IRSN
INSTITUT
DE RADIOPROTECTION
ET DE SÛRETÉ NUCLÉAIRE
Enhancing nuclear safety

Overview of Generation IV (Gen IV) Reactor Designs

// Safety and Radiological Protection Considerations

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IRSN
//in brief

The French Institute for Radiological Protection and Nuclear Safety (IRSN) was founded by **Act No.2001-398** of May 9, 2001 enacted through **Order No. 2002-254** of February 22, 2002. This Order was amended on April 7, 2007 to take into account the **Act No.2006-686** of June 13, 2006 relative on transparency and nuclear safety. The IRSN is a public establishment that carries out both industrial and commercial activities. It is jointly supervised by the Ministers for Defence, Environment, Industry, Research and Health.

IRSN employs over than 1,700 specialists: engineers, researchers, doctors, agronomists, veterinarians, technicians, experts in nuclear and radiation risks.

The Institute performs expert assessments and conducts research in the following fields:

- nuclear safety;
- safety relative to the transportation of radioactive and fissile materials;
- protection of human health and the environment from ionizing radiation;
- protection and control of nuclear materials;
- protection of facilities and transports dealing with radioactive and fissile materials against malicious acts.

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Foreword

The Institute for Radiological Protection and Nuclear Safety develops research programs and conducts studies on nuclear and radiological risks. It is responsible for public service initiatives aimed at prevention and provides technical support to the public authorities in charge of ensuring nuclear safety and security, together with radiological protection. In fulfilling these various duties, the Institute is called upon to define its position on certain scientific and technical issues.

In line with its policy of transparency and its desire to make quality information available to all partners and stakeholders for use in developing their own views, the IRSN publishes "reference documents", which present the Institute's position on specific subjects.

These documents are drafted by IRSN specialists, with the help of outside experts if necessary. They then undergo a quality assurance validation process.

These texts reflect the Institute's position at the time of publication on its website. It may revise its position in light of scientific progress, regulatory changes or the need for more in-depth discussion to satisfy internal requirements or external requests.

This document may be used and quoted freely on condition that the source and publication date are mentioned.

We welcome your comments. These may be sent to the address given in the margin above and should include the reference to the relevant document.

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Preface

Any future decision to go ahead with the industrial construction of Generation IV nuclear power plants in France will be guided by a number of strategic considerations, among which safety performance will obviously be foremost.

A number of countries are currently conducting research to develop Gen IV reactors using various fuels and cooling systems. France, in particular, has concentrated its research efforts on the construction of a new sodium-cooled fast reactor. Within this context, IRSN considered that it should submit these various reactor designs to a thorough review from a purely safety perspective. The review was based on the premise that future reactors should match, if not outperform, last-generation pressurized water reactors in terms of safety performance, in accordance with general WENRA specifications and taking into account feedback from the Fukushima accident.

It is clear from this general review that whatever the reactor system considered, and notwithstanding the intrinsic advantages of each one, significant technological progress is required before any claim can be made that expected safety levels have been met.

I would like to thank Mr Jean Couturier, the principal author of the report, for this work which is both extremely thorough given the available data and a pleasure to read in spite of its complex subject matter.

I hope you find it pleasant and informative reading.

Jacques Repussard
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Introduction

1/1
Background

Many energy-futures studies[1] foresee uranium shortages in the twenty-first century, including some that assume effective energy-demand management, drastic limitation of greenhouse-gas emissions and strong "renewable energy" policies.

In this context, the US Department of Energy (DOE) launched the Generation IV International Forum (GIF) in the year 2000. Currently, the GIF has 13 members including France. Its first action was to select the six "Gen IV" nuclear reactor technologies (concepts) which are considered the most promising, both in terms of conserving uranium resources and with respect to the following criteria:

- reduction in radioactive waste, especially long-lived high-level waste;
- safety improvements;
- robustness with respect to non proliferation and malicious acts;
- economic competitiveness.

The six chosen reactor concepts are as follows:

- Sodium-cooled Fast Reactors (SFR);
- Gas-cooled Fast Reactors (GFR);
- Lead-cooled Fast Reactors (LFR);
- Molten Salt Reactors (MSR);
- High or Very High Temperature Reactors (V/HTR);
- SuperCritical-Water-cooled Reactors (SCWR).

[1] For an example (in French) on these subjects, see the CEA's press file on the Internet « *Quatrième generation: vers un nucléaire durable* » (Fourth generation: towards sustainable nuclear power), dated March 31, 2010.

According to the studies cited above, industrial deployment of these new designs could occur towards the middle of the twenty-first century, following initial operating experience on experimental reactors, demonstrators or prototypes.

In parallel, France has demonstrated its commitment to developing fourth-generation reactors, in particular *via* the target of commissioning a fourth-generation prototype reactor in 2020, which the French President announced in January 2006, in line with the French Energy-Policy Act 2005-781. This target is closely linked with objectives concerning the sustainable management of radioactive materials and waste associated with French Act 2006-739 dated June 28, 2006, which specifies that the industrial prospects for new generations of reactors (including accelerator-driven systems) be assessed in 2012, regarding their ability to separate and transmute long-lived radioactive isotopes, so that a prototype facility can be commissioned before December 31, 2020.

During its session on December 20, 2006, the French Atomic Energy Commission directed French industry towards sodium- or gas-cooled fast reactors, in particular with regard to the objectives of conserving uranium resources and reducing waste (*via* the ability to burn plutonium or to produce it from uranium-238, and the ability to transmute minor actinides such as americium and curium). In 2009, the work centered on SFRs. This choice seems to be mainly based on the maturity of the SFR concept, available know-how and consistency with the strategic national objectives of closed fuel cycles and long-lived-waste management. A prototype SFR project (the Advanced Sodium Technological Reactor for Industrial Demonstration, ASTRID) is now being designed under CEA leadership, in partnership with AREVA and EDF.

The timetable for the first phase of the ASTRID project is as follows:

- 2008-2012: design and research by the Project regarding baseline options and alternative options; discussions with the French Nuclear Safety Authority (ASN) and IRSN;
- 2012: submittal of a safety orientations report;
- 2014: submittal of a safety options report.

1/2 Purpose

This document is a follow-on to the one that IRSN published on the same subject in 2007, Reference Document [1]. Its purpose is to provide an updated overview of specific safety and radiological protection issues for all the reactor concepts adopted by the GIF, independent of their advantages or disadvantages in terms of resource optimization or long-lived-waste reduction. In particular, this new document attempts to bring out the advantages and disadvantages of each concept in terms of safety, taking into account the Western European Nuclear Regulators' Association (WENRA) statement concerning "safety objectives for new nuclear power plants" issued in November 2010, Reference Document [2]. Although the WENRA objectives target third-generation nuclear reactors (and should be revised before 2020 as stated in [2]), it still seems useful to take these objectives into consideration given the lack of specific documents for fourth-generation reactors[2]. It should be noted that, at the current (early) stage of considerations on the safety requirements for fourth-generation reactors, it is generally stated that fourth-generation reactors must be at least as safe as third-generation reactors, and should therefore at minimum comply with the objectives in [2], where these objectives are relevant. Using an identical framework for each reactor concept, this summary report provides some general conclusions regarding their safety and radiological protection issues, inspired by WENRA's safety objectives and on the basis of available information. Initial lessons drawn from the events at the Fukushima-Daiichi nuclear power plant in March 2011 have also been taken into account in IRSN's analysis of each reactor concept.

In general, security aspects have not been covered in this document. Some general considerations have been mentioned in Reference Document [1] and other very specific comments have been added in this document.

The assessment performed includes all knowledge from IRSN's international studies and research, in particular in the context of bilateral collaborations (with Russia, China and Japan) and the following projects funded by the European Commission in the context of the Seventh EU Framework Programme for Research and Technological Development (FP7):

- V/HTR:
  - RAPHAEL (2005-2010) and ARCHER (2010-2013): studies supporting the development of the V/HTR,
  - EUROPAIRS (2009-2011): studies concerning the coupling of a V/HTR with an industrial facility;

[2] Section 5 of International Nuclear Safety Advisory Group Report 10 (INSAG 10), issued by the IAEA, covers defense-in-depth for future nuclear power plant projects, including "advanced reactors", which correspond to innovative reactors, but it dates from 1996. INSAG 12 ("75-INSAG-3 Rev. 1"), from 1999, refers to "future nuclear power plants" in Section 3.3.10, but with primary emphasis being placed on water-cooled nuclear reactors. In 2008, the GIF issued the document cited as Reference Document [3].

- GFR:
  - GCFR (2005-2009) and GoFastR (2010-2013): studies on the GFR;
- SFR:
  - CP-ESFR (2009-2012): studies on the SFR;
- projects related to multiple designs:
  - SARGEN IV: this is a new project launched for 2011-2013, with the aim of producing a safety-analysis framework for the various fourth-generation designs and identifying the R&D axes to pursue for SFRs, LFRs and GFRs. This project is led by IRSN.
  - ASAMPSA2 (2008-2010): development of a European methodology for Level 2 Probabilistic Safety Analyses (PSA2) regarding releases into the environment in the event of severe accidents (the project concerns second-, third- and fourth-generation reactors). This project is led by IRSN.
  - THINS (2010-2014): computer-modeling resources to support innovative reactors.

The assessment also includes information that IRSN has acquired through the various working groups in which it participates at the international level, for example, the group that performed a study for the DOE to assess the "source term" (a characterization of releases into the environment under accident conditions) for SFRs and an evaluation of the knowledge required to assess the safety of V/HTRs.

It should be noted that, in the context of the European Sustainable Nuclear Energy Technology Platform (SNETP, www.snetp.eu), IRSN has led a number of discussions aiming to specify R&D axes, in particular for fourth-generation "systems[3]". Three documents should be mentioned:

- regarding systems using fast-breeder reactors — SFR, LFR, GFR and ADS (advanced driven systems, *i.e.* accelerator-driven reactors) — and V/ HTRs for cogeneration[4]: the document "SNETP — Strategic Research Agenda, May 2009"[5];
- regarding SFR, GFR and LFR systems: the document "SNETP — ENSII — The European Sustainable Nuclear Industrial Initiative — A contribution to the EU Low Carbon Energy Policy — The Demonstration Programme for Fast Neutron

[3] Reactors and associated fuel cycles.

[4] Combined heat and power for industrial processes.

[5] This document covers Gen II, Gen III and Gen IV reactors combined with their fuel cycles.

Reactors — Concept paper — October 2010"; this document uses SFR as a baseline design, with two alternative concepts: LFR and GFR;

- regarding MSR systems: the document "SNETP — Strategic Research Agenda — Annex: Molten Salt Reactor Systems, final draft, November 2011".

The safety questions raised in the remainder of this document are identified as issues requiring further R&D in these SNETP documents. They also cover the need to develop modeling and instrumentation resources, subjects which are barely mentioned in this document.

Furthermore, SNETP has issued a document regarding the use of thorium (for example in MSRs): "SNETP — Strategic Research Agenda — Annex: Thorium cycles and Thorium as a nuclear fuel component, January 2011".

1/3
Possibility of taking into consideration WENRA's safety objectives for new nuclear power plants

As mentioned in Section 1/2, WENRA has recently adopted a set of general safety objectives for new projects for third-generation nuclear power plants. These objectives are presented in the document "WENRA Statement on Safety Objectives for New Nuclear Power Plants", dated November 2010, Reference Document [2]. The seven objectives, which make explicit reference to the defense-in-depth safety approach, are related to:

- normal operation, abnormal events and prevention of accidents (O1);
- accidents without core melt (O2);
- accidents with core melt (O3);
- independence between all levels of defense-in-depth (O4);
- safety and security interfaces (O5);
- radiation protection and waste management (O6);
- leadership and management for safety (O7).

These objectives are detailed in the Annex 1 to this document. Three requirements should be highlighted:

- Ensure that "accidents without core melt induce no off-site radiological impact or only minor radiological impact (in particular, no necessity of iodine prophylaxis, sheltering nor evacuation)".
- "Accidents with core melt which would lead to early or large releases have to be 'practically eliminated'[6]".
- "For accidents with core melt that have not been practically eliminated, design provisions have to be taken so that only limited protective measures in area and time are needed for the public (no permanent relocation, no need for emergency evacuation outside the immediate vicinity of the plant, limited sheltering, no long term restrictions in food consumption) and that sufficient time is available to implement these measures".

In addition to the comments made in Section 1/2, regarding the limitations inherent to attempting to position the various fourth-generation concepts with respect to the WENRA objectives [2], the following three points should be noted:

- prevention of incidents and accidents (Objective O1) — which corresponds to the first two levels of defense-in-depth — is based on various aspects which, in the current state of research and project design, are not well defined (such as system and loop architectures, equipment reliability, human-machine interfaces, operating procedures, incident and accident procedures). Nevertheless, it is possible to assess the concepts, in particular in terms of complexity, ease or otherwise of operating the reactor, and intrinsic characteristics that are helpful or less helpful (such as reactor thermal inertia or operator grace periods);
- Objectives O2 and O3, pertaining to accidents with or without "core melt" are not relevant for all concepts, in particular V/HTR and MSR (for the latter, the fuel may be liquid under normal operation). The concept of a severe accident[7], which is associated with core melt for second-and third-generation reactors, has yet to be defined for certain fourth-generation reactors;
- technological mastery of a concept, associated with significant experience feedback, is an essential condition for safety; the six fourth-generation concepts are highly disparate on this level.

[6] According to the December 2009 WENRA document cited in Reference Document [2], and following IAEA Document NS-G-1.10, an event can be considered as "practically eliminated" if it is physically impossible or if it can be considered with high degree of confidence to be extremely unlikely to arise.

[7] In the sense of the idea of "severe accident" in the IAEA glossary, which implies "significant core degradation". This is does not necessarily involve core meltdown.

With regard to this last point, it should be noted that it was possible to produce an initial specification of the safety objectives for the EPR in just a few months, in 1993, in a Franco-German context. This was the case because PWR concept had been subject to deep analyses by the various parties involved, including IRSN, since the beginning of the nuclear power program in 1973 and the EPR project was positioned as an evolution of the reactors currently in operation or under construction and as a plant series for the early twenty-first century. The situation is significantly different for some of the six fourth-generation concepts chosen by the GIF.

It should be noted that Reference Document [3], produced by the GIF, suggests particularly the following general objectives:

- *"Generation IV nuclear energy systems operations will excel in safety and reliability"*
- *"Generation IV nuclear energy systems will have a very low likelihood and degree of reactor core damage".*

In conclusion, a (brief) safety analysis framework has been consistently applied to each of the fourth-generation concepts, examining the following aspects one after another:

- normal operation, abnormal events and prevention of accidents;
- accidents without core melt;
- accidents with core melt;

and, depending on the design:

- independence between all levels of defense-in-depth;
- radiological protection;
- waste management.

1/4
Initial lessons drawn from the events at Fukushima

As far as possible, (highly preliminary) analyses concerning the robustness of the fourth-generation concepts with respect to the events of March 2011 at the Fukushima-Daiichi nuclear power plant are given in this document. With reference to the WENRA document, cited as Reference Document [4], regarding the performance of "stress tests" on current facilities, the positioning of the fourth-generation designs involves the following hazards and deteriorated operating conditions:

- earthquakes;
- flooding;
- loss of heat sink;
- loss of electrical power;
- severe-accident management.

It aims to make an initial identification of sensitive areas and requirements for the various designs with a view to ensuring the three basic safety functions (controlling reactivity, cooling the core and containment).

It should be stressed that robustness to hazards largely relies on the reactor's specific design and installation[8] provisions, which are independent of the concepts. Nevertheless, certain concepts call for comments regarding their greater or lesser vulnerability to such hazards.

With regard to loss of electrical power and heat sinks, robustness also largely depends on aspects which, at best, are only specified at the preliminary-design stage of the reactors (such as system architecture, redundancy and diversifications).

Finally, the accident at the Fukushima-Daiichi nuclear power plant was marked by hydrogen explosions, originating in the oxidation of Zircaloy fuel cladding by water vapor. Some fourth-generation reactors may have a "hydrogen risk", which is identified in this document.

[8] Such as the characteristics of adopted contingencies and site selection.

2/
Overview of concepts

2/1
Sodium-cooled Fast Reactors (SFR)

2/1/1
Presentation of the concept

An SFR is a fast neutron reactor, *i.e.* it operates without a moderator. The core is cooled by a molten metal, sodium. Compared with thermal-spectrum neutrons, fast-spectrum neutrons more efficiently convert natural uranium (uranium-238), a fertile material, into plutonium, a fissile material, which means that an SFR could be operated in breeder mode, or conversely in burner mode for increased plutonium consumption. It can also transmute the very-long-lived actinides (americium, curium and neptunium).

In an SFR, the power density in the core can be around 300 MW/m^3 (compared with 100 MW/m^3 for the current PWRs in the French fleet). The maximum temperature of the sodium when the reactor is operating is approximately 550°C, which provides for high thermodynamic efficiencies (of the order of 40%) and a significant margin with respect to the boiling point of sodium (approximately 900°C).

It is generally planned that mixed plutonium and uranium oxide fuel (MOX – $UPuO_2$) be used, although mixed carbide, mixed nitride and even metal fuels (such as UPuZr) are also under consideration.

Existing and planned SFRs are of two types: "pool-type" reactors (such as Phénix, Superphénix, PFR and CEFR) where the primary system is totally contained in a vessel where the reactor coolant pumps and heat exchangers are immersed in sodium, or "loop-type" reactors (such as Joyo and Monju) where the primary sodium flows in loops connecting a main vessel with other vessels where the large components are located.

In the SFRs that have been built or designed to date, the primary sodium does not directly exchange its heat with the water in the power-generation system. Rather, it exchanges its heat with sodium in an intermediate system, comprising several loops (three in Phénix, four in Superphénix) fitted with sodium-sodium heat exchangers, and located in the reactor vessel for the "pool-type" design.

GIF analyses highlight the ability of the SFR to close the fuel cycle and its qualities with regard to the objectives for conserving uranium resources and reducing radioactive waste.

Furthermore, GIF analyses consider two options for the SFR: an intermediate size (150 to 500 MWe) and a large size (500 to 1,500 MWe), depending on the desired usage (*i.e.* on fuel cycle considerations).

2/1/2

Current state of SFR-concept development and outlook

The SFR design benefits from a certain amount of experience, with several power reactors having operated:

- in France, the Phénix reactor (250 MWe[9] - shut down in 2009) and the Superphénix reactor (1,240 MWe – shut down in 1997),

[9] In this document, MWe is used as an abbreviation for megawatts of electrical power and MWth for megawatts of thermal power.

Figure 1
The Phénix reactor hall —
CEA-Marcoule-France).

- in the Great-Britain, the 250 MWe Prototype Fast Reactor (PFR), which operated from 1974 to 1994,
- in former-Soviet-Union countries, the 250 MWe "BN 350" reactor, located in Kazakhstan and shut down in 1998, and the 550 MWe "BN 600" reactor in Russia, operating since 1980,
- in Japan, the Joyo experimental reactor, whose power reached 140 MWth, and the 280 MWe Monju reactor which restarted operation in May 2010 after an interruption following a sodium fire in 1995. Currently, both reactors are shut down for an indefinite period for maintenance operations on their primary systems following fuel-handling incidents,
- in India, the Fast Breeder Test Reactor (FBTR), a 13 MWe experimental SFR, has operated since 1985.

Experience feedback on SFR reactors is the subject of a very detailed IAEA report (Reference Document [5]).

Two low-power experimental SFR reactors should also be mentioned:

- The EBR-I reactor, built on the Idaho National Laboratory site, in the State of Idaho, USA. This was the first nuclear power plant to be built, and was operated from 1951 to 1962, producing an electrical power of 200 kW (for a thermal power of 1.4 MW). It was cooled by a mixture of sodium and potassium (NaK). In 1955, it suffered a partial meltdown during a coolant flow test which increased core temperature. It appears that this was caused by bending of the fuel assemblies (which were laterally clamped at both ends) under thermal expansion and resultant compaction of the fissile material.
- The Fermi 1 reactor built in the State of Michigan, USA. This 94 MWe sodium-cooled fast reactor operated from 1963 to 1972. In 1966, two of the core's 105 fuel-assemblies melted, due to blockage of a subassembly foot by a primary-system structural component that had detached.

The significant events that have affected power SFRs had various causes: inappropriate operator responses, design errors, inhibition of safety protection systems, inadequate construction requirements for some equipment items, and difficulties managing implementation due to the complexity of the industrial engineering. In addition, it should be remembered that, in 1989 and 1990, the Phénix reactor suffered reactor scrams due to sudden drops in power (*i.e.* reactivity), whose origins remain unexplained and are still the subject of investigations.

In France, in the 1980s, following commissioning of the Superphénix reactor, studies were performed on a project for a 1,500 MWe SFR reactor (called the RNR 1500). These studies were then pursued in a European context, with the Great-Britain, Germany and France partnering on a project for an SFR reactor design – the European Fast Reactor (EFR) project. Ultimately, this project was abandoned.

Given this experience, SFR design seems to be in a state of maturity which means that production of new industrial prototypes can be envisaged in the medium term (2020-2030). Several projects are underway, with varying degrees of advancement:

- the 1,500 MWe Japan Sodium-cooled Fast Reactor (JSFR) project (using a loop-type design); preliminary design is planned for 2015;
- construction of the (800 MWe) BN 800 reactor in Russia, which was suspended after the Chernobyl accident in 1986, has been restarted: initial criticality is planned for 2012. Russia is then planning a commercial 1,200 MWe version (BN 1200). Construction of a pair of BN 800 reactors in China will begin in 2013 (following agreements signed in 2009);
- completion of an Indian 500 MWe Prototype Fast Breeder Reactor (PFBR), for which criticality is planned in 2012, and which should be followed by several SFRs;
- the Chinese Demonstration Fast Reactor (CDFR) project, with unit power between 600 and 900 MWe, prior to its commercial step, the 1,000- 1,500 MWe Chinese Commercial Fast Reactor (CCFR);
- South Korea's "Korean Advanced LIquid MEtal Reactor" (KALIMER) project, with unit power of 1,200 MWe;
- the French ASTRID project, with power of 600 MWe; one of the specifications for this reactor is that, as far as possible, it should be possible to extrapolate the technical solutions used to a more powerful reactor.

Note that in 2005, the USA launched design studies on an SFR project called the Advanced Burner Reactor (ABS); these were stopped in 2008.

