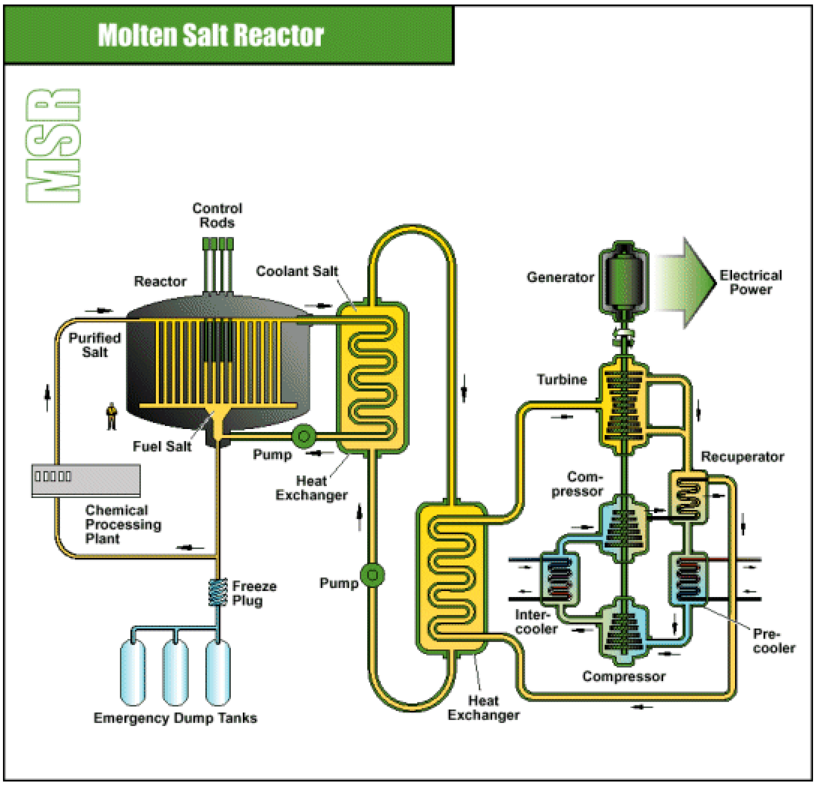


Figure 10:
Schematic diagram of a reactor in which the fuel is dissolved in molten salts.

Two experimental MSRs, with fuel dissolved in the molten salt, have already been built and operated in the USA. The first, intended for military aeronautic propulsion, was constructed during the 1950s in the framework of the "Aircraft Reactor Experiment" project. The second is the MSRE ("Molten Salt Reactor Experiment"), constructed in 1962 at Oak Ridge, which went critical in June 1965; it did not use any "fertile" material (thorium), but uranium 235-based fuel, then uranium 233; this reactor, which delivered power of 8 MW, was shut down in 1969 after approximately 13,000 hours of operation.

From an industrial point of view, the designers of molten salt reactors claim that this concept could be operated at high temperatures without requiring high pressures, with smaller diameter lines than those of helium cooled reactors (which require a much higher coolant flow rate for the same amount of energy extracted from the core). This should be advantageous if the reactor is coupled to a hydrogen industrial production facility.

Molten salt reactors have not passed the experimental stage threshold and leave complex problems open as for controlling the risk of corrosion, which should probably push R&D towards non-metallic materials. Moreover, a specific fuel reprocessing unit should be associated with such a reactor.

The construction of a molten salt reactor by 2020 seems unlikely.

2/7

Summary and generalities

As regards the three concepts for which the implementation of an experimental reactor or an industrial prototype is foreseeable by 2020 (i.e. HTR/VHTR, SFR and possibly GFR), the IRSN reviewed the technical challenges identified to date in terms of safety. They are summarized in the following paragraphs, as well as some elements concerning the radiological protection.

Concerning security, designers have not so far provided precise information in terms of proliferation resistance or protection against malicious damage (reactor containment barrier resistance, access right to different zones in the plant, etc.), nor wondered about specific sensitivities for the concepts under development. Moreover, while for SFRs (Superphénix, Monju loop reactor, EFR project, etc.) and to a lesser extent for HTR/VHTRs it is possible to refer to detailed designs, at present no GFR project design is sufficiently advanced to allow going beyond general thoughts in terms of security.

Among the generic aspects which will need to be considered for all the GEN-IV concepts, those concerning the design and the containment resistance to external aggressions, which could result from possible malicious actions, are of major importance; they mainly concern the reactor buildings.

Moreover, the physical protection devices and the monitors, which enable nuclear materials to be constantly located in order to prevent or detect their theft, loss or diversion need to be adapted depending on whether they apply to plants operated through an "open" (in the case of HTR/VHTRs) or a "closed" cycle (fast reactors). In the latter case, especially if advanced fuel separation is planned for recycling actinides, particular attention should be paid to the design of physical protection devices.

Some remarks specific to the HTR/VHTR, SFR and GFR concepts are outlined in the following paragraphs.

Finally, in what follows, the IRSN will not discuss the specific problems connected to the declared lifetime objectives for GEN-IV reactors; several documents mention life-times of 60 years and more, which implies an appropriate certification for materials and equipment which could not be replaced during service.

3/ Main safety, radiological protection and security issues identified to date for High Temperature and Very High Temperature Reactors (HTR/VHTRs)

3/1 Safety objectives and approach

While the major safety and radiological protection objectives for the GEN-IV reactors must be at least as ambitious as those defined for the EPR project, their operational application cannot, at present, be specified in details without better knowledge of the systems considered. It must be borne in mind that, for the EPR project, the preliminary definition of the safety objectives was set out as early as 1993, within a few months because the PWR concept had been investigated in-depth by the concerned parties, including the IRSN, since the beginning of the French electro-nuclear programme in 1973, and that the EPR project had been placed from its very beginning in the continuity of the current design of reactors in operation.

Nevertheless, in March 2004, the French Atomic Energy Commission submitted to the Nuclear Safety Authority a document, drawn up in collaboration with AREVA NP, presenting the “first ideas” on a “generic” safety approach for the GEN-IV systems and, more particularly, VHTRs and GFRs (letter CEA/DEN/DDIN/SF DO 73 of 18 March 2004). This document mentions the simultaneous use of deterministic and probabilistic methodologies, e.g. a “risk informed” type approach, as well as the “lines of defence”. An evaluation of this safety approach would require knowledge of its main applications to VHTR and GFR concepts.

We can note here the following safety objectives put forward in the preliminary CEA's document:

- No need for emergency measures outside the reactor site "for any accidental situation" (more detail needed on the definition of such a situation),
- "Practical elimination" of very highly degraded situations with a strong likelihood for radioactivity release, in particular severe generalised core damage (for example, core meltdown).

Concerning VHTRs, the designer's safety approach extensively exploits the physical properties of the fuel and core materials, which would allow controlling both normal operation and cooling system failures, without requiring intervention of active systems. That would be a significant advance compared to current pressurized water reactors which have several active systems devoted to control the core loss of coolant. Two aspects should be analyzed carefully:

- The major part of the first containment barrier robustness (the layers coating the fuel kernel),
- The wish of designers to adapt the GEN-III third barrier to GEN-IV system, allowing for the VHTR physical features, especially their high thermal inertia, resulting in slow temperature changes during loss of coolant transients.

3/2 Reactor-related aspects

3/2/1 Fuel

Designers largely found the safety demonstration of the HTR/VHTR concepts on the resistance of the TRISO particle, which is assumed to be able to maintain its containment capacity up to a temperature of about 1,600°C, even though it is well known that the refractory layers of the fuel would become permeable to some fission products after a long irradiation period at this temperature. Designers aim therefore at guaranteeing that this temperature would never be reached whatever the accidental transient. In this respect, higher temperatures during normal operation in VHTRs (compared to HTRs) would involve a careful examination of the demonstration of the resistance of the fuel envelope, especially in the case of a primary loss of coolant transient. This need could lead designers to search for other coating materials (zirconium carbide for example), which would resist higher temperatures. Besides, if in the near future attaining high burn-up is not necessarily a design objective for an experimental, a demonstration, or even a prototype reactor, such an objective is intrinsically related to the more general objective of GEN-IV reactors which is to replace the reactors currently in operation or planned.

It would therefore be necessary to examine more specifically:

- Whether the fuel and reactor design possesses sufficient margins against the stresses to which the fuel is subjected both in normal operation and in incidental or accidental situations, and those which would lead to the release of a large quantity of the contaminants contained in the core fuel,
- The amount of radioactive elements which would be transferred to the primary circuit, to the reactor building and to the environment in the event of loss of particle robustness, and to deduce the appropriate measures for containment.

3/2/2
Neutronics

The layout and material composition of HTR/VHTR cores differ significantly from those of pressurized water reactors and fast reactor cores already in operation, with many heterogeneities.

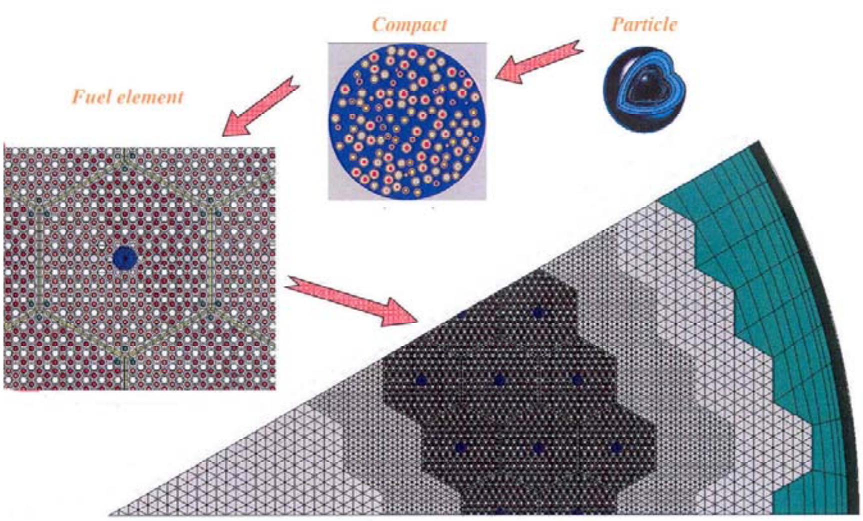


Figure 11:
Heterogeneities in a very high temperature reactor core.

This makes it necessary to develop and qualify specific calculation tools. The development of such tools and the mock-up experiment programmes needed to qualify them are considered by designers.

Besides, today there is no evaluation of tritium production by the irradiation of graphite, or even helium, in an HTR/VHTR type reactor. Tritium transfers in a coupled hydrogen production plant should also be investigated.

3/2/3

Materials

For certain safety-related components including those ensuring the core support, the concept of the VHTR implies the development of materials which can withstand very high temperatures (1000°C), much higher than those encountered in present power reactors or even in the fast reactors already in operation (550°C). Composite materials are proposed for the internal structures and the elements brought up to high temperatures. They undergo wide research and development programmes.

The design developed by AREVA would enable the reactor vessel and the intermediate heat exchanger, which constitute the second containment barrier, to be maintained at a moderate temperature in normal operation (400°C), but these components would have to withstand much higher temperatures in the event of an accident; ferritic steels already used for the reactor vessels of pressurized water reactors and chromium steels are the subject of research and development programmes, especially with the aim of evaluating their weldability.

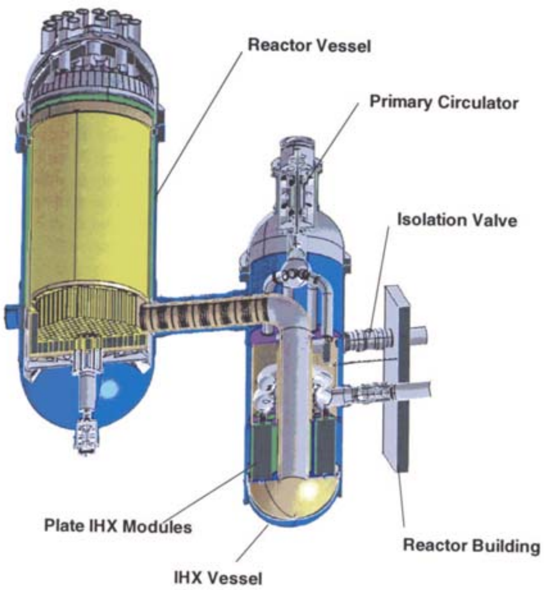


Figure 12:
AREVA very high temperature reactor project.

Concerning graphite, it would be necessary to complete the experience acquired with UNGG reactors in France and AVR concept in Great Britain: the combined impact of very high temperatures and irradiation largely remains to be studied, in particular to evaluate the dimensional variations of graphite under irradiation and the risks possibly induced on the core behaviour during normal operation and in the event of incidental or accidental transient. Research and development programmes have been undertaken by designers.

3/2/4
Passive Systems

Exploiting the high thermal inertia and low power densities (less than 10 MW/m^3, compared to about 100 MW/m^3 for PWRs and 300 MW/m^3 for Superphénix) of the HTR/VHTR concept, designers consider, for post-accidental situations, a passive residual power removal by conduction, radiation and natural water convection in various circuits. Experimental evidence of the actual capacity of the system to induce simultaneous natural convection in the circuits, in conditions representative of accidental situations could raise some difficulties.

Besides, the experience feedback from Phénix and Superphénix design (see section 4/2/4 below) suggests paying particular attention to the demonstration that no degradation of possible heat transfers by emissivity of structures would occur during the lifetime of the reactor.

More generally, evaluation of the efficiency of "lines of defence" based on passive systems will need in-depth consideration; this subject has not so far been discussed with designers.