The year 2011 was marked by the connection of the Chinese Experimental Fast Reactor to the power grid. This

25 MWe/60 MWth SFR is the fruit of close collaboration between China and various Russian institutes.

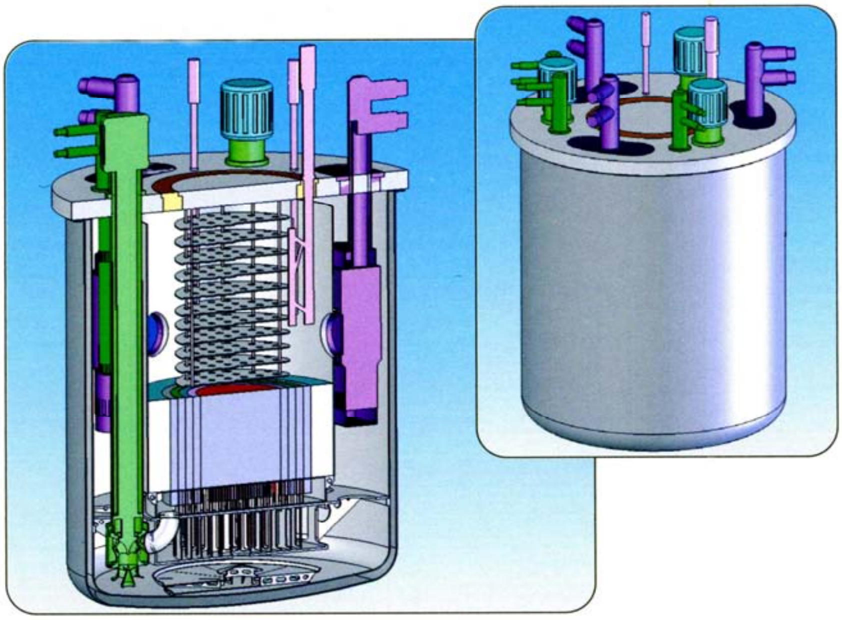

Figure 2
An "innovative" reactor block for a 1,500 MWe SFR — CEA Project — Source: SNETP-SRA 2009 (www.snetp.eu).

2/1/3
Safety aspects specific to the SFR concept

Risks specific to SFRs are mainly due to sodium's chemical reactivity with air and water.

Risks associated with the chemical reaction between sodium and air

Sodium, an alkali metal, burns in air. In the event of a leak of pulverized sodium, the consequences of its combustion are pressure increases that can be very rapid (a few seconds) and slower temperature increases (a few minutes), and the production of highly toxic sodium-oxide aerosols. These oxides react with water in the air to form sodium hydroxide, which can react with carbon dioxide in the air to form sodium carbonate.

A sodium fire can be aggravated by the heating of the concrete in the walls of the affected room, which can lead to water desorption from the concrete, and a hydrogen fire or even an explosion, in addition to concrete degradation.

It should also be noted that, given the reactivity of sodium with air, the parts of the primary and intermediate systems not filled with sodium are filled with an inert gas, generally argon (this is the case for the volume located above the sodium in the main vessel, the cover-gas space).

Risks associated with the chemical reaction between sodium and water

Water reacts with sodium instantly and exothermically, producing sodium hydroxide and hydrogen. In particular, this reaction can occur in the steam generators, which are sodium-water heat exchangers. It could lead to large dynamic loads on the affected loop of the intermediate system and possibly cause a sodium-water-air reaction in the event of failure of the steam generator's external pressure boundary, which could have extremely significant consequences (such as hydrogen explosion).

The risk of thermodynamic interactions with sodium

Besides a chemical reaction, contact between water and liquid sodium can provoke a thermodynamic interaction, leading to sudden vaporization of the water (vapor explosion), accompanied by overpressure effects. Also, and in a similar manner to what could happen on a PWR in the event of fuel melt, contact between sodium and molten fuel could also cause a thermodynamic interaction with sudden sodium vaporization.

Risk of a reaction between sodium and MOX fuel

In the event of fuel cladding failure, sodium can come into contact with the fuel (once the gaseous fission products have been released into the sodium). With MOX fuel, this creates sodium urano-plutonate compounds with a high coefficient of thermal expansion, which interact with the cladding and can split it open over a long length if the reaction between sodium and the fuel is not stopped sufficiently quickly by removing the faulty subassembly from the reactor.

Risk of coolant freezing

Sodium freezing can occur when the reactor has been shut down for a long time, when the residual heat is insufficient to compensate for heat losses. Clearly, this situation affects reactors which have been shut down indefinitely (as is currently the case for the Phénix and Superphénix reactors), but it can also occur during an SFR reactor's service life in the event of prolonged outages, such as for inspections, modifications or replacement of large components. The associated risk is the cracking of structures due to sodium contraction or expansion during a phase change (mainly during subsequent melting, given that sodium expands as it melts). For Phénix and Superphénix, two complementary provisions were adopted to prevent sodium freezing: pump operation to heat the sodium and using heat tracing cables to heat components (such as vessels, pipes and valves).

Furthermore, the viscosity of sodium may increase, or it may even solidify (in the cooler parts of the reactor), if impurity concentrations (oxides, hydrides) increase. In particular, this could lead to fuel-assembly blockage. For this reason, the Phénix and Superphénix reactors were fitted with a sodium purification system (in particular, to maintain the oxygen concentration at a few ppm), associated with detection based on measuring sodium flowrate *via* an orifice maintained at low temperature ("plugging indicator"). However, use of this detection procedure posed several difficulties during the incident where air got into the Superphénix primary system in 1990 (difficulties interpreting the measurements).

Risk of embrittlement of steels in the presence of sodium

R&D work, such as that reported in Reference Documents [25], [26] and [27], has shown that some steels can become embrittled[10] in contact with liquid sodium at typical SFR operating temperatures. While the austenitic stainless steels used in the Phénix and Superphénix reactors (type 316 and 15-15 Ti steels) do not seem to be (very) susceptible to embrittlement in the presence of sodium, the influence of the concentration of non-metallic impurities in the sodium (oxygen, hydrogen) would seem to require further study (Reference Document [26]). However, harder steels, such as T91[11], were considered for some EFR structures (steam-generator tubes, intermediate system and fuel assembly hexagonal tubes). The studies cited in [25] and [26] show that this steel is susceptible to embrittlement, even in contact with pure sodium. Therefore, for future SFRs, the risk of the embrittlement of steels in contact with sodium should be carefully studied so that, by choosing appropriate materials and metallurgy procedures, the risk of sudden failure of components under accident loads can be prevented. In any case, the risk of embrittlement of steels would appear to substantiate the usefulness of keeping the sodium in SFRs as pure as possible.

[10] The issue has general scope. It is covered again later with regard to LFRs. Also embrittlement by liquid metals was used to explain two accidents that occurred in 2004 on natural gas liquefaction plants (the Skikda accident in Algeria and the Moomba accident in Australia): these accidents were ascribed to the presence of trace amounts of mercury in the metal components.

[11] This is a martensitic stainless steel containing 9% chromium and 1% molybdenum.

2/1/4
Aspects of the safety analysis

First of all, it is important to note that the SFRs that have been built and operated clearly have a less complex design that that of pressurized water reactors in terms of their systems architecture.

Operators who have worked on both types of reactor have also stressed how much simpler SFRs are to operate. In principle, this constitutes a positive aspect for safety, with regard to the risk of human errors. However, while the systems architecture of SFRs is relatively simple, sodium-based technology is clearly less tried and tested than water-based technology, which benefits from over two centuries of experience with steam supply systems. While operations of the French Rapsodie and Phénix reactors were generally smooth, in particular thanks to the operator (the CEA) having significant support resources available (in terms of consultancy, design and R&D)[12], the move to industrial operations with Superphénix was less convincing. Incidents occurred on Superphénix, in particular air ingress into the primary system (in 1990) with resulting contamination of the sodium. This led to the French Ministers responsible for industry and the prevention of major risks and the French Directorate for Nuclear Safety (DSIN) to question "the level of technical expertise made available to the nuclear power plant" and to note "the difficulty of the diagnostics that the (operating teams) had to perform", "the excessive delay in the operating team's diagnosis due (...) to an erroneous diagnosis given by experts consulted by the operating team", and to conclude that "the difficulties associated with the sophisticated design of this reactor remain" (Reference Documents [28] and [29]).

[12] Although it was unable to explain the "reactor scram by negative reactivity" incidents that occurred on Phénix reactor in 1989 and 1990.

<u>Normal operation, abnormal events and prevention of accidents</u>

a) Anomalies that could cause accident situations

First of all, it should be noted that SFRs have characteristics that make them especially sensitive to certain events: their high core power density and neutron feedback that is less favorable than on PWRs.

In particular, with regard to neutron feedback, it should be noted that there may be a risk of a significant increase in neutron flux in the event of loss of sodium by boiling, depending on core size. This phenomenon is known as the "sodium void effect" and can also occur in the event of gas bubbles (such as argon from the cover gas space) being entrained into and through the core. This constitutes a specific characteristic in comparison with PWRs where water loss leads to a significant reduction in neutron flux.

Given these characteristics, special attention must be paid at the design stage to reducing reactivity insertion risks as far as possible, in addition to the risks of sodium boiling and gas passing through the core.

The Phénix and Superphénix reactors were fitted with "vent valve assemblies" outside the core to reduce the risk of gas passing into the fuel assemblies. With regard to the risk of sodium boiling, following the accident on the Fermi 1 reactor in the USA, specific provisions have been made in SFRs to prevent fuel-assembly blockages (and thus prevent the risk of local boiling). Thus, the circular-cross-section fuel-assembly feet have several lateral holes around the circumference to ensure that sodium flow is maintained even with migrating bodies present. Furthermore, at least for SFRs operated in France, operating provisions have been taken to ensure sodium purity (in particular, in terms of oxide concentrations, limited to a few ppm by mass), to prevent particles from penetrating fuel assemblies and blocking them. As seen above, in the context of industrial operation (of the Superphénix), compliance with these provisions was less convincing.

Managing local boiling risks in the event of fuel-assembly blockage remains a key area for design and R&D. Research is underway to reliably detect any boiling at fuel-assembly outlets, to install neutron detectors in the primary system to detect power variations and to measure the sodium flowrate in core assemblies.

The question of risks associated with sodium void effects was reassessed following "reactor scram by negative reactivity (AU/RN)" incidents on the Phénix reactor, in 1989 and 1990, which are still unexplained. However, it has been demonstrated by tests performed on the reactor itself that the reactor scram incidents were not associated with gas ingress into the core.

The RNR 1500 and EFR projects were developed with the idea of designing reactor cores that minimized positive void effects as far as possible. This objective has also been adopted by the CEA for its SFR designs and, more specifically, for the ASTRID project. Given the potential consequences, special attention must be paid to solutions used to achieve this objective.

Reactivity insertions can also be caused by inadvertent removal of absorber rods, passage of a moderator (such as oil) into the core or fuel compaction (as occurred on the EBR-I reactor in 1955). In this respect, specific provisions were made in the past on the Phénix and Superphénix reactors to reduce the risk of oil ingress (for example from reactor-coolant pump motors). Furthermore, it should be noted that operation of SFR reactors in "iso-generating" mode[13] requires little reactivity reserve, which means that, in the event of inadvertent removal of absorber rods, the amplitude of possible reactivity insertions can be significantly limited. With regard to the risk of fuel compaction, following the EBR-I accident mentioned

13 This operating mode produces as much plutonium as the reactor consumes.

above, a fuel-assembly design was adopted which has lateral clamping at only one end (at the bottom).

b) Risks associated with the reaction between sodium and MOX fuel

Major design work has been performed to minimize the risk of cladding failure. In particular, this work involved the selection of ductile austenitic stainless steels for the cladding and checking their behavior under irradiation in Rapsodie and in Phénix (in particular regarding swelling due to irradiation by fast neutrons). However, it should be noted that, for the development of SFR designs, the CEA is studying other stainless steels (such as ferritic-martensitic steels) for fuel cladding and that these steels are less ductile.

For Phénix, the results were highly satisfactory because there were only 15 cladding failures among the approximately 150,000 fuel pins that were irradiated during the 35 years of operation, bearing in mind that the reactor core was often loaded with experimental fuel assemblies using "prototype" cladding materials.

Furthermore, it should be noted that the reactors operated in France were fitted with a "cladding failure detection and localization" system linked to reactor scram, which aimed to limit the consequences of cladding failure with respect to the risk of fuel-sodium interaction. The localization system identified the failed assembly so that it could be unloaded and thus limit contamination of the primary sodium by fuel particles (the "clean reactor" operating principle). The localization system operated correctly during the cladding failures that occurred on the Phénix reactor.

c) Risks associated with the reaction between sodium and air

Experience feedback means that criteria are now available for selecting materials for equipment that contributes to sodium containment. However, as stated above, new steels under consideration for SFRs could have lower ductility and may also be susceptible to embrittlement on contact with sodium.

Leaktightness of the main vessel of reactors such as Phénix and Superphénix was permanently monitored using diversified sodium-leakage detection devices ("candle" system and sodium aerosol detection systems). It should also be stressed that significant work was performed regarding in-service inspection of the primary-system pressure boundary on the Phénix reactor, which also aimed to check that there was no degradation of core support structures. In addition to all these provisions which aimed to avoid primary sodium leakage, risk management regarding reactions between primary sodium and air is based on the presence of a safety vessel, surrounding the main vessel, and on nitrogen blanketing of the inter-vessel space.

On the Phénix and Superphénix intermediate systems, leak-detection systems were placed as close as possible to components (such as pipework and tanks). Detection led to the application of a procedure to empty the corresponding components. However, the large number of erratic detections implies that management of intermediate-system leaks must be improved.

Furthermore, sodium coming into contact with concrete leads to desorption of water from the concrete and hydrogen production by the sodium-water reaction. The main preventive measure has been to cover the concrete walls with a metal liner. Also, sodium can react with certain constituents of concrete (such as silica, SiO_2), which led to the use of special concretes for Superphénix.

d) Risks associated with the reaction between sodium and water

First of all, it should be stressed that minimizing the number of welds on the steam-generator tubes constitutes an area of significant progress identified in the context of the EFR project. Suitable selection of materials for the steam-generator tubes is also crucial for preventing the risk of a sodium-water reaction. Indeed, in addition to the usual mechanical loads to be considered, the tube materials should have good resistance to wastage effects (blowtorch effects induced on a tube by a sodium-water reaction on an adjacent tube). It should also be noted that a modular steam-generator design (as used on the Phénix reactor) can significantly minimize the consequences of a sodium-water reaction and maintain the cooling capacity of the intermediate system concerned, *via* suitable isolation of the affected module.

Furthermore, notable improvements were implemented on Phénix and Superphénix, in particular in terms of the possibility of early detection of steam-generator tube defects[14], followed by automatic isolation and dryout of the steam generator. However, experience feedback from the PFR reactor in the Great-Britain and the Phénix reactor in France shows that these detection systems are difficult to operate (the severity of the event on the PFR reactor in the UK in 1987, which led to the failure of 40 steam-generator tubes, was associated with the fact that the early detection system had been disabled following malfunctions). Finally, identification of a possible failing tube would have presented a difficulty for Superphénix, given the design of the steam generators on this reactor (tube bundle comprising superimposed helicoidal layers).

14
This involved a very sensitive procedure for measuring the hydrogen concentration in the secondary sodium, which meant that small leaks from the water tubes could be detected.

In the context of R&D performed on SFRs in France, three options have been explored with a view to preventing any sodium-water reactions:

- use of a fluid that does not react with water (lead-bismuth alloy or nitrates) in the intermediate system and with primary sodium ;
- use of a gas for the power-generation system (the steam turbine that drives the alternator would be replaced by a gas turbine);
- removal of the intermediate system and use of supercritical CO_2 for the power-generation system.

For this third option, it would clearly be necessary to study the risk of gas ingress into the primary system, an event which would cause system overpressure, could mechanically stress the fuel assemblies and possibly prevent absorber rod drop.

e) In-service inspections

In-service inspections play a major role in the prevention of incidents and accidents. In this respect, they constituted a weak point for earlier SFRs and require significant design and R&D work. At an early stage, IRSN brought this subject to the attention of industry and of the French Nuclear Safety Authority (ASN).

Given the optical opacity of sodium, in-service inspection of certain equipment items presents specific difficulties. This is the case for reactor internals involved in supporting the core. In the early 1990s, this difficulty was highlighted for the Phénix reactor in the context of a CEA request to extend operation of this reactor (following unexplained reactor scram by negative reactivity). An innovative, automated device remotely-controlled from outside the main vessel was quickly developed by the CEA and provided adequate assurances for several additional operating cycles.

Developments were also made for the Superphénix reactor: automated devices to inspect the steam-generator tubes and the main vessel from the outside ("Reactor Inspection Modules").

It is clear that, for SFRs, the inspectability of structures is one of the major subjects on which progress must be made. This applies to both pool-type reactors and loop-type reactors (according to information available regarding the Monju loop-type reactor). In France, the CEA and its partners have undertaken actions in two areas: possibilities for improving in-service inspection and repair conditions by careful design choices, and the development of appropriate systems for in-service inspections (such as non-destructive testing in a sodium environment).

It should be noted that IRSN has already drawn the attention of the CEA and the ASN to the utility of a specific inspection program during decommissioning of the Phénix reactor, for characterizing the state of major structures which were not able to benefit from direct, detailed inspections during the 35 years of reactor operation. The results of a specific inspection program such as this would confirm and possibly supplement the list of types of damage taken into account and covered during design and operation of Phénix and Superphénix, as well as the R&D work that has already been launched by the industrial partners of the ASTRID project. IRSN plans to assess the usefulness of its involvement in R&D work in the light of the results of this inspection program.

f) Taking hazards into account

With regard to taking hazards into account, earthquakes have been subject to special attention. In particular, these involve risks due to leaks from the sodium systems and reactivity insertions *via* movement of fuel assemblies. Two points deserve comment:

- Given the low pressure in the systems, the sodium vessels and pipework are thinner walled compared with PWRs, so this equipment is more sensitive to seismic loads. Furthermore, the large quantities of sodium may produce fluid-structure interaction phenomena *via* their significant mass and inertia, and these should be taken into account. Design and dimensioning of SFRs for earthquake conditions is therefore complex. The RNR 1500 project adopted an aseismic isolation system using anti-seismic pads below the reactor.
- To overcome reactivity insertion *via* movement of fuel assemblies, systems for early reactor scram on detection of an earthquake were installed on French SFRs. In particular, these systems used "articulated" absorber rods to shut down the reactor even in the event of movement or distortion of fuel-assemblies in the core.

The emergency sodium-air heat exchangers located on the upper outer part of the Superphénix reactor building could constitute a sensitive point for certain external hazards (such as external explosion or aircraft crash).

<u>Accidents without core melt</u>

Prevention of core melt involves the protection and safeguard functions provided by reactor scram and the residual heat removal systems. A level-1 probabilistic safety analysis (PSA1), aiming to assess the probability of core melt, was undertaken by EDF and AREVA for Superphénix, but was not fully completed. It was limited to internal initiator events and at-power reactor states, and did not take into account possibilities for recovery of failed equipment. The main results were that the risk of core melt was low (approximately 3.10^{-6} per year) and that the situation that contributed the most to the probability of core melt was delayed failure of the main vessel (by creep) in the event of prolonged unavailability of the residual heat removal systems, bearing in mind that in such a situation the releases could be large and unfiltered.

a) Controlling reactivity and reactor shutdown

Absorber rod drop is the only procedure for stopping the chain reaction in an SFR, given that neutron feedback is weaker than in PWRs and that this feedback can become positive in the event of sodium void or fuel compaction.

Absorber rod drop (control rods and scram rods) is initiated by measurements mainly taken in the reactor itself (such as power, flowrate through the core or sodium temperature), when the measured values deviate excessively from the nominal values. In this respect, as the power density and neutron feedback of SFRs are somewhat unfavorable with regard to core-melt risks, they require very rapid reactor scram when certain significant parameters become abnormal. This requires that changes in these parameters be detected, even when they are local, and therefore requires particularly extensive and effective instrumentation. Furthermore, control rod drop must be very reliable, which explains the requirement for a scram system that uses redundant and diversified means, as was already the case for Phénix and Superphénix (in particular *via* the supplementary shutdown system, which on Phénix used an articulated rod that could be inserted into the core even in the event of fuel-assembly distortion; Superphénix had several such rods). Three different reactor scram systems were planned for the EFR, one of which was passive. Passive systems are planned by the designers of new fourth-generation SFRs.

b) Residual heat removal

Firstly, some design provisions should be noted which aim to maintain the sodium inventory. The main vessel of the primary system is surrounded by a safety vessel. In pool-type reactors such as Phénix and Superphénix, the space between the two large vessels[15] was designed such that a leak from the main vessel would not lead to uncovery of the intermediate heat exchangers, so that

[15] The diameter of Superphénix's main vessel was 21 m.

the residual heat could still be removed *via* the intermediate loops[16].

Furthermore, SFRs have significant thermal inertia, due to the large quantities of sodium in the primary and intermediate systems (for example, the Superphénix had 3,300 metric tons of sodium in its primary system and 1,550 metric tons of sodium in all intermediate-system loops combined). Typically, in the event of loss of the normal residual heat removal systems[17], the sodium heats up slowly and temperatures reach their maximum after 2 to 3 days. This gives operators significant grace periods, during which they could restore the cooling systems if necessary.

However, prolonged total loss of the residual heat removal systems would inevitably lead to vessel failure by creep and collapse of the reactor pit, with extremely significant consequences (such emptying of the reactor, severe core damage and a large sodium fire in the air charged with fission products and fissile materials).

Also, a design objective should be to "practically eliminate" prolonged total loss of the residual heat removal systems, or to limit, *via* design provisions, the consequences of such an event on primary sodium containment. On this last point, it should be noted that on the RNR 1500 it was planned that the safety vessel be "anchored" in concrete. However, this type of solution could pose problems for safety-vessel inspectability.