3/2/5
Safety and reliability of the associated industrial processes

The advantage of VHTRs is largely based on the capacity of this type of reactors to supply high temperatures for industrial applications, chiefly the production of hydrogen. However, coupling to an industrial hydrogen production facility would require a careful analysis of possible interactions between the reactor and the hydrogen production facility (possible hazards). Though such an installation has been already settled in Germany and is underway in Japan with the HTTR, no sufficiently meaningful operation feedback exists. Even without an explosion, disruptions from the hydrogen production facility could also lead to significant load transients to be accounted for in the design basis of the reactor circuits. It would also be necessary to evaluate the consequences of such disruptions on the behaviour of the fuel in the core.

3/2/6
Other graphite-related risks

The "Wigner effect[1]", fire risk arising from the liberation of energy accumulated in graphite during irradiation, should be avoided in HTR/VHTRs, in view of target temperatures for these reactors (higher than 400°C).

[1] The Wigner effect played an important role in the fire which occurred on 10 October 1957 at the British Windscale reactor, Pile no 1.

On the contrary, for HTR/VHTRs, the risks associated with an accidental ingress of air into the primary circuit in the event of a break should be closely investigated. Such an event could indeed activate the disintegration of the graphite pebbles or "compacts" and thereby generate a widespread dispersion of fuel particles (TRISO), carbon 14 and chlorine 36. Besides, the graphite oxidation process at target temperatures (up to 1000°C) should be carefully analyzed; research and development are in progress on these aspects. Anyway, a supported demonstration that air ingress into the core of an HTR/VHTR could not lead to a big graphite fire has not yet been supplied.

The effects on the graphite of ingress of water into the primary circuit of an HTR/VHTR should also be studied.

3/2/7

Radiological protection

Helium is a gas with a very high spreading capacity, due to its low atomic weight. This feature must be taken into account in the design and operation of reactors, which should lead to impose a low threshold value on the fission product inventory in the helium of the primary circuit.

3/2/8

Fire and explosion risks related to hydrogen production

Concerning the fire and explosion risks originated in the hydrogen production plants, which could be coupled to VHTRs, the IRSN only knows functional diagrams showing a geographical separation between such plants and the reactors, the adoption of hillock separations or buried reactors. This aspect should be particularly examined depending on the amount of stockpiled hydrogen and the processes adopted for production. Should these plants be submitted to the safety rules of the chemical industry, the consistency of those rules to nuclear standards should be checked.

3/2/9

Security concern

A more in-depth examination of the risks mentioned above should contribute to identifying the sensitivity of HTR/VHTR reactors to acts of malicious damage. Designers and operators will have to define the appropriate design and operation provisions accordingly. Nevertheless, it can be pointed out that, even if the high thermal inertia of HTR/VHTRs, the existing wide operating margins preventing significant damage to the core and the fact that the coolant fluid (helium) is chemically idle are favourable factors, it would be necessary to take account of:

- The possible presence of a hydrogen production plant close to the reactor,
- The risks associated with the consequences of air and/or water ingress into the primary circuit, which are still to be examined.

Concerning hydrogen, a satisfactory resistance against acts of malicious damage can only be obtained by implementing appropriate measures both for the hydrogen production plant and the reactor itself. While an adequate separation of these two parts obviously would be beneficial, (distance, buried reactor, hillock separation, etc.), the likelihood of a drifting hydrogen cloud must be accounted for.

Limiting the possibilities of massive, provoked air or water ingress into an HTR/VHTR should be a design demand, systems (main circuits and auxiliary circuits) should be dimensioned and room architecture provisions defined accordingly: circuit redundancy, diversification of equipment items, and geographical separation of redundant circuits.

3/3

Fuel fabrication and processing related aspects

First, it should be noted that the consequences of deploying a new reactor generation on fuel cycle management strongly depend on the transition conditions established from the current NPP fleet to the new one. To date, different scenarios of transition towards a GEN-IV fleet have been investigated by *Electricité de France* and the French Atomic Energy Commission, but they have not yet been submitted to the IRSN.

The TRISO fuel for HTR/VHTRs takes the form of spherical particles of about 1 mm in diameter; the fuels studied are mainly uranium oxide or mixed uranium and plutonium oxide.

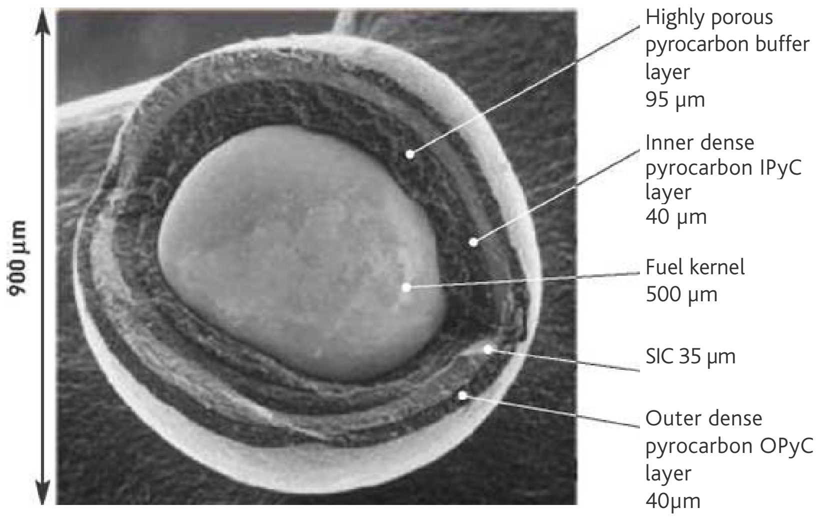


Figure 13:
Sectional view of a TRISO particle.
Photo credit: CEA.

Enrichment in uranium 235 would be less than 20%. Such enrichment can be achieved with available technologies and it is not challenging. Nevertheless, the fabrication of such a fuel on an industrial scale would require the construction of a dedicated facility. In terms of safety, the design bases of such a facility should be little different from those selected for current fuel fabrication plants.

Figure 14:
TRISO HTR fuel balls fabricated in the UO2 laboratory's GAIA facility at the Cadarache CEA centre.
Photo credit: CEA.

Potential fuel processing procedures after irradiation would include two phases: the first aimed at accessing the TRISO particles contained in the pebbles or "compacts"; the second aimed at separating the actinides from the fission products. The separation processes developed for GEN-IV reactors would aim at performing a combined extraction of actinides to be recycled, more appropriate to limit the risks of diversion of nuclear materials. The recycling would concern at least the uranium and the plutonium; present studies have developed along two paths: hydrometallurgical procedures (wet line) and pyrochemical process (dry line).