Concerning the objective of "practically eliminating" prolonged total loss of residual heat removal, it should be remembered that redundant and diversified systems were used on the Phénix and Superphénix reactors and for the RNR 1500 and EFR projects, but that achieving this objective would require better assurances on the abilities of such systems to operate passively using natural convection. For Phénix and Superphénix, the possibility of natural convection was demonstrated for certain systems on a standalone basis, but was never checked with overall certainty for all required systems working together. In many cases, this possibility was only assessed by calculations. Additionally, implementation of natural convection requires several local interventions, such as gradual opening of the sodium-air heat exchanger slide valves, which is a complex operation. Opening them too quickly could lead to sodium freezing and blockage of the natural convection, while opening that is too slow or too late could lead to excessive temperatures in the reactor. Therefore, improvements in SFR design remain paramount in this area.

16
For Superphénix, the reactor pit was designed such that the fuel pins would not be uncovered in the event of leakage from both vessels.

17
In such a situation, residual heat removal is provided by emergency systems, in particular systems outside the vessel in the reactor pit.

Finally, on the Phénix and Superphénix reactors, in addition to implementing natural convection, passive residual heat removal involved radiative heat transfer from the vessels to water systems outside the vessels. However, the emissivity of metal structures can change significantly over time. Therefore, the maintenance over time of the possibility of radiative heat transfer should be checked throughout the service life of an SFR.

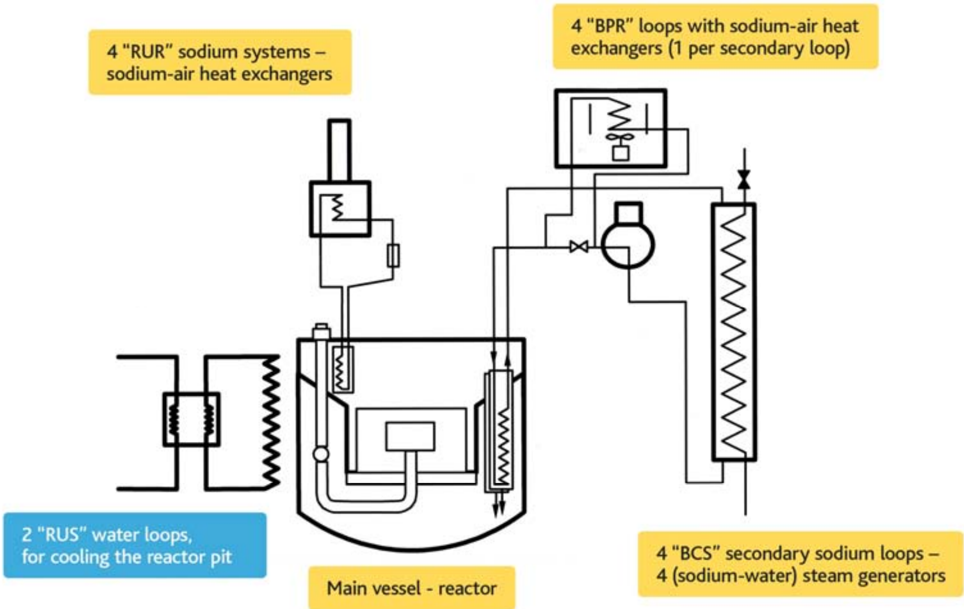

Figure 3
The various Superphénix systems that contribute to the "residual heat removal" function.

c) The radiological consequences of accidents without core melt

For accidents that do not lead to core melt, the radiological consequences are relatively minor because, due also to adoption of the "clean reactor" operating principle, it is mainly fission noble gases that are released into the systems, given that the ductility of the austenitic stainless steel used for the fuel pellet cladding prevents significant cladding cracking or failure. This may need to be reconsidered if the fuel cladding on new SFRs were to use new steels, which are currently the subject of R&D (ferritic-martensitic steels that are less ductile that austenitic steels). Furthermore, referring to the design of the Phénix and Superphénix reactors, the presence of buffer volumes in the argon cover gas circuit, gives time for the radioactive decay of fission products, as does a large holding chamber located upstream of the stack. This chamber is formed of chicanes to give time for additional radioactive decay of fission products by acting as a delay line, deferring releases and providing an opportunity to trap any volatile fission products.

d) Sodium fires

The risk of releases of toxic chemical aerosols produced in the event of sodium fires must be taken into account in SFR design. Also, as fire aggravates the dispersion of radionuclides, the containment must be designed to minimize the chemical and radiological releases which could result from a primary-sodium fire, as the primary sodium contains sodium isotopes, tritium and, possibly, fission products. The dome fitted to the Superphénix was designed to "contain" a fire involving a metric ton of primary sodium in pulverized form in the event of a severe accident. For new SFR designs, special attention should be paid to defense-in-depth with regard to primary-sodium fires, especially under accident conditions.

Buildings and rooms that house intermediate-system loops must also be designed to reduce the duration of a sodium fire as much as possible by limiting the air supply. Outlets were retrospectively installed into the wall of the Superphénix containment building, at the level of the "secondary-system galleries" (rooms containing secondary sodium loops), to reduce the pressure peak resulting from a pulverized-sodium fire. However, this solution is not totally satisfactory from a safety perspective as it creates containment bypass: while the sodium in the intermediate-system loops is only slightly radioactive, it does contain a small quantity of tritium from the reactor that has diffused through the tubes of the intermediate heat exchangers.

IRSN has activity contributed to increasing knowledge regarding sodium fires and developing related computer-modeling tools. The experimental area investigated was sufficiently wide to provide deep understanding of the combustion phenomena and qualification for the models. In collaboration with the CEA, IRSN produced a summary report on this subject, which also includes risks associated with the sodium-water interaction.

It should be noted that the current chemical toxicity thresholds for sodium aerosols are much stricter than those considered during the safety analyses for the Superphénix reactor. As an indication, the IDLH[18] value for sodium has been reduced from 250 mg/m^3 to 10 mg/m^3.

e) Sodium-air-water reactions

Although the steam-generator pressure boundary is designed be resistant to sodium-water reactions, the consequences of its failure must be examined as part of defense-in-depth with regard to the

[18] Immediately Dangerous to Life and Health: with this concentration in the atmosphere, a person can remain for 30 minutes without a gas mask, a duration considered sufficient to flee the polluted area. There are no statutory values in France.

risk of a sodium-water-air reaction. Such an event could have extremely significant consequences (for example, a hydrogen explosion which could endanger the reactor building) and should therefore be "practically eliminated".

Accidents with core melt

Accidents with core melt were taken into account[19] in the design of the Phénix and Superphénix reactors. In both cases, this concerned scenarios involving reactivity insertion into the core. In particular, taking core melt into account led to certain strength requirements for the primary system (the main reactor vessel and its upper cover slab). For this reason, the primary system was designed to resist a release of mechanical energy (for example, 800 MJ for Superphénix). Two phenomena could lead to release of mechanical energy:

- a thermodynamic interaction between molten fuel and primary sodium, a phenomenon which would lead to the formation of a bubble of sodium vapor which could distort structures and cause sodium movements during its expansion in the reactor;
- fuel vaporization, also leading to a bubble expanding in the reactor.

Furthermore, with the aim of ensuring the containment function of the Superphénix reactor in the event of core melt, in addition to the safety vessel installed around the main vessel and a metal dome above the slab (as slab leaktightness could not be ensured during the design stage), the reactor was fitted with a "core catcher" inside the main vessel, an option that was adopted on the RNR 1500 and EFR projects[20]. Use of an internal "core catcher" aims to overcome the difficulty that would be caused by a large sodium fire in the reactor pit following piercing of the vessel by corium. The Chinese CEFR reactor is also fitted with such a device. The BN 600 and Monju reactors do not have a core catcher. In its current design, the JSFR project does not seem to have one.

For Superphénix, the risks of core melt were the subject of many analyses and experimental programs, mainly at the Cabri and Scarabée facilities on the Cadarache site: this involved analyses and tests regarding the various phenomena from partial melt in a fuel pin to the consequences of fuel-assembly melt and full core meltdown, focusing particularly on recriticality risks for molten materials containing fuel.

19 This choice is partly based on experience feedback from the EBR-I and Fermi 1 reactors in the USA, mentioned above.

20 Note that no dome was planned on the RNR 1500 and the EFR projects.

Several more tests were performed on the Cabri reactor and the Silène facility following the decision taken in 1997 to definitively shut down Superphénix. These provided significant lessons, and IRSN and the CEA produced a joint summary report for all these tests.

However, a certain number of complex subjects still require deeper study with the support of appropriate analyses and experimental programs. In particular, the following possibilities should be explored:

- sufficiently early detection of local melt in the core, to prevent a partial melt becoming a general meltdown;
- prevention of the risk of recriticality by corium relocation (as this criticality would lead to a "secondary" power excursion, to be distinguished from the power excursion that initiated core melt). For this reason, the fuel assemblies on the Japanese JSFR project may be fitted with a sodium channel, which the designer says would eject the corium towards the top of the reactor in the event of a core melt accident, reducing the power and preventing the formation of a corium pool at the bottom of the reactor (FAIDUS concept – see Reference Document [6]). The Japanese have studied this concept in the EAGLE experiment program and would have obtained encouraging results;
- maintaining and cooling the corium in the main vessel, with or without an internal "core catcher". Further demonstration of the effectiveness of such a device is required;
- collecting and cooling the corium below the reactor vessel in an external "core catcher" (or corium spreading area). This assumes that the risks associated with sodium pouring into the reactor pit have been covered.

Furthermore, the designers may propose less conservative assumptions regarding "vapor explosion" phenomena and the associated energy, in particular seeking to "practically eliminate" the possibility of sudden contact between large quantities of sodium and fuel. The approach and elements used for this subject will must be carefully examined (experimental support, such as the use of powerful simulation tools or additional design provisions).

With regard to radioactive releases, it should be stressed that sodium is able to trap a significant fraction of the fission products released in the event of fuel melt, such as iodine, barium and tellurium isotopes (but not cesium isotopes or fission gases).

However, analyses of possible releases have been less complete for SFRs than for PWRs. A group of international experts working for the DOE, to which IRSN has contributed, identified several points requiring additional work:

- release of radionuclides from fuel due to a sudden reactivity insertion leading to high temperatures;
- for metal fuels (not planned on ASTRID), carrying along of fuel and the "sodium bond[21]" loaded with fission products during depressurization of a fuel pin with cladding failure;
- the rates of radionuclide transfer into the sodium *via* fuel leaching;
- high-energy interactions between sodium and molten fuel and associated radionuclide transfers into the sodium;
- enrichment of the surface of the sodium contained in the main vessel by dissolved or suspended radionuclides;
- thermal decomposition of sodium iodide in the containment building;
- reactions of iodine compounds in the containment building to form volatile organic iodides.

These points are detailed in Reference Document [7].

Furthermore, questions pertaining to the possibilities and consequences of leakages of primary sodium loaded with radioactive materials *via* the upper cover and into the containment building during an accident with core melt, particularly in the event of "vapor explosion" in the core, shall be examined more deeply for future SFR projects than they were for the Phénix and Superphénix reactors. Examination of these questions would need to take into account issues such as the design of the upper cover, whether or not it has a dome, and the design of any such dome. Among these questions, determination of the type and quantities of radioactive products from core melt that could enter the primary sodium and the cover gas (generally argon, which fills the cover-gas space above the primary sodium), and then be ejected *via* the upper cover, constitutes a subject that merits deeper consideration, as the assessments made in the past for the Superphénix reactor were essentially theoretical.

21
A sodium-filled space between fuel and cladding.

2/1/5

Assessment of the SFR concept with respect to the Fukushima accident

Preliminary comments

The accident that occurred on the Fukushima-Daiichi nuclear power plant caused significant damage not only to the reactors but also to the spent-fuel pools. In Phénix and Superphénix, spent fuel was stored in sodium in a component called the "spent-fuel storage barrel[22]", with a design similar to that of the main vessel of these reactors. The following comments could also be applied to this type of component.

The spent-fuel storage barrel solution was not adopted for the EFR and IRSN has no information on this issue for ASTRID.

Earthquake

Along with loss of electrical power and the heat sink, whose consequences are assessed below, a powerful earthquake (significantly stronger than the design-basis earthquake) could have very significant consequences for the safety of SFRs, as it could lead to core distortion that may impair absorber bar drop, an increase in core power by reactivity insertion (due to fuel compaction) or loss of the second containment barrier (the main vessel and its support structures).

The risk of core distortion should lead to designers examining the possibility of adopting design provisions to improve core rigidity, bearing in mind that this can cause problems for fuel-assembly handling, and of developing effective means to stop the chain reaction even in the event of significant core distortion (for example, by using articulated absorber rods as on the Phénix and Superphénix reactors). Finally, it should be noted that a "seismic" reactor scram system and diversified means of stopping the chain reaction (articulated rods or other designs) should have "hardened safety core" robustness, as per IRSN terminology for complementary safety assessments.

With regard to the risk of loss of primary sodium containment, it should be remembered that in reactors such as Phénix and Superphénix the presence of a second vessel (safety vessel), or even a third (on Phénix), can reduce the risk of significant losses of primary sodium and fuel-assembly uncovery if the spaces between the vessels are suitably designed. The free space in the reactor pit could also be reduced to a minimum, given that a "solid" reactor-pit

[22] The Superphénix barrel was abandoned after a leak from the barrel's corresponding vessel in 1987, due to a crack in the 15D3 steel. It was replaced by a spent-fuel transfer drum with argon (fuel transfer station) that was not used for storage.

design should provide stability in the event of severe earthquakes. Limitations may arise however, given the need for in-service inspection of these structures. It should also be noted that the aforementioned equipment items (vessels and reactor pit) require that cooling be ensured to maintain their integrity. This function is provided by cooling-water systems and air-water heat exchangers: prolonged absence of water in these systems could lead to vessel or slab failure and to a loss of mechanical strength for the reactor pit concrete.

Given these issues, at least one of the vessels (main or safety), the reactor pit, one or more cooling-water systems for these components and the inert-gas systems (to prevent sodium fires) must have "hardened safety-core" robustness.

Furthermore, a severe earthquake could cause multiple sodium leaks in the intermediate system (where this exists) and sodium fires that could jeopardize important equipment and rooms. The strategy with regard to such a situation could be to rely on draining the damaged systems, providing that robust sodium storage tanks are available, and that there is at least one robust means of residual heat removal (for example, the "RUR" emergency cooldown loops connected to the primary system in the Superphénix design). Thus, the sodium storage tanks and the emergency residual heat removal system(s) would have to be designed with "hardened safety core" robustness.

Flooding

Given the reactivity between sodium and water, which leads to a risk of hydrogen explosion, flooding of the rooms containing sodium equipment would create a serious risk, given the sodium leaks could occur in the event of a severe earthquake. Under these conditions, other than drastic choices in terms of site and "elevation" of the platform, implementation of robust "volumetric protection[23]" would be necessary to "practically eliminate" such a situation. Some components are particularly sensitive, such as the secondary sodium storage tanks, which are located in the lower part of the facility and likely to be filled by sodium during reactor outages or in the event of sodium leaks.

Total loss of electrical power or the heat sink

For reactors such as Phénix, Superphénix or the RNR 1500 and EFR projects, loss of electrical power or the heat sink would have affected residual heat removal as follows.

23 This involves provisions to ensure that rooms that house safety-related equipment are isolated from external flooding risks. This concept was implemented in the context of experience feedback from flooding on the Blayais site in December 1999.

Total loss of electrical power causes the sodium-circulating pumps of the primary and intermediate systems to stop. In principle, adequate cooling could be provided by natural convection, both in the primary system and the cooling loops such as those of the "RUR" emergency cooldown system on the Superphénix, which just uses atmospheric air as a heat sink (sodium-air heat exchangers). However, as stated in Section 2/1/4, experimental demonstration of this has never been performed and accident operation would not be straightforward, especially if it had to be deployed in an accident context like that at Fukushima. Also, the possibility of natural convection necessarily involves "deployment" of residual heat removal systems at high elevations, which could make equipment such as the sodium-air heat exchangers vulnerable to seismic loads. The leaktightness and functionality of such equipment in the event of a severe earthquake would constitute design requirements.

Also, loss of the heat sink may jeopardize the ability of dedicated systems to maintain a sufficiently low temperature for the main vessel, safety vessel, reactor slab and reactor-pit concrete. Robust provisions are conceivable for emergency systems, such as the installation of water-air heat exchangers on the systems listed above as on Phénix or the possibility of injecting water directly into these systems from fire-service type connections.

Severe-accident management

Given the events that occurred at the Fukushima nuclear power plant, the risk of losing the primary sodium inventory should also be considered. In this respect, while it is inconceivable that sodium could be added to the primary system by equipment connected in an emergency manner, it seems possible to foresee at the design stage devices, as in the case of the Superphénix, to reinject any sodium that may have leaked into the safety vessel(s) back into the primary system.

Furthermore, the possibility of vessel piercing by corium (including in the event of failure of an internal "core catcher") would seem to impose a design-stage requirement for a suitable inert-gas system (of hardened safety-core robustness) for the reactor pit.

In any case, the possibility of rapid core unloading, in particular in the event of leakage from the main vessel and safety vessel, should be studied for SFRs, this subject having already been identified during the analysis of some safety aspects for Phénix and Superphénix.

2/1/6
Conclusion

In summary, the following points should be noted for SFRs:

- Sodium reacts violently both with air, which can cause large overpressures in rooms and systems, and with water, which leads to a risk of hydrogen explosion.
- Overpressures can also result from contact between liquid sodium and molten materials (such as fuel or steel cladding) *via* thermodynamic interaction (sodium "vapor explosion"); this could be the case in the event of fuel melt in the core.
- The risk of embrittlement of steels in contact with liquid sodium, and the possible influence that impurities in the sodium may have on this risk, would seem to require further research, especially if use of steels other than austenitic stainless steels were considered.
- SFRs have risks of power increases by reactivity insertion, due to the possibility of positive neutron feedback in the event of boiling or gas ingress into the core (leading to a void effect), by fuel compaction in the core (for example, due to assembly distortion) or by relocation of molten materials (such as steel cladding or fuel). These risks depend on the core design and, in particular its size (the positive void effect is greater when the core is larger and when the fuel is loaded with minor actinides). Possibilities for reducing the positive void effect are the subject of research by designers.
- The properties of liquid sodium mean that residual heat removal by natural convection can be considered for the sodium systems, but this remains to be properly demonstrated, in particular for the possibility of "generalized" natural convection in all systems at once. It should be validated experimentally on possible future SFRs (for example during start-up tests).
- The large mass of sodium in an SFR gives it significant thermal inertia, which provides grace periods for operators under certain conditions, in particular in the event of failure of the residual heat removal systems once the chain reaction has stopped.
- Given the high power density in the core of an SFR (three times higher than for PWRs), inadvertent increases in power and fuel-assembly cooling faults, such as fuel-assembly

blockage with reactor at power, deserve special attention as they constitute possible initiator events for fuel melt, or even core meltdown. These initiator events have been widely studied for various reactors and reactor projects (in France: the Phénix and Superphénix reactors, and the RNR 1500 and EFR projects). However, reduction of the associated risks requires progress in several areas, such as design of the core and absorber rods, design and performance of monitoring devices (such as neutron detectors and power sensors) and the reactor protection system.

- Prolonged total loss of residual heat removal systems would have severe consequences. It could lead to delayed failure of the reactor vessel by creep, which could lead to severe core damage, with large unfiltered releases into the environment. Design of future SFRs should aim to "practically eliminate" such a situation.
- With regard to fuel melt, or even reactor core meltdown, which was taken into account in the design of the reactors and projects mentioned above, some aspects and phenomena require further research: molten materials moving and the possibilities of a recriticality accident, conditions leading to a high-energy "vapor explosion", the design and effectiveness of "core catchers", transfers of radionuclides from the degraded core into the environment (in particular the ability of sodium to trap radionuclides).
- The inspectability of structures in a sodium environment remains a key area for progress in safety for future SFRs and involves the accessibility of these structures, inspection procedures and associated devices.
- With regard to sodium-water reactivity, the design of future SFRs must aim to "practically eliminate" a generalized sodium-water-air reaction, as such an event could have very severe consequences (in particular due to possible hydrogen explosions).
- Given the toxicity of the aerosols produced in the event of a sodium fire, SFR designs still need to be improved to further limit releases into the environment. For example, this involves the choice of where to install sodium systems, and particularly steam generators, in the various buildings and rooms, as well as the resistance-to-sodium-fires requirements for the rooms.

- Given the above, achievement of safety at least as good as that of third-generation PWRs would apparently require specific demonstrations and significant advances in certain areas (such as core physics and instrumentation, the possibility of natural convection in the sodium systems, and in-service inspections).
- However, nothing has so far been identified that would makes this impossible.
- With regard to the events that occurred on the Fukushima-Daiichi nuclear power plant, the main point to note is that robust provisions would be required to "practically eliminate" flooding of rooms containing sodium systems, which could include drastic site-selection criteria for SFRs.
- Several SFRs of significant power have been operated (in the USA, France, the Great-Britain, former-Soviet-Union countries and Japan) and another industrial-scale reactor (the BN 600) is currently operating in Russia. Considerable experience feedback is available on SFRs.

2/2

High / Very-High Temperature helium-cooled Reactors (V/HTR)

2/2/1

Presentation of the concept

High Temperature Reactors (HTRs) and Very High Temperature Reactors (VHTRs), grouped as V/HTRs, are thermal-spectrum reactors, cooled by circulating helium under pressure (50 to 90 bars). Graphite is used both as a moderator and a neutron reflector. V/HTRs are characterized by significant heating of helium in the core (approximately 500°C) and a planned mean core-outlet temperature of 750°C to 850°C for HTRs and over 900°C in the future for VHTRs. These characteristics mean that a thermodynamic efficiency of at least 50% can be envisaged, compared with 30% to 35% for current PWRs and 40% for SFRs.

The power density in a V/HTR is around 4 to 10 MW/m^3, compared with 100 MW/m^3 for PWRs and 300 MW/m^3 for SFRs.

The most advanced fuel developed for these reactors is called TRISO (TRistructural ISOtropic). It comes in the form of spherical particles about a millimeter in diameter, with a kernel of fissile material (uranium-, plutonium- or thorium-based carbides, mixtures of oxides and carbides, or oxides) covered by several coatings (Figure 4) which give the TRISO particle good sealing and mechanical strength up to a temperature of at least 1,600°C.

Developments have been made in the manufacture of this type of particle such that low particle failure rates (approximately 10-5/particle) have been obtained, although it should be borne in mind that a reactor can contain 109 particles. For this reason, the helium is purified during reactor operation.