Concerning production of TRISO particles, mechanical processing (crushing, grinding, cropping) was studied in the USA by the Oak Ridge National Laboratory, as well as in Germany by FzJ (Forschungzentrum Jülich GmbH) in the 1960s, 1970s and 1980s; the maximum actinide recovery rate was quite small. Moreover, grinding can generate pollution of the graphite by the actinides and fission products. Studies of separation processes using combustion in a controlled atmosphere have demonstrated the impossibility of a total combustion of the graphite. They raise questions on the production of very fine dust, which could be very contaminating. A new method for producing fuel particles, at present proposed by the French Atomic Energy Commission, should make use of a high voltage pulsed-current of 10 to 20 kA (between 200 and 500 kV).

To remove the fuel nuclei from their refractory layers, a treatment using nitric acid cannot be proposed as the carbon and silicon carbide layers are not dissolved by this acid. Mechanical processing by grinding would involve the use of specific tools, on account of the abrasive nature of silicon carbide, and would not lead to a complete release of all the volatile and other fission products. Theoretically, another possible method would be the dissolution of the peripheral layers of the particle using soda or carbonates. This method has not been examined by the CEA as it presents risks of precipitating the plutonium, which is very challenging in terms of criticality risk prevention. The methods currently studied are as follows:

- The pyrochemical method, with elimination of the carbon and silicon carbide layers by attacking them with gaseous chlorine at 950°C, followed by pyrolysis in an oxygen atmosphere; the carbon/silicon carbide coat then volatilises in the form of carbon oxides and silicon chloride,
- The mechanical method with destructuration of the coating layers using pulsed-currents.

Finally, it must be emphasized that ways of reducing the volume of graphite waste from fuel processing are being studied.

In view of the above-mentioned arguments, the demonstration of the feasibility of a “closed” cycle for the HTR/VHTR concepts has not yet been completed to date, and important technological breakthroughs will in this respect be necessary.

It should also be noted that, in cycle facilities, the possibility of additional moderation from the water should be taken into account, at least for incidental operation.

Moreover, it will be necessary to evaluate the qualification level of the computational tool and chains adopted for criticality risk prevention studies. The experience available to date cannot be claimed as sufficiently representative of future calculation configurations. Nevertheless, it is important to note that this issue highly depends on the methods implemented, especially the possible need for processing large masses of “fissile” material. The analysis of processes should originate new needs in terms of calculation tool and experiments.

4/ Main safety, radiological protection and security issues identified to date for Sodium-cooled Fast Reactors (SFRs)

For sodium-cooled fast reactors, the IRSN has got a quite wide knowledge of safety questions since France has started developing the concept with Rapsodie, Phénix and Superphénix reactors. After launching the RNR 1500 project, France has been an active participant in the EFR project, mainly contributing with studies on the improvement of breeding capabilities, the reduction of core neutronics sensitivity to core coolant void (see below, paragraph 4/2/2), and the incineration of minor actinides. Today, in France, only the Phénix reactor is still in operation (until early 2009).

4/1 Safety objectives and approach

The document submitted by the CEA in March 2004 and prepared jointly with AREVA NP, does not mention any specific safety studies for SFRs. Nevertheless, in the straight line with the decision already adopted for the EFR, designers will be looking for ways to improve the safety of this concept to reduce the risk of a generalised core melting, and even, to "practically eliminate it". It may however be necessary to plan specific measures to contain a possible limited core melting (melting of either one or several assemblies), accounting for the additional risks which would result from a reactor vessel failure with an interaction of molten core materials and sodium; such measures (internal core-catcher for molten materials) have already been adopted for Superphénix and have been planned for both the RNR 1500 and EFR projects.

While awaiting documents from designers on the safety options for a Generation-Four SFR project, the IRSN planned to examine the safety orientations selected for the EFR project, established between 1988 and

1998, checking in particular their consistency with the general safety objectives fixed for the SFR project.

4/2
Reactor-related aspects

4/2/1
Specific risk inherent to metallic fuel

The use of metallic fuel (uranium, plutonium and zirconium alloy) is a variant option developed in the USA; at present it undergoes an experimental campaign for transmutation in the Phénix reactor (METAPHIX experiments). Such a fuel would require a specific reprocessing method (pyrometallurgy) which has not yet been developed in France. It should be emphasized that this fuel presents a risk of reaction with the steel in the (eutectic) compounds for the cladding, which reduces its melting temperature.

4/2/2
Neutronics effects in case of loss of coolant – the so-called “sodium void effect”

The risk of a sudden and sharp power increase in the event of a sodium void in some core zones (positive “sodium void effect”), by boiling, draining or the uncontrolled passage of a bubble of gas, is one of the main factors which determined adoption in France of several design solutions for sodium-cooled fast reactors. Measures have been taken by designers to avoid large-scale boiling of the sodium, gas being sucked into the core or the core voiding in presence of a main reactor vessel leak.

The problem of the risks associated with the “sodium void effect” has been re-examined after the unexpected automatic reactor shutdowns in the Phénix reactor, which occurred in 1989 and 1990, the origin of which remains widely unexplained. Nevertheless, it has been demonstrated experimentally that these automatic shutdowns could not be originated from insertions of positive reactivity due to any “sodium void effect”.

The RNR 1500 and EFR projects were developed with the view of designing reactor cores enabling positive “void effects” to be minimized as far as possible. This objective, which remained a major aim for the EFR project, has been adopted by the CEA for the SFR concept.

4/2/3

Risk of core meltdown

Considering the phenomena described above in section 4/2/2, core meltdown accident likelihood has been accounted for in the Phénix and Superphénix designs, leading to specific resistance requirements for the primary circuit (the main reactor vessel and its upper plug). Moreover, with the aim of reinforcing the Superphénix containment resistance in the event of core meltdown, in addition to settling a safety reactor vessel surrounding the main reactor vessel and a metallic dome above the plug, the reactor has been equipped with a core-catcher for molten materials inside the main reactor vessel, an option which has also been adopted for the RNR 1500 and EFR projects. The Chinese CEFR reactor is also equipped with such a device. The BN 350, BN 600 and Monju reactors do not have any core-catcher.

In particular for Superphénix, the risks of core meltdown have been widely studied and numerous experimental programmes were mainly conducted in the Cabri and Scarabée facilities, located in the CEA's Cadarache Centre. Those programmes included:

Reliability studies for the emergency shutdown system and all the systems ensuring residual power removal,

Studies and tests on phenomena likely to engender partial melting in a fuel pin, on the consequences of an assembly melting or a generalised melting in the core, mainly the risks for criticality of molten materials containing fuel.

A few tests went on in the Cabri reactor and in the Silène facility after the decision -taken in 1997- to shut down Superphénix definitively and supplied useful information. The IRSN has drawn up a summary document on all these tests.

According to the elements indicated above, designers should demonstrate that, for future SFRs, a partial melting of the core is not likely to degenerate into a generalised melting. This could allow reducing the presently very stringent containment design requirements (mechanical resistance of the primary circuit, capacity of the core-catcher, etc.). In this respect, advanced studies were carried out retrospectively for the Phénix reactor, on account of the absence of such a core-catcher for molten materials. Analyses, mainly those concerning the risks for criticality, are quite complicated and require adoption of advanced neutronics, thermal and mechanical computation models and schemes, the development, validation and certification of which should be pursued if necessary. If the above-mentioned orientations were confirmed, a qualitative breakthrough would need to be accomplished for future SFRs on these subjects.

Moreover if, despite those arguments, a generalised core melting was to be included in the SFR containment design bases, several specific items would have to be investigated both theoretically and experimentally, through appropriate R&D programmes, including the possibility for:

Avoiding the formation of critical zones within molten materials containing fuel,

Avoiding significant energy releases in the reactor (by expansion of fuel or sodium vapour bubbles),

Post accidental cooling of molten materials and the most suitable design solutions for an internal core-catcher (in the main reactor vessel).

4/2/4

Sodium-related risks

After the commissioning of Phénix and Superphénix, it was deemed necessary to take into account the risks associated with sodium pulverisation following a pipe breach for the design of rooms, because that could quickly generate quite high overpressures as well as the heating of the walls giving rise to water leaks from the concrete and to a hydrogen combustion or explosion. Appropriate technical solutions have been introduced.

The IRSN has been actively involved in acquiring knowledge on sodium fires and has participated in the development of calculation tools in this domain. The experimental area explored was sufficiently wide to allow obtaining an in-depth phenomenological knowledge of the combustion and supporting the qualification of calculation tools. Nevertheless some aspects need further investigation, especially the risks of sudden or deferred ignition of pulverised sodium when at "low" temperature (from 100°C to 300°C) and the risks of fire rekindling in rooms should air blow back on aerosols of unburned sodium.

The risks of sodium-water reaction in a steam generator must also be accounted for: they can lead to significant dynamic loads in the secondary circuits and, if the accident is not very rapidly brought under control by appropriate automatic systems, it can start an uncontrolled sodium-water-air reaction up. In this respect, significant improvements have been implemented in Phénix and Superphénix, mainly aimed at allowing an early detection of defects in steam generator tubes. Nevertheless, experience feedback from the British PFR reactor and the Phénix reactor in France shows that these detection systems are quite difficult to operate efficiently (the seriousness of the event which occurred in 1987 on the British PFR reactor, which led to the rupture of 40 steam generator tubes, is due to a temporary inhibition of a detection system, as a consequence of malfunctions). Moreover, on account of the steam generator design, the identification of a possible tube failure would have been a challenge for Superphénix.

Concerning the SFR concepts without intermediate circuits, i.e. in which the primary sodium which may contain fission products after a cladding rupture would directly exchange heat with the water of the electricity production system, the IRSN does not possess enough elements to evaluate the impact of the technological solutions proposed (double wall steam generator, etc.) on safety. Other solutions could be investigated by designers (intermediate circuits containing a fluid less reactive with water than sodium, etc.)

4/2/5

Passive systems

In the straight line of the options already adopted for reactors such as Phénix and Superphénix, the design of an SFR should call on passive systems, mainly for residual power removal. However, while the possibility of a natural convection has been verified for some individual circuits, no experimental evidence has ever been acquired for all the circuits as a whole; only a computation-based demonstration has been obtained. The possibility of performing an overall check by tests in satisfactory safety conditions should be examined.

The likelihood of implementing generalised natural convection in the case of a "loop" SFR should also be investigated.

Besides, the design experience for Phénix and Superphénix has shown that determining the residual power level and guaranteeing its removal is not easy. This has supported either addition of supplementary residual power removal systems (as it was the case for Superphénix), or downscaling the operating power (Phénix). Moreover, the emissivity of the structures can significantly change in time. The possibility to maintain a sufficient radiation thermal exchange over time should therefore be checked throughout the lifetime of SFRs. This would be a necessary condition for the demonstration that active systems are not necessary and can be avoided.

4/2/6

Structure Inspectability

Due to the presence of sodium in the SFR circuits, in-service inspection of some components is quite hard; this is especially the case for the reactor internal core support structures. This difficulty was highlighted for the Phénix reactor in the 1990s, in the framework of the CEA's request to stretch out the operation of this reactor after the unexplained emergency shutdowns by negative reactivity. An automated and innovative remote control device, operated from outside the main reactor vessel, was then developed very quickly by the CEA and provided sufficient assurance to allow the reactor a few more operation cycles.

Developments were also made for Superphénix (automated devices allowing remote control of the main reactor vessel -MIR device- and steam generator tubes).

Inspectability of structures is undoubtedly an important field for progress in both "integrated" and "loop" SFRs (according to the information available on the Monju "loop" reactor, in particular).

Thus, designers and associated research and development organisations should undertake actions in two fields, namely the possibility for improving the conditions of in-service inspection and repair by judicious design choices, and the development of appropriate devices for in-service inspection (non destructive sodium tests, etc.).

Finally, it must be pointed out that the IRSN has already drawn CEA's attention to the interest of taking advantage from Phénix dismantlement to set up a specific inspection programme aimed at characterizing the state of the major structures and components, which won't have been checked in depth during the 35 years of operation of the reactor.

4/2/7

Security concern

The general arguments put forward in section 3/2/9 concerning HTR/VHTR security concern can be extended to SFRs.

The very high sensitivity of the SFR concept to acts of malicious damage is due to the large amounts of sodium stored in the reactor: sodium is actually a metal which reacts chemically with air and water. The prevention of acts liable to initiate a generalised sodium-water-air reaction is thus an important challenge for the liability of the concept. This should induce designers to take a special care to the systems involved in fundamental safety functions, mainly those in charge of the chain reaction control and the residual power removal, in terms of redundancy, separation and diversification.

Moreover, an exhaustive study of the risks engendered by the presence of sodium as a coolant would be necessary (likelihood of chemical reactions with other materials than water and air with energy delivery).

Besides, the design of the primary circuits of SFRs provides several regions free of sodium, filled with gas (generally argon). If future SFR design options do not allow the power increase effects in case of gas transit in the core (positive "void effects") to be eliminated, the possibility that the gas could be injected into the core will have to receive special care.

Finally, according to the future SFR design's sensitivity to a total and prolonged loss of electrical sources which could lead to a temperature increase in the core support structures, appropriate resistance of the electrical sources to acts of malicious damage will need investigation.

4/3 Fuel fabrication and processing related aspects

The mixed fuel (uranium and plutonium) of SFRs can assume an oxide, carbide, nitride or metallic form. The cladding materials are mainly stainless steels.

The fabrication of fuels in mixed uranium and plutonium oxide form benefits from the experience of manufacturing the fuel for the Rapsodie, Phénix and Superphénix reactors at the CEA’s Cadarache Centre, as well as the manufacturing of fuel for PWR reactors, mainly in the MELOX plant of Marcoule. This fabrication is carried out in glove-box dedicated units ensuring sufficient, but limited, protection for workers against ionizing radiation (neutrons and gamma). The predictable change of the isotropic characteristics of the plutonium, unfavourable in terms of radiological protection of operators, as well as the recycling of very active minor actinides (americium, curium), could force equipment remote maintenance, which would have a significant impact on the design of manufacturing plants and on the doses received by operators.