For new fuel, work collated in the IAEA report in Reference Document [8] gives a value of 2,000°C as the threshold above which significant breakdown of the particle coatings was observed, leading to large releases of fission products. However, other degradation mechanisms come into play above 1,700°C. For this reason, the value of 1,600°C has been adopted by V/HTR designers as the criterion to be met, including under accident conditions. The melting point of the fuel kernel itself is approximately 2,700°C.

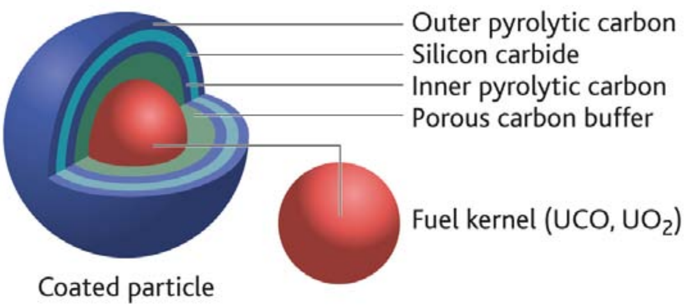

Figure 4
TRISO fuel particle;

The TRISO particles are distributed in a graphite matrix to form the fuel elements. These are in the form of pebbles or compacts (Figure 5). The former are used in pebble-bed reactors.

Figure 5
Pebble (left), compact forming a prismatic block (right).

The density of TRISO particles in a fuel element can be adjusted over a wide range, depending on the type of fuel and the power density sought. This is a key point in understanding the flexibility of this design and its robustness with regard to loss of coolant accidents (LOCAs), as explained in Section 2/2/4 below.

Finally, it should be noted that the V/HTRs currently under consideration use enriched uranium fuel, without later reprocessing of irradiated particles. However, several types of fuel kernel have been studied and irradiated (plutonium and thorium) and inert matrices (such as magnesium oxide) could also be used in the place of uranium-238 to form transmutation targets. Techniques for reprocessing the particles and graphite have also been explored at the laboratory scale. The V/HTR concept is therefore not necessarily only associated with a once-through fuel cycle.

2/2/2

Current state of V/HTR-concept development and outlook

The first HTRs were developed in the 1960s to 1980s, particularly in the USA, the Great-Britain and Germany.

Two HTRs have been operated in the USA, the first (a 200 MWth reactor) on the Peach Bottom site from 1966 to 1974 and the second (a 330 MWe, 842 MWth reactor) at Fort St Vrain from 1977 to 1992.

In Germany, a (15 MWe, 40 MWth) experimental pebble-bed reactor (the AVR reactor) operated from 1966 to 1988. A 300 MWe prototype power reactor (the Thorium High-Temperature Reactor, THTR 300) operated from 1985 to 1988. It should be noted that the AVR operated using two types of fuel, one made of slightly-enriched uranium and another made of a uranium-thorium[24] mixture. The THTR used thorium fuel.

Until 1988, Germany was involved in industrial projects: the HTR-Modul, PNP-200 and PNP-500 projects.

In Great-Britain, a 20 MWth experimental reactor (the DRAGON reactor) operated from 1965 to 1976.

Currently, two experimental reactors are operating:

- in China, the 10 MWth HTR-10 reactor;
- in Japan, the 30 MWth High Temperature Test Reactor (HTTR), which has achieved a reactor-outlet coolant temperature of 950°C. Safety tests have been performed on

24 Thorium-232 is a fertile isotope, which produces the fissile isotope uranium-233 in the reactor

this reactor (the cooling was stopped), and the results of these tests are being analyzed, by IRSN among others.

Significant events that affected the HTRs which have operated, and that have been reported in the literature, are as follows:

- On the Fort Saint Vrain reactor, neutron-power instability problems were caused by graphite-block movements in the core and water ingress into the primary system at the motor-driven blowers; the NRC report in Reference Document [32] provides a comprehensive summary of the 279 events catalogued on this reactor.
- On the THTR, there were failures of insulation-component fastenings on core outlet pipework.

In terms of radiological protection, the available information (in particular Reference Documents [33], [34] and [35]) generally reports low collective doses:

- For the AVR, except at the beginning of operations, which was marked by collective doses between 1 and 1.25 person-Sv/yr due to the need for intensive maintenance, the collective dose gradually declined to approximately 0.2 person-Sv/yr by the end of operations.
- For the THTR, the collective dose was approximately 0.1 person- Sv/yr at the end of operations.
- For the Fort Saint Vrain reactor, Reference Document [35] reports collective doses not exceeding 0.03 person-Sv/yr over the period 1974 to 1978.

These values can be compared with 0.7 person-Sv/yr for operating the 900 MWe PWRs in the current French fleet. However, they must be used with caution, as the volume of maintenance (including in-service inspections) used on these high-temperature reactors is unknown, and it seems difficult to extrapolate the values to future V/HTR power reactors.

There have been few industrial projects. Nevertheless, in terms of R&D, the EU has backed several projects and the IAEA various Coordinated Research Projects (CRPs).

The withdrawal of Germany has been a significant brake on development of the concept in Europe, despite the fact that Poland is examining HTRs with interest (Polish universities and industry participated in the EU's EUROPAIRS project), with a view to launching a nuclear power industry to replace its coal-fired power stations.

In the USA, the Next Generation Nuclear Plant (NGNP[25]) project for a combined-heat-and-power reactor paired with an industrial facility, led by the DOE and currently at the beginning of the pre-conceptual design phase, is blocked in negotiations with the Federal Government on the question of public/private funding issues, to the point that the NRC has decided to halt certification analyses. Nevertheless, R&D activities commissioned by the DOE do have funding for 2012. It should be noted that AREVA developed an industrial VHTR project, with a thermal power of approximately 600 MW, to be able to respond to a possible DOE invitation to tender (the ANTARES project, which was the subject of some technical correspondence with IRSN).

In South Africa, construction of the Pebble Bed Modular Reactor (PBMR) planned for Koeberg was finally abandoned in the 2010, due to a lack of clients and investors.

Finally, China is the only country to have developed a prototype industrial reactor, the High-Temperature-Reactor Pebble-bed Modules (HTR-PM) project, comprising two modules each of approximately 250 MW thermal power, which together can produce 210 MW of electrical power (see Figure 6 and table below). The option retained is very similar to the most recent German projects and is based on experience acquired with the HTR-10 reactor over more than 10 years. According to Reference Document [9], the reactor should be commissioned in 2013, but the main components (reactor vessel and steam generators) are currently still under manufacture.

With regard to site-specific aspects, it should be noted that, given the high temperatures aimed for, the V/HTR concept is highly suitable for combined heat and power generation paired with industrial facilities. Various safety assessments have been performed regarding the pairing of a V/HTR with an industrial facility: for example, the NRC-organized assessment for the NGNP project (Reference Document [30]) and work performed in the context of the EUROPAIRS project (Reference Document [10]). The NRC-organized assessment highlighted specific risks such as, in the event of a leak from the industrial facility, the risk of ground-level dispersion of a layer of dense (cold) gases which could explode. Work performed in the context of the EUROPAIRS project, led by IRSN, has shown the need for consistency in combining the safety approaches adopted for the reactors with those adopted for the industrial facilities, but did not identify any incompatibilities. It would seem possible to pair a V/HTR with an industrial facility while complying with acceptable distancing conditions.

[25] http://www.nextgenerationnuclearplant.com/

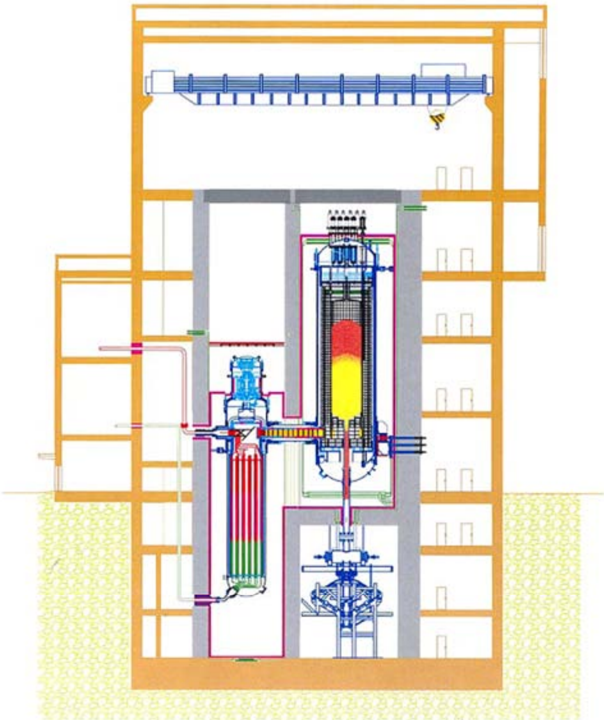

Figure 6
HTR-PM project – an Institute of Nuclear and New Energy Technology (INET) document.

HTR-PM Designs Parameters

Plant electrical power, MWe	211
Core thermal power, MW	250
Number of NSSS Modules	2
Core diameter, m	3
Core height, m	11
Primary helium pressure, MPa	7
Core outlet temperature, °C	750
Core inlet temperature, °C	250
Fuel enrichment, %	8.9
Steam pressure, MPa	13.25
Steam temperature, °C	567

2/2/3

Safety aspects specific to the V/HTR concept

The risks specific to a V/HTR are mainly associated with the presence of graphite in the reactor.

Air ingress into the reactor could therefore lead to oxidation of graphite, or even a graphite fire[26]. Water ingress could lead to corrosion of graphite and the production of flammable gases, together with increased core reactivity.

26
A graphite fire is self-sustaining combustion, *i.e.* rapid oxidation at high temperature which maintains itself without needing external heat.

The risk of a graphite fire due to oxidation following air ingress has been studied in detail, in particular following the accidents at Windscale and Chernobyl. It should be noted that in both cases an additional source of heat (other than oxidation) initiated and maintained the graphite fire: the Wigner[27] effect at Windscale and the fuel's residual heat at Chernobyl which was combined with signification break-up of the graphite following the reactor's explosion. It would seem difficult for these conditions to combine in a V/HTR. Indeed, the energy stored by the Wigner effect is negligible at the temperatures reached by the irradiated graphite in a V/HTR (> 350°C). Furthermore, in a V/HTR, to maintain graphite oxidation, there would need to be air circulation strong enough to bring sufficient oxygen but not so strong as to significantly cool the core. Finally, the oxidation of graphite is strongly influenced by the presence of impurities, which favor early oxidation. Oxidation of graphites like those involved in the Windscale and Chernobyl accidents becomes significant at considerably lower temperatures than for the high-purity graphites envisaged for future V/HTRs.

In summary, it may be possible to exclude graphite fires for V/HTRs, although further research is required. It should be noted that, for the PNP reactor projects, German experts considered that the risk of graphite fires could be excluded in the event of a LOCA, even if the blowers continued to be operated (Reference Document [11]). Furthermore, analyses show that, after several hours, a single-ended pipe break would lead to a very thin layer of oxidized graphite (various NACOK tests performed by the Jülich research center).

The risk of corrosion of graphite is smaller in the event of water ingress than in the event of air ingress, in particular because the reaction with water is endothermic.

With regard to the risk associated with gases produced in the event of water or air ingress, it should be mentioned that, in the context of the EU's "Advanced high-temperature Reactors for Cogeneration of Heat and Electricity R&D" (ARCHER) project, IRSN is contributing to research into fire and explosion risks. According to German designers, this risk can be excluded if the power of the reactor is limited to 200-250 MWth (which is relevant for pebble-bed core reactors).

Air and water ingress transients have been studied in detail for the reactors built in Germany, Japan and China. IAEA published several summary documents on these subjects in the 1980s and 1990s (see the 1993 Report in Reference Document [11]).

[27] Graphite stores energy during irradiation at low temperature (below 115°C). If, after cooling, the graphite is reheated (above 80°C), it suddenly releases the stored energy (Wigner effect), which can lead to a graphite fire, as occurred during the Windscale accident.

2/2/4
Aspects of the safety analysis

Normal operation, abnormal events and prevention of accidents

a) Prevention of significant core damage

Clearly, detailed experience feedback would need to be gathered for consideration of new V/HTRs. IRSN has no specific information on the solutions envisaged with regard to certain events, in particular the graphite-block movements in the core that affected the Fort Saint Vrain reactor.

It should be noted that, in the case of V/HTRs, the favorable physical characteristics of the fuel and the reactor in general play an essential role in preventing aggravation of incidents or accidents (characteristics such as feedback due to the Doppler Effect and feedback due to the graphite moderator). Thus, in the event of loss of the cooling systems, the strong thermal feedback due to graphite virtually stops the nuclear reaction, even without the intervention of a scram system.

Furthermore, adjustment of the moderating ratio (by altering the density of fissile matter) can be used to optimize reactor behavior in the event of water ingress, such that the increase in power could be compensated for by thermal feedback alone.

It should also be noted that the operating mode for pebble-bed reactors does not require reactivity reserve, which means that the neutronic weight of the control rods can be limited, thus limiting the amplitude of the reactivity insertion in the event of inadvertent withdrawal of these rods.

b) Risk of oxidation of graphite due to air ingress

In the event of air ingress into the core, as described in Section 2/2/3, it is the characteristics of the system (limited air circulation) and the graphite (limited impurities) that prevent or limit oxidation of graphite and deterioration of the TRISO particles. Indeed, in pebble-bed or graphite-block designs, the 5-to-10-mm-thick graphite layer, which surrounds the area loaded with TRISO particles, protects these particles which, without this protection, could be vulnerable to external stresses (such as thermal transients).

c) Risk of corrosion of graphite due to water ingress

Currently, prevention of water ingress remains an issue under discussion, but it should be noted that developments in industrial power generation mean that it is now possible to use entirely gas-based systems. Nevertheless, sources of potential water ingress, such as the blower cooling systems, have yet to be examined.

d) Materials

While industry has materials that are tried and tested for temperatures characteristic of HTRs, R&D is underway on materials for structures and components that may be used for VHTRs (in particular, alloys containing chromium), with regard to issues such as their high-temperature strength or weldabilty.

e) In-service inspections

IRSN does not have experience feedback on in-service inspections for HTRs.

In terms of radiological protection, primary helium purification during reactor operation is a favorable aspect of the design, but risks associated with the deposition of carbon dust loaded with fission products in the primary system should be taken into account; this aspect is mention again farther.

Furthermore, in some projects, it is considered possible to exclude "cross vessel failure", *i.e.* a break in the large-diameter pipe section linking the reactor to the intermediate heat exchanger, which has countercurrent flow of "hot" and "cold" helium. However, the provisions for design and operation of such a component, in particular with regard to in-service inspections, must be specified before a position can be taken on this subject.

Accidents without core melt

a) Prevention of the core-melt risk

According to available studies, the design specifications used in the context of GIF work regarding the (low) power density, and the ability to remove the residual heat, mean that core melt can be "practically eliminated" for a V/HTR, in particular meltdown of the fuel itself (the kernel of the TRISO particles — with a melting point of approximately 2,700°C) with graphite degradation (as graphite does not melt).

Thus, as stated above, in the event of loss of cooling (the blowers stopping), the strong thermal feedback due to the graphite virtually stops the chain reaction, without the intervention of the reactor scram system. The residual heat is then removed by conductive and radiative heat transfer through the reactor vessel wall. However, for residual heat removal by radiative heat transfer to be effective, the concrete of the reactor pit must be maintained at a sufficiently low temperature by a specific system that, in some projects, may operate passively using natural convection. According to designers' analyses, the temperature reached by the fuel should then remain below the 1,600°C criterion. Such behavior is only possible thanks to the thermal inertia of the graphite (residual heat storage) and the low power density in the core, combined with the possibility of residual heat removal by natural mechanisms (radiative heat transfer from the vessels).

However, IRSN considers that failure of the reactor pit cooling system could lead to core temperature levels exceeding 1,600°C, at which significant damage of TRISO particles and significant releases of fission products would be possible. Therefore, the reliability of the reactor pit cooling system of a V/HTR would seem to be a key issue for the safety demonstration.

It should be noted that, in addition to the favorable neutronic characteristics of V/HTRs, designers also plan a passive reactor shutdown system, which would trip in the event of abnormal heating of the helium (using a Curie temperature magnetic system).

Finally, as stated above, prevention of significant damage to TRISO particles, *via* degradation of their protective graphite layer, shall be taken into account for the design of V/HTRs and explicitly covered in the safety demonstration.

b) Limiting the consequences of accidents without meltdown or severe core damage

The design-basis accident that could lead to radioactive releases into the environment is a loss of coolant accident (LOCA) leading to depressurization of the primary system (in a few minutes). Three specific points should be noted:

- The temperature of the fuel must not exceed the criterion of 1,600°C to ensure a very low level of radioactive releases from the fuel particles. This should be checked taking into account air ingress following depressurization, which could contribute to core heating *via* oxidation of graphite, in addition to the residual heat (see Sections 2/2/3 and 2/2/4-c with regard to preventing a graphite fire).
- Once the primary system is depressurized, there is nothing to drive later dispersal of fission products into the reactor containment (there is no safety injection of coolant in a V/HTR).
- As helium is non-condensable, it would seem unrealistic to confine it in the reactor containment when the primary system depressurizes: the gas will be released into the

environment *via* filters, as the core's radioactive content has not been released from the fuel at this stage.

With regard to this last point, the short-term radioactive content released into the containment in the event of a LOCA would mainly contain fission and activation products associated with carbon dust present in the primary system under normal operation, which will be partly resuspended. The concentration of contaminants flowing in the helium under normal operation will be very low, as determined by the purification system. In the IRSN analysis performed in the context of its participation in the EU's "ReActor for Process heat, Hydrogen And ELectricity generation" (RAPHAEL) project (Reference Document [12]), it was shown that while the elements associated with dusts are relatively well identified, the quantification of these elements is complex and has relied up until now on the mainly empirical analyses produced on this subject. Furthermore, the carbon-dust fraction potentially released into the atmosphere is currently difficult to assess, although the sensitivity of dust transfer mechanisms to the size of the break considered may be stressed. The very low dose values — of the order of ten microsieverts — given for HTR-Modul are associated with a branch line pipe break[28] (see Reference Document [13]). If larger breaks cannot be excluded, other phenomena could come into play that could lead to larger releases (possibility of a cliff-edge effect on the quantity of dust resuspended). In any event, "containment management" (based on the principles of static confinement and dynamic confinement — by the ventilation and filtration systems) is yet to be specified by V/HTR developers.

Accidents with core melt

While it seems that core melt in the strict sense (melting of the fuel kernels of TRISO particles, *i.e.* reaching a temperature of 2,700°C) can be "practically eliminated" for V/HTRs for the reasons given above, it is still necessary to assess the risk of significant fuel damage in the core, specifically deterioration of the sealing of the refractory layers coating the fuel kernels, which could cause large releases of fission products into the primary system or the containment. This risk must be assessed on the basis of deterministic and probabilistic safety analyses, taking uncertainties into account. In other words, there must be consideration of the possibilities of the fuel particles reaching temperatures above 1,600°C (see Section 2/2/1). It would seem that for V/HTRs, this large release of fission products in the core can be considered as a severe accident (as defined in Footnote 6), analogous to core melt for other designs. It should be noted that it is difficult to assess the

28
For this reactor, the bounding accident scenario corresponds to a break in a 65-mm-diameter pipe (branch line on the primary system). This scenario was also envisaged by AREVA for its ANTARES project. The consistency of this choice with the types of break used for the safety demonstrations on SFRs or EPRs merits reassessment.

resultant releases into the environment, in the absence of a clear strategy to limit these consequences on the part of the designers (*i.e.* a "containment management" strategy). However, it can be noted that, for the example of a LOCA as an initiator event, release of radioactive products in the core would not occur until after system depressurization, such that there would no longer be an effective drive mechanism to transfer radioactive products into the containment.

In any event, the absence of any true core meltdown risk for V/HTRs means that the risks associated with the flow of molten materials are excluded, in particular the risk of basemat piercing and resultant soil contamination.

IRSN has no knowledge of an accident scenario leading to significant releases adopted in the safety demonstration for reactors that have been built, other than the Japanese HTTR. For this reactor, a penalizing bounding accident scenario was used at the request Japanese Safety Authority ("regulatory approach"): this involves partial oxidation of the fuel with release of a fission-product content fraction similar to that used for core melt accidents on PWRs. Accident management is partly based on fission-product filtration.

For the HTR-Modul project, which was abandoned, the hypothetical bounding accidence scenario corresponded to a prolonged loss of power supplies, leading to the loss of reactor pit cooling. Use of a mobile system to inject water into the reactor pit cooling system was planned.

Radiological protection and waste management

At the current design stage, a few comments can be made regarding radiological protection and waste management.

The sealing characteristics of the fuel, and helium purification, should facilitate worker protection. However, attention should be given to analyzing the issue of areas in which primary-system contamination may be concentrated (blowers and heat exchangers).

In proportion to the electrical power produced, a V/HTR operating with enriched uranium produces similar types and quantities of waste to a PWR. Reactor projects operating at high burn-up (over 200 gigawatt-days per metric ton of UO_2) have been studied. This type of operation would have the effect of reducing the quantity of waste produced by V/HTRs and would mean that actinides from PWRs could be incinerated.

With regard to graphite, progress in its manufacture (such as reduced impurities and elimination of chlorine for graphite purification[29] meaning less chlorine-36 produced under irradiation) means that simpler reprocessing following irradiation can be envisaged than for Gas Cooled Reactors. In particular, the EU has funded the CARBOWASTE project for research into the "Treatment and Disposal of Irradiated Graphite and other Carbonaceous Waste".

However, the graphite in the various components (graphite compacts or pebbles, graphite blocks used as moderators or reflectors, cladding containing silicon carbide and carbon, matrices containing nitrogen compounds, and carbide or nitride fuels) would produce carbon-14 (with a half-life of 5,730 years) by neutronic activation in quantities much greater than those produced by reactors currently in operation. This radionuclide is relatively mobile in natural environments. This aspect should be taken into consideration for the safety and radiological protection of operators in the upstream fuel cycle, as well as for decommissioning and waste disposal.

2/2/5

Assessment of the concept with respect to the Fukushima accident

In the following sections, some qualitative information is given regarding the presumed behavior of a V/HTR (like HTR-Modul or ANTARES) in the case of events such as those that occurred on the Fukushima-Daiichi nuclear power plant. Possible means for the mitigation and management of accidents are also mentioned.

Earthquakes and floods

It is hard to assess the vulnerability of the V/HTR concept to a severe earthquake (stronger than the design-basis earthquake), as the general design of such reactor projects (in terms of buildings, reactor containment and equipment) is not sufficiently advanced.

However, one relatively intrinsic aspect of the V/HTR can be mentioned: an earthquake would have very little effect on the reactivity of a core using blocks. This type of core is not very compactable due to the solid nature of the moderator and the prism stacking. For a pebble-bed core, the effects of compaction should be analyzed to demonstrate that they would be compensated for by neutron feedbacks. Furthermore, a gravity absorber-ball injection system, whose effectiveness would be little altered by core distortion, is planned in some V/HTR projects. It

29 IRSN made this suggestion to AREVA during discussions on the ANTARES project.

should also be noted that in the event of loss of normal cooling, reactor power would be naturally reduced by neutron feedback.

Residual heat removal would not be affected by changes to the geometry or by significant core disorder (such as cracks and blockages) as the residual heat is mainly removed by conductive and radiative heat transfer through the reactor vessel wall.

Finally, flooding of the site and rooms of a V/HTR should not cause major difficulties for residual heat removal. It should be noted that, in some V/HTR projects, the power-generation system uses an helium-air turbine.

Nevertheless, the risks of air and water ingress into the primary system in the event of site flooding should be analyzed, bearing in mind that, in the event of a severe earthquake, one or more breaks could occur on this system.

Total loss of power or the heat sink

Consideration should be given to the aggravating factor of loss of the normal means of cooling, leading to loss of the cooling systems outside the reactor vessel. In such a situation, it is useful to estimate the time available before the maximum acceptable temperatures for the vessel and the surrounding structures (in particular the reactor pit) would be reached. In principle, given the thermal inertia of graphite, this would be several hours. This situation is to be studied in the context of safety tests planned for 2012 on the Japanese HTTR reactor (the OECD's Loss of Forced Cooling project, LOFR). It should be remembered that, for some designs, natural convection based systems could be used to cool the reactor pit. Furthermore, the ground around the reactor pit could also play the role of a heat sink (for a buried configuration), but in this case partial destruction of the reactor vessel is not excluded (depending on the residual heat to be removed). Simple emergency means could also be planned to ensure reactor pit cooling by direct spraying or by water injection *via* fire-service type connections planned in the design.

Severe-accident management

The question of the definition of a severe accident for a V/HTR has already been covered.

The results of the available safety analyses show that a LOCA does not lead to a requirement for coolant injection, as the core temperature would stabilize at around 1,600°C. However, this situation would be dependent on effective reactor-pit cooling, as mentioned above.

As stated previously, "containment management" in the event of a severe accident is yet to be defined.

Loss of cooling for the fuel storage areas

Given the low power density of the fuel assemblies used (a few tens of kW/m^3), natural convection cooling of stored fuel seems to be a robust solution.

2/2/6
Conclusion

In summary, the following points should be noted for V/HTRs:

- V/HTRs use as coolant a gas such as helium, which does not react with air or water.
- V/HTRs use a specific fuel: this involves a kernel of fissile material coated with several layers of refractory materials forming a millimeter-sized particle. This fuel, called TRISO, can withstand temperatures of up to 1,600°C. Furthermore, given the (low but non-zero) rate of particle sealing failure and the very large numbers of these particles in the core, continuous helium purification is planned. TRISO fuel has already been tried and tested in past HTRs and is currently used in two small experimental reactors (HTR-10 in China and HTTR in Japan).
- V/HTRs use large quantities of graphite in the core as moderators and neutron reflectors. Graphite also plays a role in fuel-particle protection, for example with regard to thermal transients. In contact with air, graphite can oxidize at varying rates depending on its purity. Special attention must be paid to the possibilities and consequences of air ingress into the reactor, both to prevent degradation of the graphite protecting TRISO particles and because a graphite fire could lead to significant radioactive releases. Thus, the possibilities and consequences of air ingress into the core constitute a key aspect of the V/HTR safety demonstration. According to available analyses, a graphite fire could be made highly improbable, but clearly this would need to be confirmed on the basis of such aspects as the detailed design options adopted on a V/HTR and the quality of the graphite used.
- The large mass of graphite in the core gives the system high thermal inertia such that, under certain accident conditions, operators have grace periods to repair or restore failed equipment.

- Furthermore, with suitable reactor and core design (especially by limiting the overall power of the reactor and the power density in the core), the system can present intrinsically favorable characteristics in terms of safety. In particular, the chain reaction can be virtually stopped by neutron feedbacks and the residual heat removed passively (*via* conduction inside the vessel and radiative heat transfer outside the vessel) without the fuel reaching temperatures that could degrade the sealing of its cladding layers. These intrinsic characteristics can also limit the reactivity consequences of water ingress into the core.

- However, in the various projects of which the IRSN is aware, the systems specifications (in terms of aspects such as architecture, redundancy, diversification and the use of active or passive systems) and the safety demonstration seem to be at an early stage and in need of more work. Although fuel melt as such seems highly improbable, which puts the V/HTR concept in a favorable position, IRSN has questions regarding possible scenarios for significant damage to the sealing of a large number of particles in the core, as well as the possible releases in the event of a LOCA, taking into account the possibilities for total failure of the cooling systems. Furthermore, "containment management" is yet to be specified. R&D is underway to identify the radioactive products that could be released from a V/HTR in the event of an accident, for example in the event of a LOCA leading to reactor depressurization, in particular regarding the carbon dust that may exist in the form of deposits in the primary system and trap fission products under normal operation and which would be resuspended from the first moments of depressurization.

- The dosimetry consequences of deposition in the primary system of carbon dust from the core, which could contain radioactive elements, need to be assessed, in particular with regard to in-service maintenance (including in-service inspection).

- While progress in graphite manufacture means that graphites containing less chlorine-36 after irradiation than those used in GCRs are now available, the production of carbon-14 is a major aspect to be taken into consideration for safety and radiological protection for downstream operations in the V/HTR fuel cycle, for the decommissioning of these reactors and for waste disposal.

- Flooding of the site and rooms of a V/HTR should not lead to major difficulties for residual heat removal, but the possibilities for air and water ingress into the primary system would need to be analyzed bearing in mind that, during a severe earthquake, one or many breaks could occur on this system.
- In the 1960s to 1990s, three power HTRs were operated (Peach Bottom and Fort Saint Vrain in the USA and the THTR in Germany), with favorable experience feedback in terms of radiological protection, although it is difficult to extrapolate this to future power V/HTRs. Since then, analyses and R&D have been pursued by designers and research bodies.

2/3
Gas-cooled Fast Reactors (GFR)

2/3/1
Presentation of the concept

The GFR concept aims to combine the advantages of a fast neutron reactor, with regard to the objectives of uranium conservation and minimizing final waste, with high-temperature operation for efficient power-generation.

The main design options associated with this concept have been specified in the context of the EU's FP6 Gas Cooled Fast Reactor (GCFR) project (see Reference Document [14]):

- a reactor producing thermal power of 2,400 MW;
- a power density in the core of 100 MW/m^3;
- helium as a primary-system coolant, at a pressure of 70 bar;
- mean core inlet and outlet temperatures of 400°C and 850°C respectively.

The core comprises hexagonal-cross-section assemblies made of refractory materials housing fuel pins. The fuel would be in the form of uranium and plutonium carbide pellets (10 metric tons of plutonium per GWe), with silicon carbide cladding. As experiments performed in HTRs have shown that rare-earth carbides diffuse rapidly in carbide pellets and corrode the silicon carbide, the cladding is to be made of fiber-reinforced silicon carbide with a thin refractory-metal liner for sealing, to protect the silicon carbide from corrosion by these rare-earth carbides. At equilibrium, the reactor would be an iso-generator of plutonium[30]. It would also be

[30] Equilibrium is reached when the composition of new fuel is constant at each refueling.

theoretically possible to transmute minor actinides in the core of a GFR.

The reactor design (Figures 7 and 8) comprises a metal vessel surrounding the core, connected to three primary-system loops containing intermediate heat exchangers. Three secondary-system loops, connected to the intermediate heat exchangers, provide power generation and could use gas turbines instead of steam turbines. In addition, tertiary-system water/steam loops are planned to supplement the power-generation system.

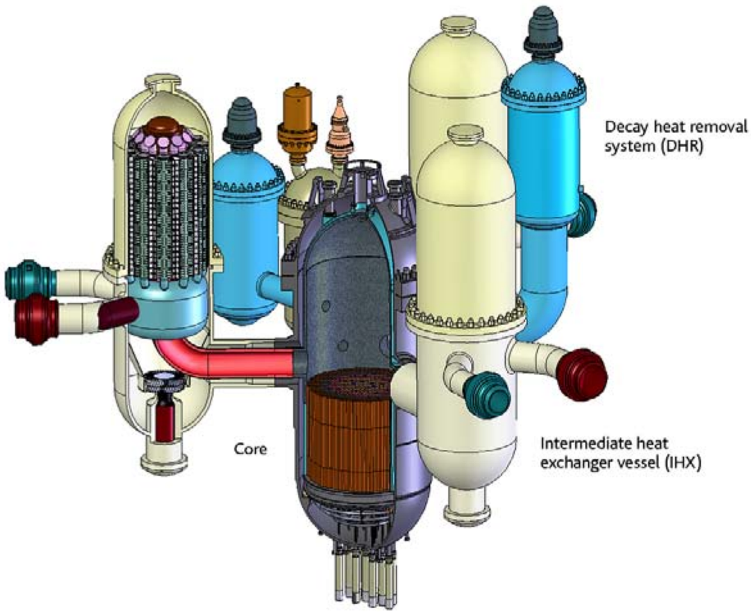

Figure 7
Schematic diagram of a (2,400 MWth) GFR – CEA data.

Furthermore, Decay Heat Removal (DHR) systems independent of the normal systems would be provided for residual heat removal. IRSN does not have information on the design of the reactor pit and the systems planned to ensure that the concrete would be kept sufficiently cool.

Finally, helium purification is planned during reactor operation.

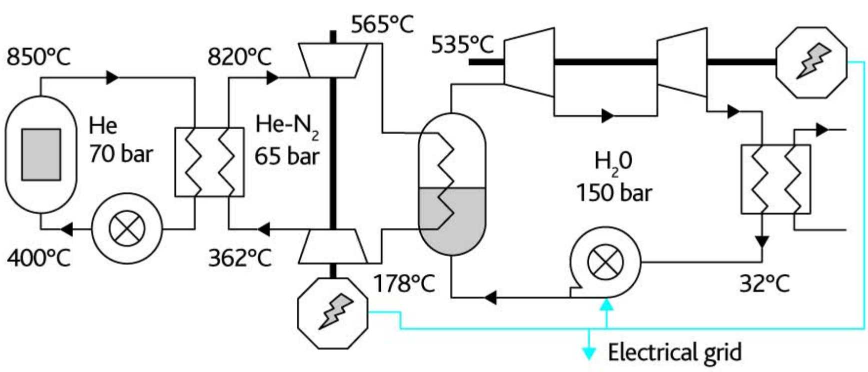

Figure 8
Architecture of GFR systems – CEA data.

2/3/2

Current state of GFR-concept development and outlook

The EU's Sustainable Nuclear Energy Technology Platform (SNETP) currently considers the GFR as an "alternative" solution to SFR technology. With this in view, an agreement was signed in 2011 between the nuclear research establishments of the Czech Republic, Hungary and Slovakia to design and build a 70 MWth experimental reactor (the ALLEGRO project[31]). The initial cores of this reactor would use Phénix-type assemblies with steel cladding and would consequently operate at a limited temperature (core outlet gas temperature of approximately 500°C).

The EU has funded GFR research work since 2005, with the GCFR and GoFastR projects.

Various aspects of reactor design were covered in the GoFastR project. For example, benchmark activities were defined for transient analyses using computer models. Furthermore, IRSN is leading a working group for Technical Safety Organizations (TSOs) within this project, covering the safety approach and accidents (in particular, depressurization and core accidents). The industry has organized a "mirror" group on the same subjects. In April 2011, R&D and safety approach support work for ALLEGRO was proposed[32] to the EU in the context of FP7 EURATOM projects, but it was not adopted. A reactor such as ALLEGRO would also be a facility for fast-neutron irradiation, which could serve the development of SFRs or even ITER.

Currently, given the objectives combining uranium-resource conservation, minimization of final waste and high efficiency, a key technical obstacle is the design of fuel assemblies — in particular the fuel itself and its cladding — that are able to withstand the specified operating conditions (in terms of fast neutron flux, power density and coolant temperatures). For this reason, the maximum acceptable temperature for GFR fuel is currently unknown, in contrast to that of the TRISO fuel for V/HTRs or the fuel pins for SFRs. Safety analyses, in particular those concerning residual heat removal in the event of depressurization, assume acceptable temperatures of 1,600°C to 2,000°C. Significant R&D work has been performed by the CEA. In particular, in 2007, irradiation capsules were inserted into the Phénix reactor: the FUTURIX-MI experiment for structural materials, and the FUTURIX-CONCEPT experiment for fuel and cladding materials. Currently, the Belgian Nuclear Research Centre (SCK-CEN) is responsible for the design and scheduling of

31
In principle, the ALLEGRO's limited power and operating temperature would lead to different safety characteristics than an industrial GFR, in particular for residual heat removal.

32
By the consortium formed by Hungary, the Czech Republic and Slovakia.

the irradiation of various potential GFR fuels (plates and pins) in the BR2 reactor, under the IrrDemo Program.

R&D is underway with regard to materials that can be used up to 850°C (covering V/HTRs and GFRs). Thermal shielding is also under consideration for some GFR structures, which could complicate the in-service inspection of such structures and be a source of loose parts in the reactor.

The helium purification system has also been the subject of development and specific research.

Generally, it can be said that the GFR could only be adopted on condition that the various innovations envisaged are validated in yet-to-be-built experimental facilities.

2/3/3
Safety aspects specific to the GFR

Compared with SFRs (which have risks associated with sodium), V/HTRs (which have risks associated with graphite) or even LFRs (which have risks of lead corrosion in particular – see Section 2/4 below), a GFR has fewer risks associated with the fluids and materials used: an inert gas (helium) is used as coolant and there are no materials susceptible to violent reactions with air or water under normal operation. However, under accident conditions at very high temperatures, all fuel-pellet and cladding materials can be oxidized by air or water.

In addition, compared with sodium, the use of helium in a fast reactor offers some safety advantages:

- a significant reduction in reactivity insertion in the event of a LOCA ("void effect");
- the coolant cannot change phase (it is a non condensable gas).

However, GFRs have some specific issues with regard to safety and these are discussed in the next section.

2/3/4
Aspects of the safety analysis

Normal operation, abnormal events and prevention of accidents

Prevention of helium leakage would seem to be made easier by the fact that it is an inert gas with no corrosive effects. However, it should be noted that at the high temperatures targeted, helium purity would be an important safety issue.

Prevention of accidents, particularly core damage, requires special attention on GFRs, due to the use of a gas as coolant and the low thermal inertia of the system. This concerns helium-leakage (LOCA) scenarios in particular. In the event of depressurization of a GFR, it is necessary to maintain coolant circulation to remove the residual heat, at least during the first few hours following reactor scram. For this, designers envisage using "close containment" (called "guard containment" on the GoFastR project) around the primary system, to provide a fallback pressure of around 10 bar in the event of a break in the primary-system pressure boundary. In the current design, this close containment would be a 33 m-diameter metal sphere, full of air or inert gas, at a pressure slightly above atmospheric pressure[33]. The quantity of residual heat removal available *via* natural convection alone is dependent on the fallback pressure. The value currently adopted, which is considered reasonable in terms of manufacturing feasibility, would not provide adequate residual heat removal in the short term following depressurization: low-power (battery-powered) forced convection systems would be required during the first few hours. Coolant circulation would be provided by DHR systems designed to cover the whole range of pressures from 1 to 70 bar (normal operation). Injection of a heavy gas, such as nitrogen or carbon dioxide, is being studied, with the objective of limiting the effects of depressurization in addition to the provisions aiming to maintain coolant circulation.

For all cases other than a hypothetical LOCA with "close containment", the DHR systems are to be designed and sized to provide core cooling by natural convection, without emergency power. However, it should be noted that this residual heat removal mode would not be totally passive, in particular as it would require valves to be operated.

Finally, special attention should be paid to the prevention of significant water ingress, an accident which would lead to reactivity insertion and oxidation of the fuel cladding.

Accidents without core melt

As for other fourth-generation reactor designs, the GFR is designed using the classic "barriers" approach (fuel, primary system, containment building) to reduce the risk of radioactive releases as far as possible, in particular for accidents without core melt. However, it should be noted that is not envisaged that the "close

33 Another solution would involve a reinforced-concrete containment building.

containment" be considered as a barrier for the safety demonstration.

For accidents without core melt, the bounding accident scenario with regard to the potential consequences has not yet been specified. As for V/HTRs, a LOCA leading to the release of helium into the "close containment", followed by leakage of this helium into the reactor hall and ultimately into the environment should be considered. IRSN is not aware of any analysis of such a scenario. However, the primary helium should contain very little contamination under normal operation, due to its purification and the strict requirements on cladding sealing (with a liner).

Accidents with core melt

It seems that total or partial core meltdown on a GFR needs to be considered. However, the materials that comprise the core have varied high-temperature behaviors; they may melt, break down or sublimate. IRSN is not aware of any analyses on GFR core meltdown, or on means of limiting the consequences of such a meltdown. However, several remarks can be made on accidents with core melt:

- In contrast to in SFRs, if a molten pool forms, it cannot be efficiently cooled by the coolant (helium). A spreading area would be necessary, as in third-generation PWRs such as the EPR and the AP1000.
- As for SFRs, there is a risk of recriticality.
- Thermal radiation from the high-temperature core could degrade the surrounding structures without them being in contact with molten materials, a phenomenon which does not exist for SFRs due to the presence of sodium. The thermal shielding mentioned above should contribute to protecting such structures under these conditions.
- Unlike SFRs, where only small amounts of the iodine present the fuel pins are transferred into the primary system's cover gas if the cladding fails, GFRs do not benefit from this favorable effect provided by sodium.

These aspects highlight additional difficulties for the management of accidents with core melt on GFRs.

2/3/5

Assessment of the GFR-concept with respect to the Fukushima accident

Earthquakes and floods

In principle, in terms of neutronics, GFRs are as sensitive to seismic loads as SFRs. In terms of mechanics, the refractory core and internal structures are lighter than those of SFRs, PWRs or V/HTRs, because ceramics have about half the density of steels. In the event of an earthquake, this would provide an advantage by reducing the inertial forces transferred onto the core support structures.

Given the risk of a LOCA in the event of an earthquake, a "hardened safety core" seismic scram system, with a suitable detection threshold, would be appropriate for shutting down the reactor as soon as possible. Also, as mentioned above, forced convection in the core would initially be indispensable to remove the residual heat in the event of a large break, whence the need for a preinstalled "hardened safety core" power supply.

Total loss of power or the heat sink

All the residual heat removal systems require water (there are no air heat exchangers — cooling-water pools are planned for the DHR). As loss of the heat sink could lead to core melt in the short term, a preinstalled "hardened safety core" heat sink may be necessary.

In the event of total loss of power supplies, residual heat removal could be performed by emergency systems operating by natural convection. However, the operation of active equipment items (such as valves) implies that these receive emergency power from batteries. All this equipment must be of "hardened safety core" robustness.

Severe-accident management

As mentioned above, the consequences of core overheating and degradation are not well understood. Therefore, IRSN cannot currently take a position on this aspect for GFRs.

With respect to a possible LOCA (helium leakage), it would seem necessary to exclude the possibility of water injection into the primary system (due to the risk of vapor explosion).

2/3/6

Conclusion

In summary, the following points should be noted for GFRs:

- GFRs do not have any risk of violent reactions between the coolant and air or water.
- Due to the use of a gas (helium) as coolant and the absence of moderators in the core (in contrast to V/HTRs), GFRs do not benefit from high thermal inertia.
- Use of a gas as coolant also makes GFRs very sensitive to situations such as a LOCA. In such situations, the safety cannot be totally based on passive systems: active systems are indispensable. Because of this, reactor design and systems architecture may be complex.
- With regard to the possible consequences of an accident with severe core damage, GFRs do not benefit from the trapping of volatile fission products that occurs in SFRs thanks to the sodium.
- It is currently difficult to have an opinion on whether the safety level is at least equivalent to that of third-generation reactors.
- With regard to events such as those that occurred at Fukushima, the characteristics of GFRs would mean that, from the design stage, there would be a need for automatic safety systems including support systems, such as power supplies and heat sinks, that are of "hardened safety core" robustness and able to operate very quickly.
- There is currently no fuel qualified for the temperature levels planned for the coolant (850°C at the core outlet). This constitutes a technological obstacle.
- Materials that can withstand the temperatures planned for normal operations are the subject of R&D work, which is also being performed for V/HTRs. However, in the event of core melt in a GFR, important structures could be severely stressed by direct thermal radiation from the damaged core. Thermal shielding is planned, but this could complicate in-service inspection and be a source of loose parts in the reactors.
- No GFR has ever been built.

2/4

Lead-or lead/bismuth-cooled Fast Reactors (LFR)

2/4/1

Presentation of the concept

LFRs are fast reactors cooled by a molten metal such as lead, or a lead-bismuth alloy often called LBE (for Lead Bismuth Eutectic).