Concerning the fabrication of fuel in the form of carbide or nitride, no industrial experience has been achieved. Apart from the risks inherent to the processes adopted for handling and manufacturing the fuel, it is worth mentioning the high reactivity of carbides with water, oxygen and, to a lesser extent, nitrogen; that would require these compounds to be manipulated in an inert environment (when they are in solid form).

Processing of the fuels in mixed oxide form can be based on the experience feedback of processing similar fuel for the Rapsodie and Phénix reactors, performed at the La Hague plant or in the pilot workshop at Marcoule (APM). Although the industrial feasibility of the whole process has been demonstrated, it remains that a new factory would be necessary to process the irradiated fuel of a whole fleet of fast reactors. In terms of safety, the design bases of such a factory should show only few differences from those of the present factories at La Hague.

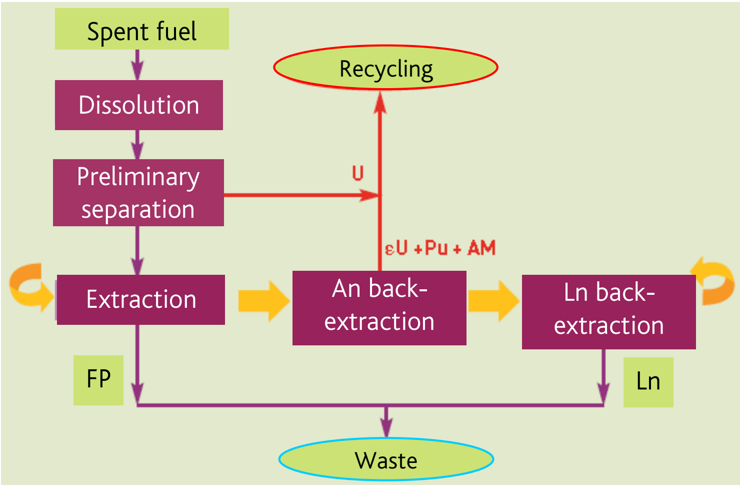


Figure 15:
Schematic diagram of the Ganex process studied by the CEA for the grouped extraction of actinides in the framework of studies on generation-four reactors closed cycle (FP: fission products; An: actinides; Ln: lanthanides).
Photo credit: CEA

Concerning the post irradiation processing of the fuel in carbide or nitride form, the numerous laboratory experiments performed throughout the world and chiefly the Indian pilot experiments (processing several hundred kilograms of materials) apparently have not revealed any major solubility problems during fuel dissolution phases. It seems therefore possible to rule out the risks of criticality engendered by the presence of undissolved components in the processing equipment. Nevertheless, concerning carbides, the formation of carboxylic acid, due to the presence of residual carbon in the solution, could activate the piling-up of complex compounds, quite difficult to extract. These compounds could combine with a fraction of the plutonium and thus limit the efficiency of the whole process. Concerning the nitrides, the production of ammonium ions as by-products of the process should be allowed for as it could generate an explosion risk. Moreover, suitable trapping devices could have to be developed to trap Carbon 14.

Concerning the prevention for criticality risks, it should be noted that some minor actinides have much more constraining neutronics features than plutonium or uranium, e.g. the minimum critical masses of curium 245 and americium 242 are a few tens of grams, as compared to 510 grams for plutonium 239. The minimum critical masses could be even lower for some minor transuranics piling-up at very high burn-ups; moreover, at present, significant uncertainties exist on the minimum critical masses of these actinides. Therefore, it would be necessary to validate calculation tools for minor actinides environments, knowing that todate few critical experiments are available. Finally, care should be taken to ensure that the composition of the fuel after irradiation can be correctly predicted in order to determine the most reactive environments to be selected for studies relative to the various stages of fuel cycle.

Some useful data on carbide and nitride type environments (crystalline densities, etc.) are available, as well as results from criticality calculations performed for laboratories, assuming that the characteristics of the carbide and nitride type and oxide type environments are quite similar. Nevertheless, additional studies would be necessary to cover the whole fuel cycle in the framework of an industrial deployment of reactors using fuel based on carbide and nitride. The CEA experimental facilities at Valduc could be used when the time comes to obtain complementary data to qualify the calculation diagrams.

5/ Main safety, radiological protection and security issues identified to date for Gas-cooled Fast Reactors (GFRs)

In view of the few indications available on the GFR design, this document only provides a preliminary evaluation of the safety-related issues which should be addressed, in the present state of knowledge. It is clear that significant differences could result from the choices made by designers as regards the purposes of the reactor (production of electricity or heat, combined cycle, coupling to a chemical production unit, etc.), its overall design and the fuel selected.

5/1 Safety objectives and approach

An overall safety approach for GFRs is presented in the CEA's March 2004 letter mentioned above (section 3/1), together with the main "generic" safety options adopted. In view of the very high power density of a GFR core compared to a VHTR core, recourse to active safety systems to ensure core cooling in accidental situations seems a priori unavoidable. Besides, in the CEA's document, it appears that a total core meltdown might not be retained by designers as a design basis for the containment.

5/2 Reactor related aspects

The neutronics calculations for GFR reactor cores imply the development or improvement of nuclear data relative to current fast reactors, notably as regards the refractory materials, graphite and minor actinides. These data should be acquired through development and experimental programmes planned in particular in the CEA Masurca critical mock-up located at Cadarache.

Concerning GFR fuel, as indicated in section 2/, several options are being developed by the CEA, with the objective of reaching a resistance equivalent to that of TRISO particle fuel and limiting at about 1,600°C the maximum temperature reached in case of depressurization, an accident situation particularly severe for this system.

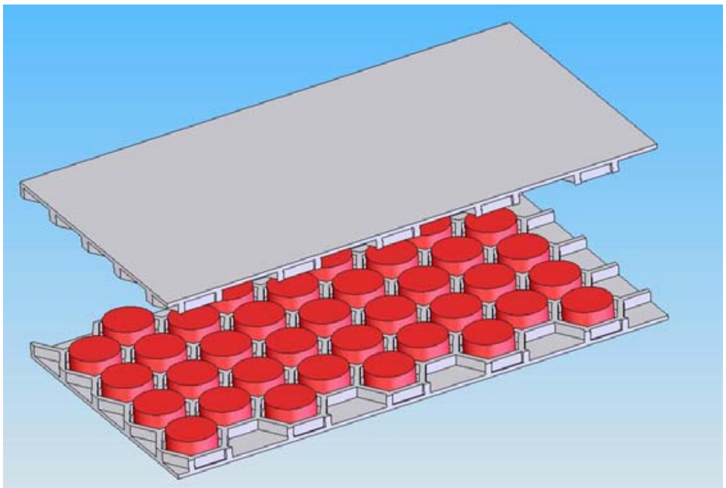


Figure 16:
Example of fuel developed for GFRs.