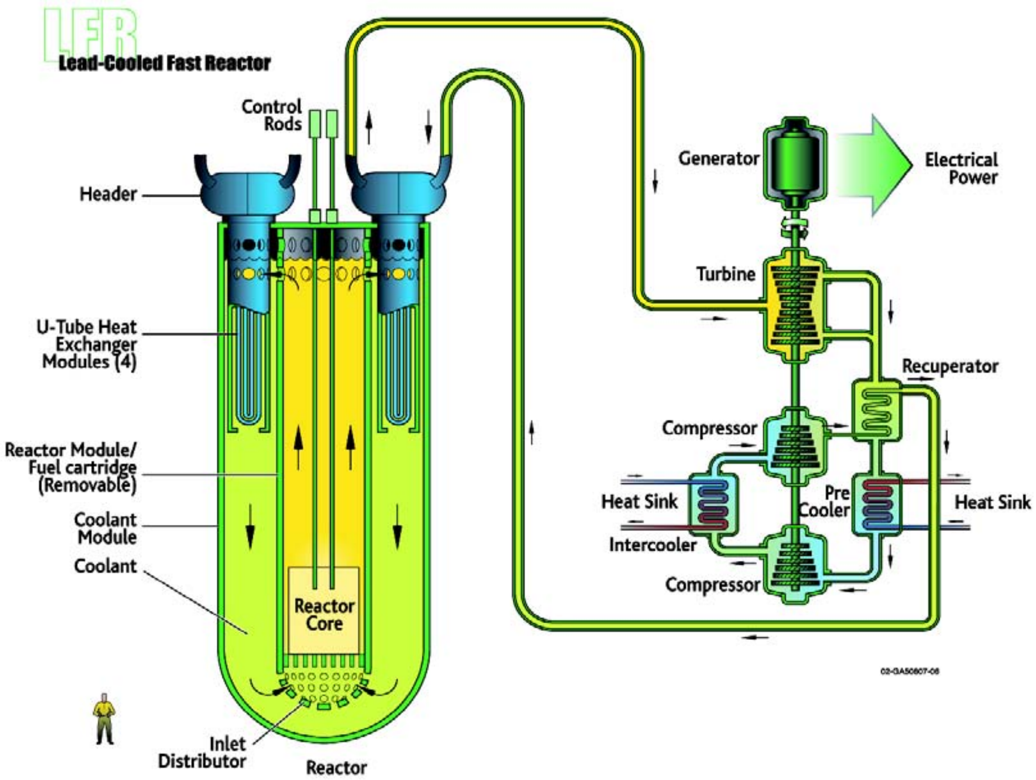

Figure 9
Schematic diagram of a lead-cooled reactor.

As for SFRs, LFRs can operate with low-pressure systems due to the high boiling point of lead or LBE (1,745°C for lead, 1,670°C for LBE).

In this Section, IRSN uses the LFR conceptual design of the European Lead-cooled SYstem (ELSY) project developed as part of the EU's Sixth Framework Programme (FP6) (see Reference Documents [15], [16] and [17]). Figure 10 gives a schematic diagram of the ELSY primary system.

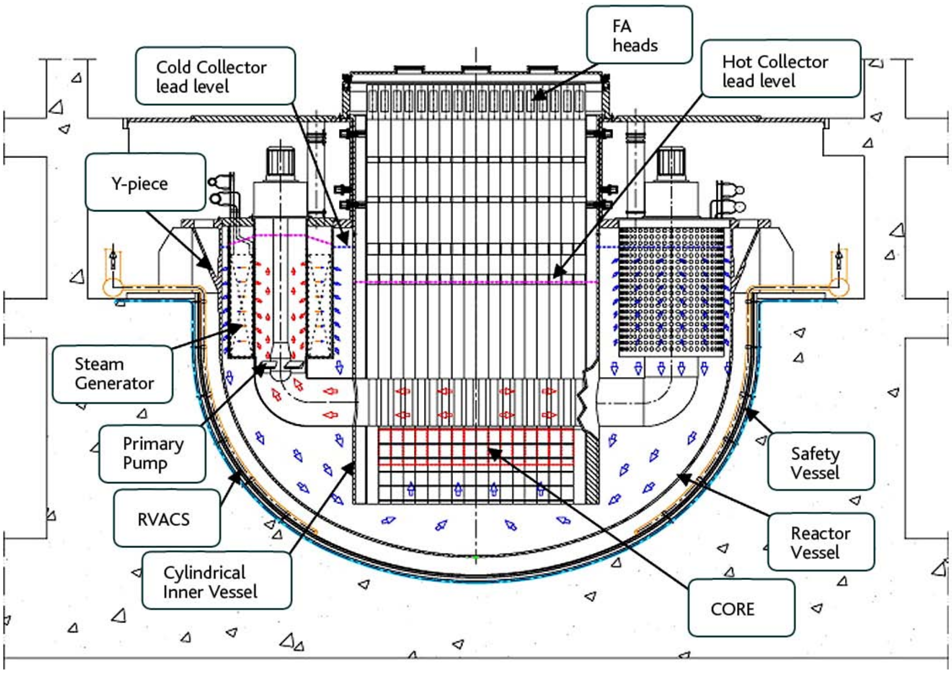

Figure 10
Schematic diagram of the ELSY primary system.

As for the French SFRs Phénix and Superphénix, the ELSY project is "pool-type": the primary lead is confined in the (main) vessel.

In contrast to sodium, one advantage of lead or a lead-bismuth alloy lies in the absence of any violent chemical reaction with air or water. Because of this, the designers of ELSY have opted for a design without an intermediate system, with six in-vessel steam generators.

The thermal and electrical powers of ELSY are 1,500 MW and 600 MW respectively, with the temperature of the lead coolant ranging from 400°C to 480°C. Lead circulation is provided by submerged pumps.

Lead and LBE are weak moderators and neutron absorbers but excellent reflectors, properties which enable the designers to "space out" the fissile material in the core. Compared with SFRs, this enables a higher proportion of coolant to be used in the core, larger cross-section coolant paths and a lower coolant speed (2 m/s) to limit the structural erosion. With respect to SFRs, this also means lower head loss across the core which promotes natural convection of the coolant.

The power density in the core is approximately 110 MW/m^3.

MOX fuel with steel cladding is used in the ELSY project. The fuel-pin bundles are laid out in a square pitch, forming "open" assemblies like in PWRs. Another solution is the one used in SFRs (fuel pins assembled into closed hexagonal housings).

Nitride- or carbide-based fuels, with higher densities than MOX, can also be considered for LFRs. They would have the advantage of operating at lower temperatures (below the boiling point of lead or LBE), of having no risk of forming oxides with lead or LBE in the event of cladding failure, and of reducing the loss of reactivity during a fuel cycle, which means that the neutronic weight of the control rods can be reduced, reducing risks in the event of inadvertent removal of these rods.

Due to the very large mass of lead involved (6000 metric tons), the mechanical design of a reactor such as that of the ELSY project presents a challenge, especially with regard to seismic loads. In particular, this has led to reducing the ratio between vessel height and diameter as much as possible.

The safety vessel, whose internal wall is insulated, is "anchored" in the reactor pit. The space between the two vessels is maintained at a sufficiently low temperature by the Reactor Vessel Air Cooling System (RVACS). Such a system may be sufficient to remove relatively low residual heat, in particular for reactors of low operational power or, in the case of ELSY, after the first month of reactor shutdown (see Reference Document [17]). In the case of the ELSY project, the additional Direct Reactor Cooling System (DRCS) is therefore needed to remove the residual heat from the reactor in the first few weeks following shutdown. Reference Document [16] also mentions the Reactor Concrete Cooling System (RCCS) which cools the reactor pit walls.

2/4/2

Current state of LFR-concept development and outlook

Reactors using LBE coolant were developed and built in the Soviet Union for submarine propulsion. Seven Alfa-class military submarines powered by 155 MWth OK-550 reactors, and later BM-40A reactors, using highly enriched uranium-235 fuel were operated between 1967 and 1983. In 2007, the IAEA published an experience feedback document on fast reactors (Reference Document [5]), which includes LFRs and gives details of notable events that occurred on three submarine reactors:

- An accident in 1968: the core was blocked by oxides of LBE (in particular), which had accumulated mainly during outages for maintenance on the depressurized primary system, causing partial core meltdown due to a lack of suitable procedures in response to the signals received in the control room. There were deaths due to acute radiation exposure. The most notable provision made on submarine reactors after this event was a "hardening" of the procedures for monitoring oxygen in the primary system and for purification to remove any oxides formed.

- An accident in 1971: damage to primary-system pipework was observed, due to corrosion of the external surface of these pipes caused by excessive humidity in the reactor compartment due to a lack of leaktightness on the steam generator. This led to a leak of radioactive LBE.
- An accident in 1982: widespread corrosion of the steam generator tube bundle was caused by poor-quality feedwater. This led to steam ingress into the primary system containing the LBE. At the end of a fairly complex chain of events, including human errors, 150 liters of radioactive LBE leaked into the reactor compartment.

There is no operational experience on an industrial lead- or LBE-cooled fast reactor anywhere in the world.

Russia continues to be interested in this type of reactor and is developing two prototypes: a lead-cooled reactor (the 300 MWe BREST-OD-300) and an LBE-cooled reactor (the 100 MWe SVBR-100). According to information recently gathered by IRSN, commissioning of both reactors is planned around 2020.

The EU's Sustainable Nuclear Energy Technology Platform (SNETP) has adopted the LFR, alongside the GFR, as an "alternative" solution to SFR technology. In consequence, the ELSY project was launched in 2006, followed by the Lead-cooled European Advanced DEmonstration Reactor (LEADER) project in 2010, both funded by the EU as part of FP6 and FP7 respectively. The LEADER project aims to provide the conceptual design for an industrial-scale LFR demonstrator called the Advanced Lead Fast Reactor European Demonstrator (ALFRED), which Romania wishes to build on its territory by 2025.

Additionally, the Belgian Nuclear Research Centre (SCK-CEN) is going to build a demonstration reactor, called the Multi-purpose hYbrid Research Reactor for High-tech Applications (MYRRHA), for an Accelerator Driven System (ADS) using a particle accelerator to drive a sub-critical nuclear fission reactor. The reactor is to use LBE as a coolant (and as a spallation neutron source activated by the beam of protons). A preliminary design should be the result of the Central Design Team (CDT) project included in FP7. Submission of the MYRRHA safety options file to the Belgian Nuclear Safety Authority is planned for 2014. Reactor commissioning is planned for 2024.

Finally, there are projects for small-scale reactors in the USA, with the 45 MWth "Small, Sealed, Transportable, Autonomous Reactor" (SSTAR), and in South Korea, with the "Proliferation-resistant, Accident-tolerant, Self-supported, Capsular and Assured Reactor" (PASCAR), and for experimental reactors in Sweden, such as the 0.5 MWth European Lead Cooled Training Reactor (ELECTRA).

2/4/3

Safety aspects specific to the LFR

Risks of structural erosion and corrosion

This seems to be a key issue for LFR-type reactors.

Lead is highly eroding and for this reason its speed is limited to approximately 2 m/s. Lead is also highly corrosive for steel structures (in particular, for the cladding and vessels). Furthermore, lead oxides can lead to blockages in fuel assemblies. The method used on the Alfa-class submarines, which remains the baseline solution, was to create a layer of iron oxides on the surface of steels in contact with lead, by injecting oxygen into the lead, combined with continuous purification to remove the lead oxides created. The solubility of iron oxides in the lead depends on oxygen concentration and temperature, which leads to different behaviors between fuel-pin cladding and the main vessel for example. For the ELSY project, the temperature of the lead at the core outlet has been set at 480°C to limit the risk of structural corrosion on the primary system[34]. According to Reference Document [16], operation at typical SFR temperatures (550°C at the core outlet) would create a corrosion risk, at least at the current state of knowledge regarding the compatibility of the steels qualified for SFRs with lead or LBE. With regard to the fuel cladding, where operating temperatures would reach approximately 560°C, Reference Document [15] states that the corrosion-risk management method mentioned above would not be adequate and that the cladding would require coatings.

Given the diversity of materials and operating temperatures in an LFR, corrosion-risk management, which requires even distribution of

34 Although not explicitly mentioned in the reference documents consulted, it would seem that a system for injecting oxygen into the primary system would also be indispensable.

the injected oxygen in the whole primary system, could be complex for large-scale reactors like ELSY.

Finally, given the risk of lead corrosion, it is difficult to imagine that the components could remain in the reactor for several decades. The possibility of replacing the in-vessel primary coolant pumps and steam generators is an objective that has been adopted for the ELSY project (see Reference Document [16]).

Risk of embrittlement of steels in the presence of lead

The report in Reference Document [25] shows that, independent of corrosion risks, certain steels could become embrittled in contact with lead or LBE, which could be exacerbated by irradiation. The report mentioned previously (in Footnote 8) concerns T91 steel, which is a candidate for ADSs. It would therefore seem necessary that designers and associated R&D bodies examine this subject in depth, so that the risk of sudden failure of components such as a reactor vessel under accident loads can be excluded *via* a suitable choice of materials and metallurgy processes.

Risks associated with slow chemical reactions between lead or LBE and air or water

In the event of air ingress into a lead or LBE system, or in the event of a small water leak from a steam-generator tube, compounds such as oxides, hydroxides and hydrides would form in the lead or LBE. These would increase its viscosity, reduce its heat-exchange properties and lead to overheating or even fuel meltdown, as occurred during the 1968 accident on a Soviet submarine reactor.

Risk of coolant freezing

Lead freezes at 327°C, which is only 73°C lower than the core inlet temperature during LFR operation, while for SFRs the margin is approximately 300°C. This requires heat devices for all the systems that contain lead, to prevent lead freezing during reactor outages in particular. As with sodium, lead expands when it melts. IRSN does not have information on the consequences of lead freezing or melting in a LFR.

The main advantage of LBE is that it freezes at 125°C, which is much lower than the freezing point of lead. This means that the freezing risks for an LFR using LBE are in principle much easier to manage. However, it should be mentioned that LBE froze in the reactors of three Soviet submarines (other than those mentioned in Section 2/4/2, which had accidents), including one when it was under the icecap. This led to their permanent shutdown.

Chemical and radiological risks

Lead is highly toxic. Its Occupational Exposure Limit (OEL) is 20 times lower than that for sodium hydroxide (a compound considered with regard to chemical releases from a sodium fire in an SFR).

Furthermore, the use of LBE would lead to significant production of polonium-210, which is highly toxic. However, it should be noted that, even if lead were used, polonium-210 would still be produced due to the presence of bismuth impurities, albeit in significantly lower quantities. The choice of lead as the coolant for the ELSY project was mainly aimed at reducing production of polonium-210.

Risk of thermodynamic interactions

While lead and LBE do not react violently with air or water, contact between these molten metals and liquid water could lead to "vapor explosion" thermodynamic interactions, shockwaves and the displacement of structures and fluids. In particular, this led the designers of the ELSY project to adopt several lines of defense, especially *via* design provisions for the in-vessel steam generators, to "practically eliminate" not only the possibility of serious consequences on the reactor's internal structures but also gas ingress into the core in the event of a steam generator tube rupture (see Reference Document [16]). Furthermore, outlets with blowout diaphragms are connected to the cover-gas space over the lead, which acts as an expansion volume to dampen the overpressure[35]. It should be noted that the risk of overpressure in the reactor could be more radically excluded by adopting an intermediate system that uses a low-pressure fluid that cannot interact thermodynamically with the primary lead.

Initial analysis suggests that any contact between molten fuel and liquid lead would probably not cause a high-energy "vapor explosion". However, this would need to be confirmed.

2/4/4

Aspects of the safety analysis

Normal operation, abnormal events and prevention of accidents

Experience feedback on LFRs is limited to a small amount of information available about submarine nuclear reactors from the former Soviet Union.

As stated above, the management of corrosion risks for metal structures, in particular those that form the primary-system

35 However, it is not stated in the document consulted whether these outlets release directly into the reactor hall or into special containers.

pressure boundary, constitutes a key issue and could be complex for a large-scale reactor like ELSY. For future LFRs, this issue must be resolved and it must be possible to perform in-service inspections on the structures, in particular to check that there is no harmful corrosion. With regard to this, lead is opaque (like sodium) and specific techniques need to be developed for inspecting the internal structures of the primary system. However, it should be noted that in-service inspections on the ELSY project should be facilitated by the relative simplicity of the internal structures, in particular the absence of a diagrid supporting the core (as the fuel assemblies are suspended). The "sizing" of the space between the two vessels on the ELSY project should be a compromise between the need to introduce devices for inspecting the two vessel walls and the need to avoid too large a drop in the lead level in the main vessel in the event of leakage from it.

Furthermore, it has been stated above that the possibility of replacing the major in-vessel components (coolant pumps and steam generators) has been adopted as a design objective for the reactor.

Accidents without core melt

As for SFRs, LFR cores could experience reactivity insertion due to the "void effect", for example in the event of lead overheating (due to fuel assembly blockage) or uncovery. Although this effect is weaker on LFRs than on SFRs due to the coolant density, IRSN has no information on the management of such an accident for the projects currently underway. Nevertheless, it can be noted, with regard to the coolant boiling risk, that the margin is greater on LFRs than on SFRs as the boiling point of sodium is 900°C compared with 1,750°C for lead (which is above the melting point of the fuel-pin cladding and steel structures).

LOCAs have been analyzed (see Reference Documents [13] and [16]). In this respect, the combination of a core with moderate power density, high thermal inertia[36] and use of a coolant with a very high boiling point makes the LFR design tolerant with regard to LOCAs, including in the event of scram failure. Even under such conditions, the melting point of the cladding would not be reached according to Reference Document [16].

Some comments are made below on the safety functions "Control of reactivity", "Removal of heat from the core", and "Confinement of radioactive materials".

[36] Essentially just the same as for the sodium in an SFR, for the same coolant volume.

a) Control of reactivity

There is no information available on the design of the control rods but the design should be such that they can be inserted quickly into the core despite the high density of lead (twelve times that of sodium).

b) Removal of heat from the core

As stated above, the ELSY design facilitates core cooling by natural convection in the event of primary-coolant pump shutdown, for example in the event of loss of normal and emergency power supplies. Low head loss across the core and lead's high density are factors that promote natural convection in the primary system.

The sources consulted (Reference Documents [15] and [17]) give different architectures for the ELSY project's residual heat removal systems. According to Reference Document [15], in the event that the normal residual heat removal systems are unavailable, two redundant and diversified systems, which are said to be able to operate passively, remove the residual heat. One of these systems comprises two loops, each having a lead-water heat exchanger immersed in the primary system, into which water is fed by gravity from a tank, which then evaporates and is released into the environment *via* a stack. The second system also has two loops, each using a steam generator as a heat exchanger. IRSN has no specific information on the design requirements and possible operating modes for the two RVACS and RCCS cooling systems for the space between the two vessels and the reactor-pit concrete respectively.

The high thermal inertia of an LFR strongly limits coolant temperature-rise kinetics in the event of total loss of cooling systems, whence providing significant grace periods (several hours) for installing means of residual heat removal before the reactor reaches temperatures that could lead to core damage.

c) Confinement of radioactive materials

Given the high toxicity of lead, its containment requires special attention to prevent releases into the environment as far as possible (design provisions, in-service inspections, etc.).

Accidents with core melt

Available analyses on accident sequences performed for the ELSY project (see above) do not reveal any scenarios that may lead to core melt. However, it should be noted that, as for SFRs, prolonged total loss of all residual heat removal systems could lead to vessel failure by creep and possible failure of the reactor pit, an event whose consequences have not been analyzed. However, the high thermal inertia of an LFR's primary system provides a significant grace period to repair some of these systems, or install others, before damage occurs.

IRSN does not have information on the behavior of molten fuel and its relocation in an LFR. Molten MOX fuel should float on the primary lead and, for this reason, a "core catcher" at the bottom of the reactor would be of no use. IRSN has no information on the risk of recriticality of corium floating on the primary lead.

Furthermore, as stated in Section 2/4/3, special attention has been paid in the ELSY project to the "practical elimination" of reactor damage in the event of a steam generator tube rupture. The associated design provisions have been supported by simulations performed using the SIMMER simulation tool (Reference Document [16]). However, with respect to experience feedback concerning steam generators, use of the rupture of only one steam generator tube in the analysis would clearly require substantiation.

Finally, in principle, lead (like sodium) can trap fission products (such as iodine), which are significant contributors to the radiological consequences of an accident.

IRSN does not have information on the design of the third containment barrier (in particular, the reactor building and ventilation systems) for the ELSY project.

2/4/5

Assessment of the concept with respect to the Fukushima accident

Earthquake

Earthquake is a hazard which requires design precautions on LFRs. Movements of fuel assemblies could cause reactivity insertions and the considerable mass of lead contained in the reactor vessel could impose large loads on structures, especially given that an earthquake could cause lead displacements with wave effects. In the ELSY project, an aseismic isolation system using anti-seismic pads under the reactor is planned and an EU project on the subject was launched in 2012: "Seismic-Initiated events risk mitigation in Lead-cooled Reactors" (SILER).

As for SFRs, a "hardened safety core" seismic scram system should be planned.

Flooding

Unlike for SFRs, there is no risk of chemical interactions between lead and water. However, simply because of the highly toxic nature of lead, robust prevention (with "hardened safety core" construction provisions) of the risk of contact between lead and water should be planned, such as that described in Section 2/1/5 for SFRs.

Total loss of power supplies

As stated above, the ELSY project seems to have passive residual heat removal systems (until the water in the tanks is exhausted) and to have the ability to circulate lead by natural convection in the core, which, in terms of safety, positions it favorably with respect to other designs (especially GFRs) in the event of total loss of power supplies. However, as also stated above, IRSN lacks information on the requirements adopted by the designer concerning the cooling of the vessel walls and the reactor-pit concrete.

A difficult situation for LFRs could result from loss of the means of heating the lead contained in the primary system when the reactor has been shut down for sufficiently long that its residual heat is insufficient to prevent the lead from freezing. Maintaining the integrity of a vessel containing solidified lead is a subject that needs analysis. Use of LBE would have fewer risks in this respect.

Loss of the heat sink

In the ELSY project, there are two heat sinks: sea or river water for the turbine (the normal means of heat removal) and the atmosphere for the residual heat removal systems. For these latter systems, the stacks to discharge water vapor could constitute a sensitive area with respect to hazards. The possibility of water top-up into the tanks associated with one of the two residual heat removal systems would seem to be necessary ("hardened safety core" provision).

Severe-accident management

For situations similar to those that affected reactors at Fukushima-Daiichi, LOCAs should be considered. The issue is quite similar to that for SFRs. It is hard to envisage supplementing the lead in the reactor vessel using equipment connected in an emergency, although the risks of such an operation would be lower than for SFRs as lead does not react violently with air or water. Preinstalled, "hardened safety core" provisions would be desirable, in particular for reinjection into the main vessel of any lead that may leak into the safety vessel.

IRSN has no information on the possibility of rapid core unloading.

2/4/6
Conclusion

In summary, the following points should be noted for LFRs:

- Compared with sodium, lead has the advantage of not reacting violently with air or water.
- However, even without a violent reaction, contact between air or water and lead or LBE would cause an increase in their viscosity, which could cause overheating of fuel in the core.
- As for sodium, liquid lead can cause thermodynamic interactions when in contact with easily vaporizable fluids (such as water), which would cause overpressures. In this respect, the LFR design with in-vessel steam generators would require special precautions.
- Depending on the materials envisaged, detailed research may be required on the risk of lead or LBE causing embrittlement of metal structures resulting in their sudden failure.
- Lead is highly toxic. The risk of releases in the event of leaks from systems or components containing lead therefore requires special attention, and should be "practically eliminated" in the event of flooding.
- Given the specific properties of lead, it is possible to design cores with lower power densities than for SFRs, which are less sensitive than SFRs to loss of cooling and which promote natural convection.
- The large mass of lead gives LFRs high thermal inertia and available analyses regarding the ELSY project do not reveal any core-melt initiators: in the event of total loss of residual heat removal systems and scram failure, the melting point of the cladding should still not be reached. However, IRSN finds it hard to pronounce on the exhaustiveness of the analyses performed, and if core melt cannot be excluded, the accident phenomena would be quite original as the corium would float on the surface of the lead. No analyses are available on this subject.
- The risk of structural corrosion by the lead is a crucial aspect which, at the current state of knowledge, limits the operating temperatures (460°C at the core outlet). While the risk-management provisions made on Soviet submarine reactors

would seem to have been validated by experience, it may be more complex to design analogous provisions for large-scale LFRs like the ELSY project.

- Experience feedback is limited to Soviet submarine reactors. Three of these reactors suffered "serious damage".

2/5

Molten Salt Reactors (MSR)

2/5/1

Presentation of the concept / Current state of development

The MSRs studied fall into two groups: reactors in which the molten salt acts only as a coolant and reactors whose fuel is dissolved in a lithium-fluoride-based eutectic molten salt (FLi, FLiNaK or FLiBe[37]). For this second type of MSR, the molten salt contains a mixture of natural thorium[38] (thorium-232) and uranium-233. The reactor operates by fission of the uranium-233 produced from the thorium-232 (the fertile material) and therefore requires an initial loading of uranium-233 or plutonium for start up.

Dissolved-fuel MSRs are a completely different concept than other fourth-generation reactors, in particular because the fuel and coolant are mixed together. Furthermore, a special fuel reprocessing unit must be associated with such a reactor to continuously eliminate neutron-absorbing elements, to prevent the chain reaction from being stopped by the generation of neutronic poisons in the coolant.

MSR designs can include both thermal and fast neutron reactors. The size of the reprocessing unit and its coupling with the reactor strongly depends on the neutron spectrum used. Reprocessing-unit and reactor operations are less correlated for fast neutron reactors than for thermal ones, as the neutronic poisons have more effect on the latter.

As for SFRs and LFRs, MSRs use a low-pressure coolant.

Two experimental MSRs were built and operated in the USA. The first was a reactor designed for military aircraft propulsion, built during the 1950s as part of the Aircraft Reactor Experiment (ARE) project. The second was the Molten Salt Reactor Experiment (MSRE), which was built at the Oak Ridge National Laboratory (ORNL) in 1962 and reached criticality in June 1965. It did not use a

37
FLiNaK is a LiF-NaF-KF mixture, with a melting point of 454°C; FLiBe is a LiF-BeF_2 mixture with a melting point of 459°C.

38
Thorium is four time more abundant than uranium. There is a major thorium deposit in Brittany.

fertile material, but rather fuels based on uranium-235, plutonium and then uranium-233. This 8 MWth reactor was shut down in 1969 after approximately 13,000 hours of operation.

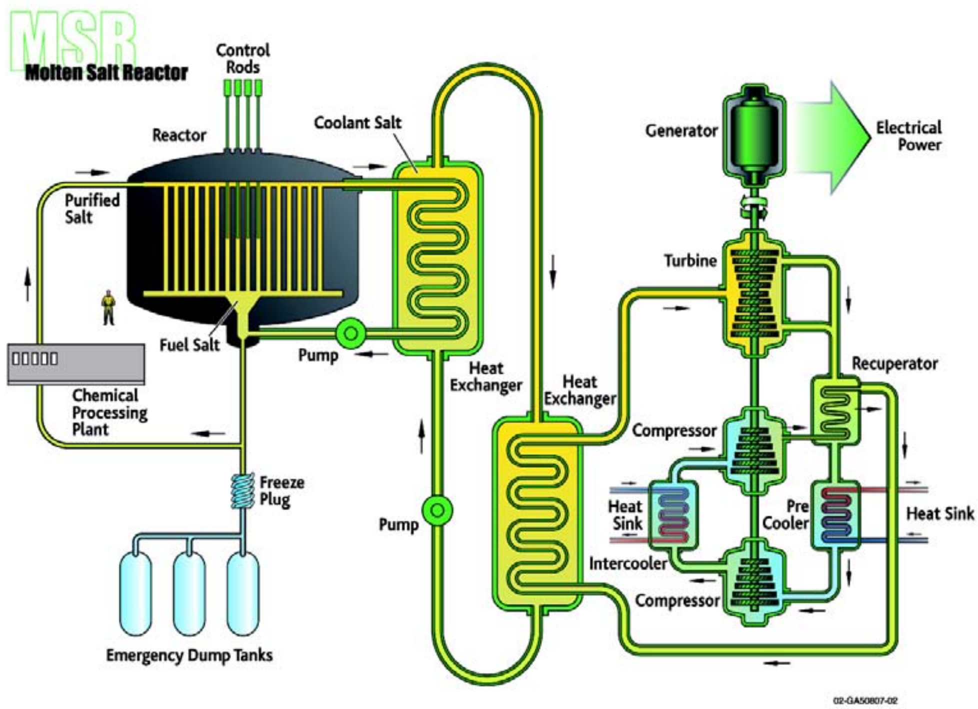

Figure 11
Schematic diagram of a reactor where the fuel is dissolved in molten salts.

The ARE project encountered difficulties associated with structural corrosion by the salt. Experience feedback led the designers to develop a procedure to manage this phenomenon by controlling the salt's redox potential. This procedure was implemented on the MSRE, which does not seem to have encountered corrosion problems.

Studies were then performed until 1976 at ORNL as part of the Molten Salt Breeder Reactor (MSBR) project, a planned prototype 2,500 MWth reactor.

Five countries are currently taking an interest in MSRs:

- France, where the National Centre for Scientific Research (CNRS) has been involved since 1997. The CNRS started by producing a reassessment of the MSBR design and is currently studying the Molten Salt Fast Reactor (MSFR) design, a fast breeder reactor (which is also able to incinerate the transuranium elements produced in current reactors).

- Russia, where the Rosatom Nuclear Energy State Corporation has developed the "Minor Actinide Recycling in molten Salt reactor" (MARS) project, which is also a fast reactor (with a view to incinerating minor actinides).
- Japan, where Mitsubishi is studying a small-scale (350 MWth) MSR thermal reactor, the FUJI-12, which is very similar in design to the MSBR.
- The USA, where the Oak Ridge National Laboratory produced a catalogue of technology options for Fast-Spectrum Molten Salt Reactors (FS-MSRs) in 2010.
- China, which committed itself to an MSR project in 2011 (with funding of $250m).

Furthermore, the EU’s "Evaluation and Viability Of Liquid fuel fast reactor" (EVOL) project was launched in early 2011 as part of the FP7-EURATOM-FISSION program. In particular, it aims to produce the preliminary design for the MSFR and an initial safety approach.

The discussions below are based on information obtained from the CNRS concerning the MSFR. The 3,000 MWth MSFR is to use uranium-233 and thorium-232 as fuel, containing 18 m^3 of (LiF-ThF4-UF4) salt fuel, with 9 m^3 in the core area. Due to its high boiling point (1,800°C), this salt can reach temperatures of 700°C to 750°C without requiring high pressures, which means that thermodynamic efficiencies of around 50% can be envisaged. The core comprises a space filled with salt fuel inside the first barrier (the "core chamber"), which is itself located inside an inerted vessel containing the primary-system components.

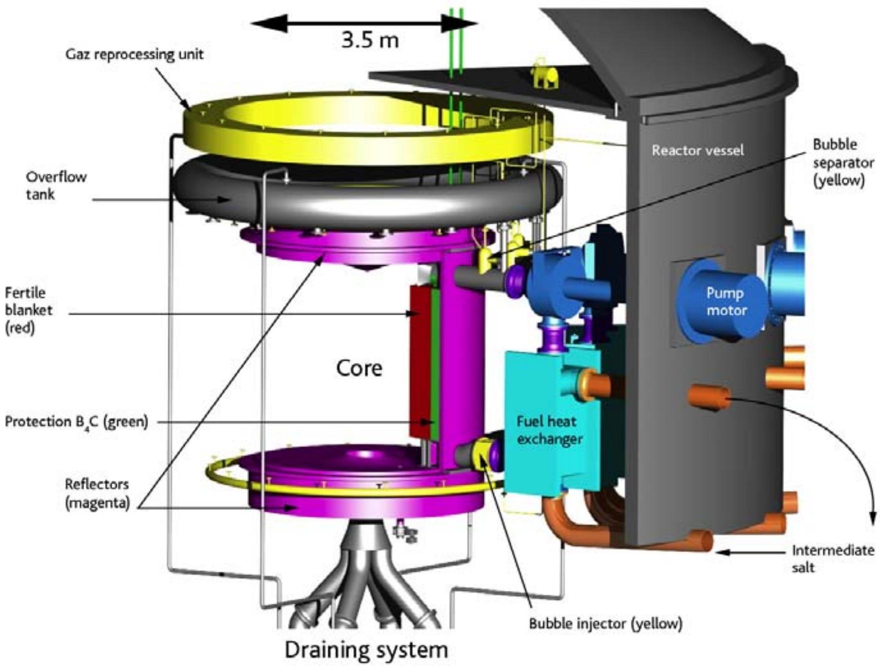

Figure 12
MSFR primary system and core (schematic view) –
Source: CNRS.

The MSFR has a high power density: 330 MW/m^3. The power is produced in the coolant itself with no structural components in the high neutron flux areas.

2/5/2

Safety aspects specific to the MSR

Risks of structural corrosion

The corrosive effects of molten-salt fuels are a major risk that must be considered, especially with regard to the design of the containment barriers. The procedure for managing redox potential used in the MSRE seems to provide an answer to the problem, and was validated by MSRE operations. According to the CNRS, such a procedure could be applied to a power reactor like the MSFR, by measuring the redox potential of the extracted salt and correcting this potential in the injected salt. Clearly, this would need to be validated in a prototype or demonstration reactor.

In any event, structural corrosion by salts has been the subject of R&D work since the 1950s, in particular in Japan and the USA, which are also supporting the development of fusion reactors using molten salt technology. In summary, Reference Document [18] essentially states that understanding corrosion mechanisms still requires significant R&D work to better determine the corrosion resistance of various metals that could be used in an MSR, depending on factors such as the type and composition of the salts, their purity, their oxygen concentration and various possible electrochemical effects. For the MSFR project, CNRS has adopted (nickel-, chromium-, molybdenum- or tungsten-based) Hastelloy® alloys.

Salt toxicity

Risks associated with the chemical toxicity of the salts used should also be examined. FLiBe seems to merit special attention, given the toxicity of beryllium[39]. The salt itself is not toxic but it would be necessary to consider its reduction or decomposition by overheating (above about 1,800°C), leading to the release of beryllium oxide or beryllium metal. This question clearly requires deeper analysis. However, it should be noted that the CNRS has not adopted FLiBe for the MSFR, as they consider that beryllium, as a neutron multiplier, could lead to risks of recriticality in the reprocessing unit.

39 Beryllium is carcinogenic, even in very low doses. It can also cause an allergic reaction called berylliosis.

Chemical risks and risks of thermodynamic interactions with molten salts

According to the sources consulted, none of the salts mentioned in Section 2/5/1 react violently with air or water, including the sodium- and potassium-based salt (FLiNaK). Clearly, this would need to be confirmed.

IRSN has no information regarding the thermodynamic interaction between salts and water. This issue also needs further analysis.

Control of reactivity

MSRs do not have absorber rods or burnable poisons to control the chain reaction. Shutdown is performed by draining the fuel salt. Power is controlled by "demand", *via* acting on the intermediate-system pump. A decrease in flowrate leads to a reduction in reactor power. The feedback coefficients are sufficiently negative to provide this control in a few tens of seconds for a 50% power variation, without a significant variation in temperature (just a few °C). It should also be noted that the void coefficient is strongly negative.

Removal of heat from the core

For the MSFR, it is envisaged that after the fuel salt has been drained into the dump tank(s) located below the reactor, residual heat removal can be provided by passive systems, using natural convection in the tank(s) and surrounding heat pipes. Residual heat removal by active systems under normal conditions is also planned.

Confinement of radioactive materials

MSRs are unusual in terms of containment barriers as there is no cladding to contain the fission products in the fuel. The first containment barrier is provided by all the pipework and components in which the fuel salt circulates (such as the "core chamber", pumps, heat exchangers, upper rings and dump tank(s)).

The second containment barrier is the inerted vessel containing the primary system and the dump tank(s) for the fuel salt loop.

No information is available on the reactor building which should provide the third barrier. According to the CNRS, this should be similar to a PWR or SFR reactor building.

2/5/3

Aspects of the safety analysis

Obviously, the WENRA objectives in their current form cannot be applied to MSRs, in particular due to the fact that the fuel is molten during normal operation and due to the absence of cladding. Furthermore, an appropriate application of defense-in-depth and the safety demonstration would be required for MSRs. An unavoidable initial step, especially important for the safety analysis of an MSR design, would be to determine the radioactive contents (in terms of aspects such as the locations, quantities and spectra of radionuclides) in the system as a whole, including the reactor itself and the reprocessing unit, which would contain fuel salt and fission products. This analysis should cover normal operation and various possible failures (such as leakage), to systematically substantiate aspects such as the architecture and design of the barriers and to identify any risks of criticality.

Nevertheless, some comments on various safety aspects can be made here.

Firstly, it should be stressed that molten-salt technology has been used for several decades in industrial heat-transfer processes (in particular in aluminum manufacture). However, these processes use different salts than those involved here. With regard to nuclear reactors, as stated above, corrosion risks seem to have been well managed on the MSRE using a procedure for controlling the redox potential of the salts, which clearly would need to be validated for larger reactors.

With regard to controlling reactivity, the MSFR has favorable neutronic characteristics, in particular a negative void coefficient. As stated above, "reactor scram" is performed by fuel salt drainage. Thus, drainage is a safety function which must be very highly reliable, like the redundant and diversified shutdown systems on other designs. A passive drainage system using a "freeze plug" is planned for the MSFR. An important aspect is that the fuel salt recovery device is designed to avoid any risk of criticality.

The design should also reduce as much as possible risks of water ingress into this salt recovery device, as the introduction of a moderator could make the fuel salt critical again (addition of a moderator into fast-neutron fuel may make it go critical and lead to a power excursion).

The risk of criticality can be reduced by adopting a design using several tanks installed in separate cells.

The MSFR has low thermal inertia. For example, in the event of loss of intermediate-system cooling, fuel salt drainage should occur within about ten minutes to prevent temperatures being reached which are unacceptable for the structures of the first containment barrier, and within 30 minutes to prevent salt boiling.

In-service inspection is an important aspect, both with regard to structural corrosion risks and to check that thermal shielding remains in place, as this could be a source of loose parts. IRSN has no data on this topic, particularly from the R&D point of view. However, reactor emptying will be a ‘routine’ task and as such should open up prospects in this area.

Furthermore, advantages of MSRs highlighted by designers are continuous (or small batch) fuel reprocessing, the absence of highly-irradiated structures (no cladding or fuel assemblies) and the recycling of actinides (whether minor or otherwise) in the reactor.

Finally, an MSR fleet would minimize the transportation of irradiated fuel due to on-site fuel reprocessing. However, each MSR itself presents a major proliferation risk.

2/5/4

Robustness with regard to the events that occurred at Fukushima

Earthquake

IRSN has no analysis available regarding the seismic resistance of this type of reactor and it is difficult to take a position on this issue. However, the small mass of the combined fuel and coolant compared with SFRsand LFRs should be a favorable aspect for MSRs. The fuel salt is homogenous and totally fills the "core chamber". The overflow tank is located above the reactor and has too small a volume to present a risk of criticality. Waves in the tank may pose mechanical problems. The general design of this overflow tank has not yet been fixed for the MSFR (Figure 12 shows a possible solution with a ring located above the reactor).

Flooding

An important issue to be examined is the risk of water ingress into the dump tank(s) containing the fuel salt in the event of flooding of the facility or in the event of an earthquake leading to leakages from the tank(s).

Site selection and robust ("hardened safety core") design provisions should aim to "practically eliminate" such risks.

Total loss of power supplies

The MSFR seems to be tolerant of total loss of power supplies, in particular due to the fact that the dump tank(s) can be cooled passively. It should also be noted that the "reactor scram" system, based on gravitational drainage of the fuel salt, can be activated passively by a "freeze plug" in the event of excessive salt temperature.

Loss of the heat sink

The MSFR seems to use only the atmosphere as a heat sink under accident conditions. As stated above for SFRs and LFRs, air heat exchangers located high up on the facility could be a sensitive point in the event of an external hazard (such as aircraft crash or explosion).

Severe-accident management

The spread of fuel salt into the environment would have severe consequences. However, the approach to preventing and limiting the consequences of such an event is yet to be specified. It would need to identify possible scenarios (such as leakage from the dump tank(s) containing fuel salt combined with leakage from the third containment barrier).

2/5/5
Conclusion

In summary, the following points should be noted for MSRs:

- Compared with sodium, the salts envisaged for MSRs do not react violently with air or water.
- Among the salts envisaged, those that contain beryllium may have to be excluded due to criticality risks in the reprocessing unit and to prevent any spread of beryllium, which is highly toxic, in the event of reduction or decomposition of the salt.
- The fact that the fuel is dissolved in the salt is a particularity which means that the defense-in-depth approach must be adapted, paying special attention to the characterization of the radioactive contents (in terms of aspects such as the locations, quantities and spectra of radionuclides) in the system considered as a whole, including the reactor itself and the reprocessing unit, which will contain fuel salt and fission products. This analysis should systematically substantiate the architecture and design of the barriers and identify risks of criticality, in particular in the event of flooding.

- It seems possible to use passive systems to stop the chain reaction and remove the residual heat on MSRs.
- MSRs have low thermal inertia, which means that operators only have a limited grace period (of the order of ten minutes) to intervene in the event of failure of the "reactor scram" drainage system.
- There are currently no analyses available regarding conceivable accident scenarios and their consequences for the environment.
- The risks of structural corrosion by the salts are a significant issue that has been the subject of R&D work since the 1950s. With regard to these risks, there is a procedure for controlling the redox potential of the salts, which seems to have been validated during operation of the American MSRE reactor. Clearly, this procedure would need to be validated for a larger reactor. Alloys are available that are compatible with the use of salts at the temperatures targeted for MSRs (700°C): nickel-, chromium-, molybdenum- or tungsten-based alloys.
- IRSN has no information on MSR in-service inspection.

2/6

SuperCritical-Water-cooled Reactors (SCWR)

2/6/1

Presentation of the concept

Supercritical water-cooled reactors are the only design selected by the GIF that uses water as coolant. In this concept, water is maintained under "supercritical" thermodynamic conditions, in practice above 221 bar and 374°C (see Figure 13 below), which means that an efficiency approaching 45% can be envisaged, compared with 33%-35% for pressurized water reactors.

In the reference design, the reactor would operate at a pressure of 250 bar with core-inlet and core-outlet water temperatures of 280°C and 500°C respectively (possible core-outlet temperatures ranging from 500°C to 625°C are mentioned). Figure 13 above summarizes the thermodynamic conditions for water in pressurized water reactors (PWRs), boiling water reactors (BWRs) and SCWRs.

As for boiling water reactors, the turbine is directly fed by the supercritical reactor coolant water. However, a version with an intermediate system has been studied, with a view to avoiding the risk of contamination of the turbine and auxiliary systems. See Reference Document [19].

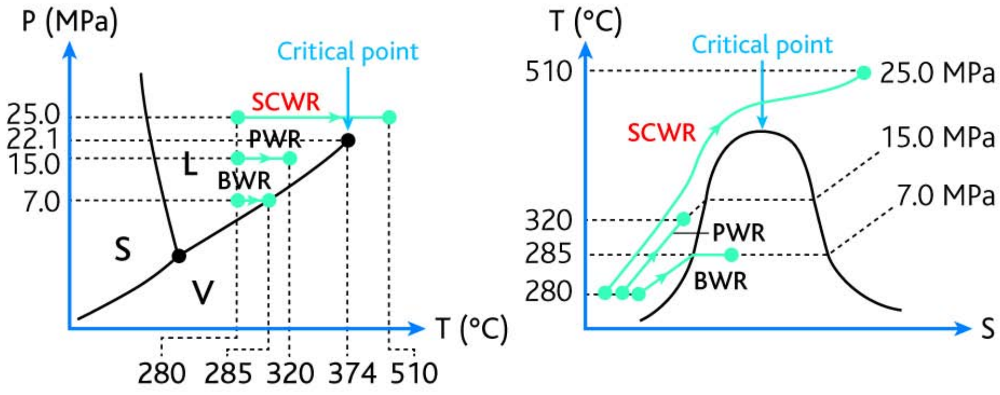

Figure 13
On the left is the phase diagram for water (S for solid, L for liquid, V for vapor) — the critical point and the operating "points" for BWR, PWR and SCWR reactors are shown; on the right is the temperature-entropy diagram for water.

Two design options are envisaged, one using a pressure vessel (see Figure 14) and one using pressure tubes as in CANDU reactors (see Figure 15).