It should be noted that the power density of a GFR core (50 to 100 MW/m^3) would be more than ten times higher than a VHTR reactor core, with a much smaller reactor thermal inertia.

Therefore, several specific measures are studied by the CEA, especially setting up an intermediate containment shell under helium pressure around the reactor vessel (between 5 and 20 bar depending on the options), completed by active systems for core cooling. Setting up a containment shell under pressure around a reactor would constitute an innovation requiring a very detailed safety analysis.

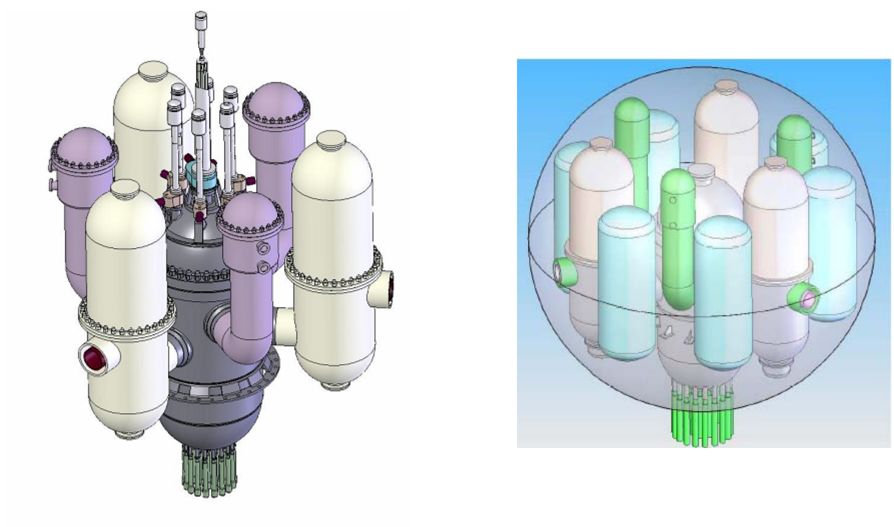


Figure 17:
Schematic diagram of a gas-cooled fast reactor (reactor vessel, components vessel, intermediary containment).

On the other hand, by design (use of a gas for heat removal from the reactor core), GFRs should be slightly affected by the coolant "void effect", thus eliminating several total core meltdown initiators retained for SFRs. Thus, the main initiators would be related to breaches, as well as to failures of the residual power removal systems, which could generate a very sharp temperature increase, damageable to the fuel. The reliability of the residual power removal systems (active and passive ones) and the temperature resistance of the fuel are, at a first sight, major safety issues for GFRs, mainly if the designer aims at "practically eliminating" a generalised core meltdown. The qualification of active systems functioning at high temperatures seems very challenging.

The risks associated with water ingress into the primary circuit, in particular the risk for a power increase due to a neutron moderation effect, with a sharp increase of the power, are still to be assessed.

As regards the materials, according to the design choices adopted, a specific aspect could concern the risk of embrittlement for components subjected to a considerable flux of fast neutrons and the risk of a sudden rupture of equipment also subjected to a significant pressure (70 bar).

Concerning the resistance of GFR reactors to acts of malicious damage, on account of the specific safety risks of this concept due to the high power densities in the core associated with a primary fluid of low calorific capacity, design measures (redundancy, diversification, geographical separation) should be implemented to limit as far as possible the risks for the core coolant being affected at the level of the main components (main reactor vessel, intermediate containment shell, piping) and the primary circuit pumps.

5/3 Fuel fabrication and processing related aspects

The mixed (uranium and plutonium) fuel of GFR reactors could be in the form of carbide or nitride oxide. The cladding materials considered are ceramic. Concerning the studies conducted by the CEA, the reference fuel is today in the form of carbide and the reference ceramics is silicon carbide.

Concerning fuel fabrication and processing, several of the considerations developed in sections 3/3 and 4/3 also apply here.

6/ Waste production and management in the framework of the deployment of a Generation-Four reactor fleet

The question whether deploying a fleet of GEN-IV reactors may have a significant impact on the production and management of radioactive waste includes several aspects. The two criteria put forward today are reducing the harmfulness of waste and minimizing their volume.

Concerning harmfulness, the main expected gains are generally expressed by designers in terms of potential "radiotoxicity[1]" decrease and are related to the reduction by transmutation of the spent fuel content of plutonium and minor actinides or of residual waste, after reprocessing. The GEN-IV reactors which are the subject of this document have advantageous features in this regard since they are all able to burn plutonium and/or actinides out, but at a variable extent. It should nevertheless be emphasised that a significant reduction in the dosimetric impact for the storage of waste and/or spent fuels in deep geological formation cannot be expected. All the evaluations performed to date (European EVEREST[2] and SPA[3] exercises, "dossier 2005 Clay" by ANDRA) show that the impact is mainly due to long-lived fission and/or activation products (mainly, iodine 129 and chlorine 36), as the actinides remain confined over very long periods in a small volume of rock surrounding the stored waste packages. Even adopting very unfavourable assumptions (transfers through fractures, low retention properties of the rock), the dosimetric impact of actinides remains of the same order of magnitude as for fission products, and it is attributable to the uranium 238 in secular equilibrium concentration with its daughters, and not to plutonium and minor actinides.

[1] Radiotoxicity: Sum of the products of the activity of each radionuclide contained in the waste by the dose coefficient (Sv/Bq) of the radionuclide which indicates the effects on man.

[2] EVEREST (EValuation of Elements Responsible for the effective Engaged dose rates associated with the final STorage of radioactive waste): European project for modelling storage site for reprocessing waste (vitrified wastes, enclosures and end-pieces, etc.) in three types of geological formation: clay, granite and salt.

[3] SPA (Spent fuel disposal Performance Assessment): European project devoted to modelling a storage site for spent fuels in deep geological formation.

Reducing the harmfulness of waste and spent fuels (if the latter were to be stored), would require the new reactors to produce less iodine 129 (and, to a lesser extent, less chlorine 36). Concerning iodine 129, the GEN-IV reactors do not appear as significantly advantageous compared to the current reactor fleet for the same installed electrical power. Moreover, chlorine 36, the generation by activation of which depends on the graphite component fabrication mode (impurity content, addition of chlorine through the purification process, diverse pollutions during fabrication) and on the irradiation history (radial or axial position in the core, neutron spectrum and fluxes, radioactive decay, incidents), shows a trend toward an increased production compared to current reactors if the graphite component fabrication specifications do not impose drastic thresholds on the content of relevant impurities. In this respect, industries are searching for graphite purification processes which avoid the use of chlorine (fluorine based processes). As regards the graphite in the components ("compacts" or pebbles, blocks used as moderators or reflectors, cladding containing silicon carbide and carbon, matrix containing nitrogen component, carbide or nitride based fuels), it will be able to generate, by neutron activation, carbon 14 (the radioactive period of which is 5,730 years) in much larger quantities than for the operating reactors. This radionuclide is quite mobile in natural environments and the capacity of the storage site selected for the corresponding waste to ensure a satisfactory containment over several tens of thousands of years should be examined.

Therefore, the capacity of GEN-IV reactors to significantly reduce the long-term dosimetric impact of waste and fuels to be stored in deep geological formations, does not supply convincing arguments for the development of such reactors. Moreover, it is known that the operations of separation and fabrication of the fuel from separated products would be complex. From a strictly technical point of view, a real assessment should be carried out, taking into account the doses received by operators and the waste produced by the above-mentioned operations.

Concerning the reduction of the volume of spent fuels and waste produced, the very high burn-ups of the new reactors should, in principle, reduce the quantities of spent fuel to be unloaded and, consequently, the amount of waste from reprocessing. However, this conclusion assumes implicitly that the adaptations of the processing procedures necessary to accommodate the characteristics of the unloaded elements (new claddings, carbide matrix) would not increase the amount of process waste and waste generated per unit mass of processed fuel. It would also be necessary to check that the intrinsic containment properties of packages and fuels remain sufficiently high to obtain an overall gain in terms of short and long-term waste management, as a result of the smaller volumes produced. Today the state of research does not allow formulating a judgment on all these issues. Besides, the reduction in the volume of spent fuels and waste to be stored does not necessarily generate a proportional reduction of the depot area of a geological storage site, which is strongly dependent on the heat production associated with waste. In a qualitative approach, the reduction in the amount of minor actinides to be stored would not allow lower thermal emissions after a few centuries than the fleet of reactors in operation. Nevertheless, it will be necessary to make sure that the short and medium-lived fission products contained in the new waste can decrease within a sufficiently short time to reach, while they are stored, criteria of thermal emissions compatible with their introduction into the geological formation.

In conclusion, at this stage, it is hard to evaluate whether GEN-IV reactors could provide any significant profit to waste management. Nevertheless, it is worth emphasizing that the actual advantage can be provided by the management of nuclear materials in a "closed" cycle, rather than by the features of the GEN-IV reactors considered. Such a management could indeed avoid spent fuel storage, which would facilitate the long-term safety demonstration of a geological storage installation, even if such a demonstration does not seem impossible should these fuels need to be stored. These considerations also apply to the existing stocks of depleted uranium. These stocks, of low activity today, will become active and highly productive of radon once the secular equilibrium between uranium 238 and its daughters is established, just like very rich deposits which cannot be exploited without extensive radiation protection measures; hence the interest in using these stocks.

These considerations show that the analysis of the potential of GEN-IV reactors to improve the management of radioactive waste cannot be dissociated from all the elements of decision relative to the exploitation of the energy potential of nuclear materials.

7/ Conclusion - Actions undertaken and planned by the IRSN

In conclusion, only the helium-cooled HTR/VHTR (High or Very High Temperature Reactor) and the SFR (Sodium-cooled Fast Reactor) concepts appear to be at a sufficient level of maturity to consider the construction of a prototype in France by 2020. Concerning GFRs (Gas-cooled Fast Reactors), in the framework of a progressive approach, only the construction of a great flexibility experimental reactor is conceivable, which would, for example, allow individual testing of the performances of some sub-assemblies and components (fuel and core, components, gas turbine, etc.)

A precise definition of the safety approach for the different types of GEN-IV reactors would require the design studies to be further advanced, especially in the identification of the incidents and accidents to be dealt with. In this respect, the IRSN has already undertaken discussions with designers and partner organisations on a number of general aspects. In this framework, the role which could be assigned to the Probabilistic Safety Analysis (PSA) and its reliability, especially in case of recourse to passive systems, will be discussed.

Concerning the helium-cooled HTR/VHTR concept, technical discussions have been undertaken between the IRSN and AREVA NP on the ANTARES project, on the basis of a protocol signed on February 2005, detailing the framework of the discussions and their non-committing character. Several specific actions cover the technical aspects shared by HTR/VHTRs and GFRs.

Concerning SFRs, discussions were undertaken in 2006 between the IRSN and the French Consultative Group for Safety (GCSF), including the Nuclear Energy Directorate (DEN) of the CEA, AREVA and Electricité de France (a GIF “mirror” group). The IRSN is reactivating its own internal main competencies, in order to have fundamental knowledge and tools at its disposal, especially in the fields of accidents leading to fuel damage and the risks related to sodium. The neutronics of future SFR cores may cause problems, as SFR cores are significantly different from cores of reactors in operation ("heterogeneous cores" with a large amount of minor actinides); on these open subjects, the IRSN plans to rely on a specialist competency group.

Concerning GFRs, the CEA is pursuing developments in the perspective of the REDT project, a low power experimental reactor. The IRSN and the GCFS also undertook consultations on this subject in 2006.

Apart from the discussions, participations and involvements mentioned above, the IRSN has selected a few privileged strategic axes for short-term actions of study, research and development corresponding to GEN-IV safety items it considers as very important:

- The behaviour, in normal operating, incidental and accidental conditions, of refractory fuels for HTR/VHTR and GFR reactors, predictive calculation models, uncertainties and associated statistical analysis,
- The neutronics of the HTR/VHTR, SFR and GFR reactor cores, with the progressive setting-up of the competency group so as to acquire the necessary knowledge and calculation tools, and to be able to implement them as soon as sufficient data are available (for example the reduction of the "sodium void effect" in the SFR cores),
- The risks associated with water ingress into a GFR, and the implementation of the associated calculation models,
- The transfers of radioactive products out of refractory fuels and into the circuits in the event of an accident, for gas reactors,
- The risks of propagation of a local melting in SFR and possibly GFR cores, the risks of a return to criticality of a corium (molten core material) in the reactor vessel, the propagation modes of the corium inside and outside a reactor vessel.

The studies undertaken will allow the future identification of the needs for experimental programmes specific to the IRSN, to be performed either in mock-up facilities or in test reactors.

Concerning the aspects related to the fuel cycle of GEN-IV reactors, the IRSN will keep a technological watch on the development of the fuel fabrication and processing procedures. The possibility of participating in international working groups concerned with the safety and management of the cycles proposed for GEN-IV reactors will be examined. Experimentation and development of models cannot however be usefully defined as long as the cycle and material fabrication and reprocessing procedures have not been precisely described. On the other hand, the IRSN can already benefit from the existing knowledge concerning the scenarios of GEN-IV reactor deployment in order to determine the impact of different strategies on the use of nuclear materials and on the waste elimination methods and to go deeper into the considerations presented in this document.

Lastly, it is clear that, in the development of GEN-IV systems, security-related issues will have to be investigated at the design stage.