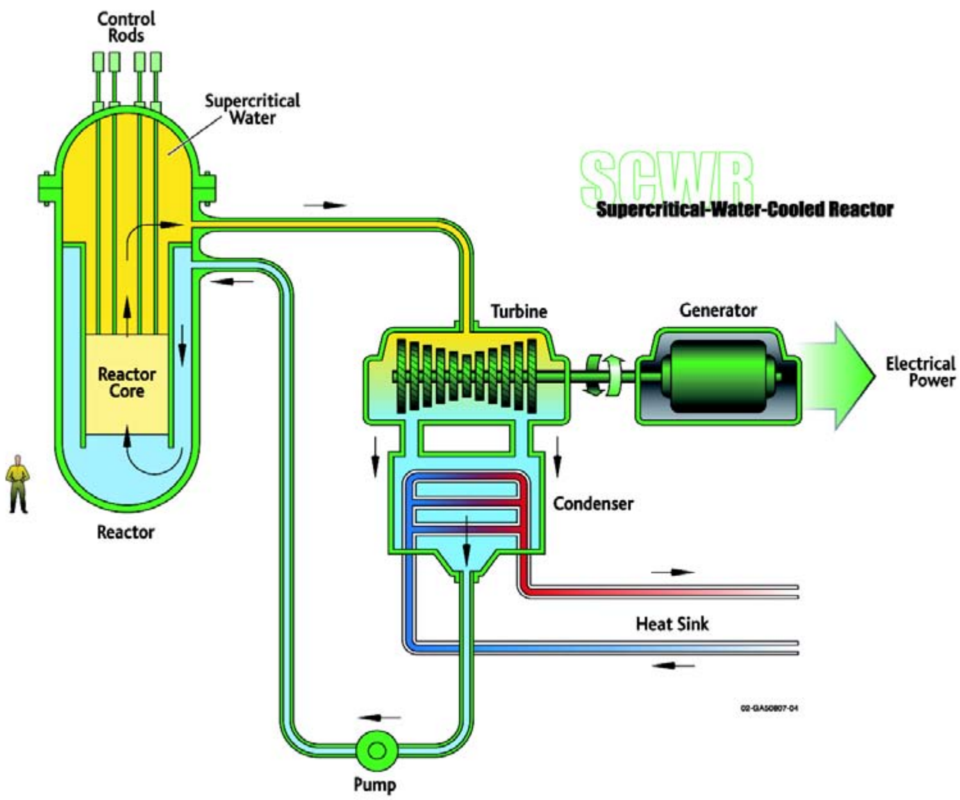

Figure 14
Schematic diagram of a pressure-vessel-type SCWR.

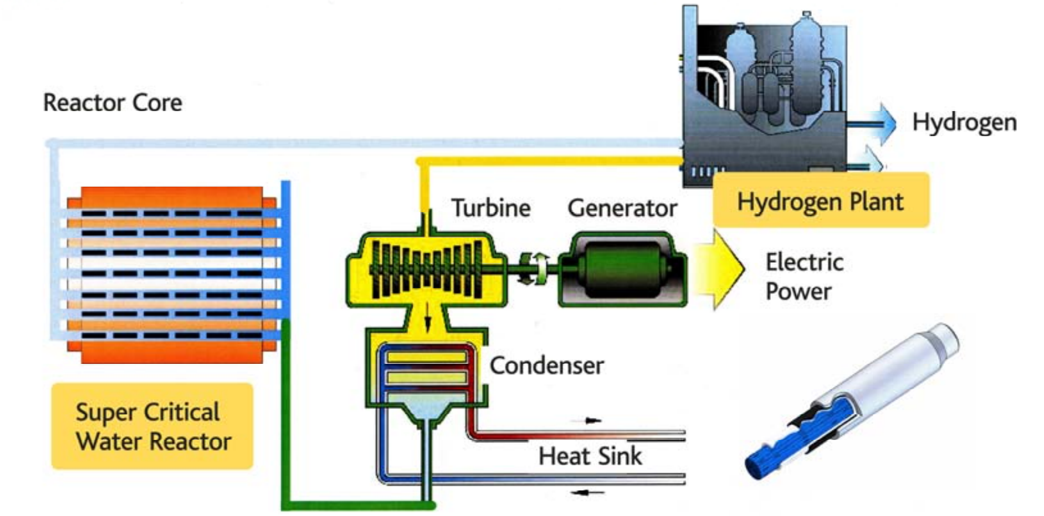

Figure 15
CANDU-style pressure-tube-type SCWR, here paired with a hydrogen production facility; to the right: a pressure tube and its fuel bundle.

Two fuel cycles are also envisaged: an open cycle using UO_2-based fuel in a thermal reactor and a closed cycle using MOX fuel in a fast reactor.

Given the low density of supercritical water, thermal reactors would require moderator elements in the core. For this reason, several projects use "water rods", analogous to control rods, in which the supercritical water flows from top to bottom. The loss of reactivity as the fuel is irradiated can be compensated for by controlling the flow rate in these "water rods", which avoids the need for high reactivity reserve.

For fast SCWRs, preliminary analyses show that a negative void coefficient can be envisaged.

In both cases, the electrical powers cited are between 1,000 and 1,700 MW.

The power density in the core of an SCWR is 100-110 MW/m^3.

With regard to fuel cladding, the zirconium-based materials generally used in light water reactors could not be used with supercritical water. Nickel-based alloys are planned.

2/6/2

Current state of SCWR-concept development and outlook

The SCWR concept is an optimized version of current light water reactors, in particular with regard to economics. The very high heat capacity of supercritical water means that the mass flowrate in the core can be limited (it would be eight times less than in a PWR), reducing the pumping power required. In addition, the use of supercritical water as a coolant avoids the problems associated with liquid-vapor phase changes, such as departure from nucleate boiling (DNB) or dryout, which is one of the limiting factors for PWRs and BWRs. In principle, the SCWR concept, with or without an intermediate system, eliminates the need for pressurizers and steam generators (as needed in PWRs) or recirculation pumps[40], steam separators and driers (as needed in BWRs).

Furthermore, supercritical water has been used in coal-fired power stations for many years, which provides significant experience feedback for the conventional island, along with advances in terms of turbine technology and corrosion-prevention provisions (see Reference Document [23]).

Nevertheless, the use of supercritical water in a nuclear reactor raises many questions, especially concerning materials (cladding materials and materials for reactor structures), which are frequently considered to constitute the major difficulty for this design with regard to the characteristics of supercritical water (chemistry, high temperatures and high pressures). In addition, the very particular behavior of water in the "pseudo-critical[41]" region — with significant variations in its thermodynamic properties depending on the thermal flux in the fuel and its mass flowrate — would require much research, including with regard to the normal transients of reactor start up and shutdown. These considerations mean that four subjects can be highlighted that would require major developments so that the feasibility of an SCWR could be assessed:

- materials;
- the chemistry of supercritical water;
- the thermo-hydraulics of supercritical water combined with neutronics;
- the loss of coolant accident (LOCA) scenario, during which the water would change from the supercritical state to a "normal" state, with separation of the steam and water phases and

[40] In a BWR, the unvaporized water is redirected to the core inlet by recirculation pumps.

[41] On the left-hand diagram in Figure 13, "pseudo-critical" temperature and pressure conditions correspond to the extrapolation of the liquid-vapor curve.

large variations in heat exchange depending on the composition of the mixture.

The main countries involved in developing the SCWR are as follows:

- Canada, *via* Atomic Energy of Canada Limited (AECL) and various universities, which is probably the country most committed to the concept and has developed the CANDU-SCWR design, a pressure-tube-type reactor using a thorium cycle. The reference design would provide a thermal power of 2,550 MW (1,200 MWe), with core inlet and outlet temperatures of 350°C and 625°C respectively and an efficiency of 45%.

- Europe, with the High Performance Light Water Reactor (HPLWR), funded by the EU under the Framework Programmes for Research and Technological Development. The reference design would provide a thermal power of 2,300 MW (1,000 MWe), with core inlet and outlet temperatures of 280°C and 500°C respectively and an efficiency of 43.5%. The main stakeholders are AREVA NP and the Karlsruhe Institute of Technology (KIT). The CEA is listed as a key stakeholder in the program but its contribution is modest.

- Japan, in particular the University of Tokyo, which has been pursuing work on a thermal SCWR (the "Super LWR") since 1989 and on a fast SCWR (the "Super Fast LWR") since 2005.

- South Korea (*via* KAERI), which is developing the SCWR-SM design using a solid moderator (ZrH_2).

- Russia, which is developing various designs, including the SKD with primary and secondary systems, using natural convection in the primary system.

Work has also been performed in the USA with the Nuclear Energy Research Initiative (NERI) project funded by the DOE (see Reference Document [21]). The reference design would provide a thermal power of 3,600 MW (1,600 MWe), with core inlet and outlet temperatures of 280°C and 500°C respectively, 250 bar pressure and an efficiency of 45%.

In addition, it should be noted that the IAEA has launched two Coordinated Research Projects (CRPs), one on the "Benchmarking of Structural Materials Pre-selected for Advanced Nuclear Reactors" (2010-2014) and the other on "Heat Transfer Behavior and Thermo-hydraulics Code Testing for SCWRs" (2008-2012). All stakeholders mentioned above are involved.

However, the increasing number of projects could give a false impression. In fact, the financial and human resources involved are very limited.

2/6/3
Safety aspects specific to the SCWR

As mentioned in the last section, the specific safety issues associated with the SCWR concern the unusual properties of supercritical water.

Heat transfer

Reference Document [22] gives a fairly clear summary of the problems associated with heat transfer when using supercritical water, due to the significant variations in its thermodynamic properties around the so-called pseudo-critical temperature. Around pseudo-critical conditions, a deterioration of heat transfer between the fuel and water can occur under certain specific thermal-flux and mass-flowrate conditions. Much research exists on the subject, which shows that these phenomena appear for high thermal fluxes and low mass flowrates. Thus, it can be assumed that the consequences of an abnormal event leading to an increase in thermal flux or a decrease in mass flowrate could be amplified if the values that trigger the heat transfer deterioration phenomenon were reached. This phenomenon could occur abruptly and lead to an equally sudden rise in cladding temperature. While the extensive research has established a certain number of laws that can be used to predict the appearance of this phenomenon and its consequences, the complexity of reactor-core design means that developments are being pursued on this subject to explore in detail the geometrical configurations and mass flowrate distributions that could be encountered.

Furthermore, even without considering the incident transients mentioned above, the temperature ranges associated with SCWR operation (well below the pseudo-critical temperature at core inlet and well above at core outlet), and sudden changes in the thermodynamic properties in the pseudo-critical region, mean that the reactor’s normal start-up and shutdown phases must be studied in detail. Study of these phases is not easy, given the physical complexity of an SCWR core: several zones with different levels of uranium-235 enrichment, strong coupling between neutronics and thermo-hydraulics, water flowing up the fuel assemblies and down the "water rods" and multiple passes through the core to heat the water from 280°C to 500°C (the EU's HPLWR project has a three-pass core).

The chemistry of supercritical water

Like its thermo-hydraulic properties, the chemical properties of supercritical water are very different to those of the water in PWRs or BWRs. Much research is underway to better understand this chemistry, using the experience acquired in conventional power stations, to determine the parameters to be managed to limit its aggressiveness towards the reactor's structural materials and cladding. Furthermore, it should be noted that the chemistry of supercritical water under irradiation, especially with regard to radiolysis, is currently unknown.

In particular, research is aiming to assess the resistance of various alloys to generalized corrosion, stress corrosion and irradiation-assisted stress corrosion cracking (IASCC) phenomena in supercritical water. Data is very sparse in the literature and there is a major need for experiments. Given the operating conditions (in terms of temperature and irradiation), assessment of alloys must also cover their resistance to heat creep and irradiation creep.

2/6/4
Aspects of the safety analysis

The issues discussed above show that the most important current research is based on the objective of obtaining "robust" normal operating conditions and adequate accident prevention. At this stage, much R&D is still required simply to become convinced that the SCWR concept is well mastered and can be well managed in all phases of normal operation.

The information that is currently available is barely adequate to take a detailed position on the issues associated with preventing accidents and limiting their consequences. Nevertheless, given the analysis performed by the IAEA in particular (see Reference Document [31]), it is reasonable to assume that the SCWR design could benefit from the safety advances obtained from developing third-generation light water reactors, especially regarding the possibility of implementing passive systems. The safety systems outlined in developing the SCWR concept reuse the major principles of BWRs (using systems such as borated water tanks and especially a safety injection system), and core melt is taken into account at the design phase. For example, hydrogen recombiners and a "core catcher" are planned in Reference Document [24].

2/6/5

Robustness with regard to the events that occurred at Fukushima

It can simply be said that the SCWR concept is in a similar position to current light water reactors with regard to sensitivity to the issues raised by the Fukushima accident (earthquake, flooding, loss of power supplies and heat sinks, and severe-accident management). Naturally, design of an SCWR should involve taking into account developments on current light water reactors to improve their robustness with regard to such events.

2/6/6

Conclusion

On paper, the SCWR concept offers very economically attractive characteristics, mainly in terms of being an extrapolation of current light water reactors, in particular BWRs. However, while much research exists on SCWRs (including studies of incident and accident transients), a certain number of issues require specific R&D to enable the industrial viability of the design to be confirmed, *i.e.* the possibility of well-managed normal operations. Currently, the SCWR would seem to be relatively complex and ongoing work is aiming, in particular, to simplify the design. In principle, the SCWR should be able to comply with the major safety principles adopted for third-generation light water reactors, and the lessons drawn from the Fukushima accident.

3/ Overall conclusion

In summary, the overview of fourth-generation reactor concepts, as given in Section 2 above, highlights the following major points.

Sodium-cooled Fast Reactors (SFR)

SFRs benefit from significant industrial experience on the international level (in France, the Great-Britain, the USA, former-Soviet-Union countries, Japan, China and India).

While there is no technological obstacle as such for this reactor type, significant advances are nevertheless required to allow for in-service inspection of safety-related structures (in particular those that support the core), which is made difficult by the fact that sodium is opaque.

Accidents involving fuel damage, or even total core meltdown, have been taken into account in the past in the design of a certain number of SFRs, in particular Phénix and Superphénix, the "RNR 1500" project and the EFR project. For current projects, in particular ASTRID, how core-melt accidents are to be taken into account has not yet been discussed, but some aspects would require further analysis, concerning issues such as the flow of molten materials, the possibility of keeping it within the reactor vessel, the triggering of a sodium "vapor explosion" and its extent, and the transfer of radionuclides from the corium into the primary sodium, the containment and the environment.

In any case, the prevention of accidents with severe core damage, in particular meltdown, should be strengthened, in particular with regard to:

- scenarios involving reactivity insertion in the core, taking into account the possibility of unfavorable neutron feedback (a positive "void effect") depending on the core design, as well as fuel assembly blockage (an accident which can become serious very quickly given the high power density in the core of an SFR); these scenarios should only require limited countermeasures to be taken, with an adequate grace period for their implementation

- the scenario of prolonged total loss of the residual heat removal systems (despite the high thermal inertia provided by the large quantities of sodium in the reactor); provisions should be taken to "practically eliminate" this scenario.

The actual possibilities for removing the residual heat by natural convection in all or some of the sodium loops (as claimed by the designers) will partly depend on the architecture of the systems.

Risks associated with the violent reactions between sodium and air (a sodium fire) or between sodium and water (leading to hydrogen production) remain major safety issues for SFRs.

High or Very High Temperature Reactors (V/HTR)

HTRs benefit from significant experience feedback, with power and experimental reactors having operated in Germany, the Great-Britain and the USA in the past, and currently in Japan and China.

V/HTRs use the most advanced fuel called TRISO: this comprises spherical fissile particles coated with layers of refractory materials to ensure containment up to a temperature of 1,600°C. V/HTR designs plan to use core power densities that are ten times lower than in PWRs and thirty times lower than in SFRs.

There are no technological obstacles for HTRs. However, to reach higher temperatures (for VHTRs), more robust materials must be developed for structures and fuels.

One of the potential risks specific to V/HTRs is a graphite fire, as large quantities of graphite are used in the reactor as moderators and neutron reflectors. This has been the subject of extensive R&D work for several decades, in particular following the major graphite fires that occurred at Windscale in 1957 and Chernobyl in 1986. According to available analyses, a graphite fire could be "practically eliminated" on V/HTRs, but clearly this would need to be confirmed on the basis of aspects such as the detailed design options of a V/HTR and the quality of the graphite used. Furthermore, water ingress into a V/HTR could cause an increase in core reactivity and production of flammable gases *via* the reaction of water vapor with graphite. There is past and ongoing R&D on this subject and design provisions are planned to avoid both risks.

In terms of the safety demonstration, the available analyses produced by the designers mean that, given a low power density in the core (as mentioned above), it is possible to envisage residual heat removal by radiative heat transfer from the reactor core and a very low risk of fuel cladding damage.

Furthermore, fuel meltdown as such can be excluded, which is a favorable aspect of V/HTRs. However, this observation would need to be supplemented by a comprehensive assessment of the risk of reaching core temperatures above 1,600°C, which could lead to extensive damage to the fuel particle coatings, taking into account the risks of total cooling system failure, including the reactor pit cooling system (which may operate passively in some designs).

The dosimetry consequences of deposition in the primary system of carbon dust from the core, which could contain radioactive elements, need to be assessed, in particular with regard to in-service maintenance (including in-service inspection).

Finally, with reference to the events that occurred on the Fukushima- Daiichi nuclear power plant, the risk of air or water ingress into the reactor, in the event of a LOCA or flooding, would need to be covered.

Gas-cooled Fast Reactors (GFR)

GFRs do not benefit from any experience feedback (as no power or experimental GFR has ever been built or operated).

High temperatures (over 800°C at the core outlet) are targeted in order to obtain high power-generation efficiencies. However, there is currently no fuel that is compatible with such conditions, which constitutes a technological obstacle. R&D work is underway.

Compared with SFRs, favorable aspects include the use of an inert gas (helium) for core cooling, a much weaker "void effect", and the general absence of materials that may react violently with air or water.

However, in terms of the safety demonstration, prevention of core melt is a major subject for GFRs. In particular, managing the consequences of a LOCA is a major safety issue for the design, which could lead to complex architecture for protection and safeguard systems, necessarily involving active cooling systems. Furthermore, these systems would have to be highly robust with respect to events such as those that occurred on the Fukushima-Daiichi nuclear power plant. In addition, water ingress into a GFR could lead to a reactivity accident and appropriate prevention would be required.

Lead-or lead/bismuth-cooled Fast Reactors (LFR)

For LFRs, relatively limited experience feedback has been acquired, uniquely for the reactors on Soviet Alfa-class submarines. Three of these reactors suffered serious damage, including one case of partial core meltdown.

LFRs use molten lead or a lead-bismuth eutectic alloy (called LBE) as coolant, which have the advantage of not reacting violently with air or water. There do not seem to be technological obstacles to overcome for LFRs. Management of the risk of corrosion by lead or LBE is nonetheless a key safety issue, but a procedure exists to manage this risk, based on maintaining an appropriate oxygen concentration in the lead and then purifying the lead to remove any oxides formed. This procedure seems to have been validated during operations of the submarine reactors mentioned above, although its suitability for the operation of a large reactor would need to be confirmed. It should also be stressed that the oxygen concentration in lead or LBE would need to be carefully adjusted, as the presence of oxygen (or hydrogen) in lead or LBE increases their viscosity, which can cause overheating in the core.

Special attention would need to be paid to preventing the lead or LBE from freezing on an LFR.

In addition, the risk of embrittlement of metal structures by lead or LBE and their sudden failure is a major issue that has been the subject of research.

In terms of the safety demonstration, the concerns mentioned above for SFRs also seem to apply to LFRs, albeit with the favorable aspect of a lower power density in the core than for an SFR (approximately three times less).

With regard to the events which occurred on the Fukushima-Daiichi nuclear power plant, although lead has the advantage of not reacting with water, it is toxic and robust design provisions would need to be taken to prevent releases of lead into the environment, for example in the event of a break in a loop containing lead or LBE in a flooded room.

Ultimately, the operation of an LFR seems complex, with the need to manage the risk of reactor overheating due to excessive oxygen or hydrogen (increasing the viscosity of lead or LBE) combined with the need to manage corrosion risks by using a technique based on injecting oxygen into the lead or LBE. While this is conceivable for an "overmanaged" operational context or an R&D context

(MYRRHA project), industrial operation could prove more problematic.

Molten Salt Reactors (MSR)

Experience feedback on MSRs is limited to two experimental reactors built and operated in the USA in the 1960s: the Aircraft Reactor Experiment (ARE) and the Molten Salt Reactor Experiment (MSRE), which operated for 13,000 hours.

The CNRS is currently developing the Molten Salt Fast Reactor (MSFR) project, using uranium-233 and thorium-232 dissolved in a lithium-based fluoride salt. At the current state of the project, the MSFR seems to benefit from favorable neutronics characteristics. Reactor shutdown would be provided by fuel salt drainage and passive residual heat removal from the dump tanks is being examined.

There does not seem to be a technological obstacle, even though corrosion by the salt is a significant issue that has been the subject of R&D since the 1950s. There is a procedure for managing this risk, which seems to have been validated during operations on the American MSRE reactor. Clearly, this procedure would need to be validated for larger reactors. Alloys that are compatible with the use of salts at the temperatures targeted for MSRs (700°C) are available.

However, given the originality of the MSR design as described by the MSFR project, associated with the use of a fluid that is both fuel and coolant, there would need to be a revision of defense-in-depth and the safety approaches that have been produced and implemented for current reactors. Special attention would need to be paid to characterization of the radioactive contents (in terms of aspects such as the locations, quantities and spectra of radionuclides) in the system considered as a whole, including the reactor itself and the reprocessing unit, which will contain fuel salt and fission products. This analysis should systematically substantiate the architecture and design of the barriers and identify risks of criticality, in particular in the event of flooding for example. Finally, use of beryllium salts would be discouraged, given the toxicity of beryllium and its characteristics in terms of neutron moderation (and the consequent risk of criticality).

SuperCritical-Water-cooled Reactors (SCWR)

While supercritical water technology has been used in conventional industry, no SCWR has ever been built and operated. The SCWR concept has some very economically attractive characteristics, mainly as it can be considered as an innovative version of light water reactors, in particular BWRs. SCWRs reuse a certain number of the general characteristics of these reactors (such as control of reactivity using absorber rods and emergency borated water, and a safety injection system). However, a certain number of subjects merit specific R&D before a position can be taken on its industrial viability. Furthermore, the engineering associated with this design appears relatively complex and current research is focusing primarily on subjects that deserve development and is also aiming to simplify the design. In principle, the SCWR should also be able integrate the main safety principles that are being developed on third-generation light water reactors, and experience feedback from the accident that occurred on the Fukushima-Daiichi nuclear power plant, in particular the improvements that may be made to BWRs.

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5/
What is the SCWR?

// the next logical step in the LWR path toward simplification

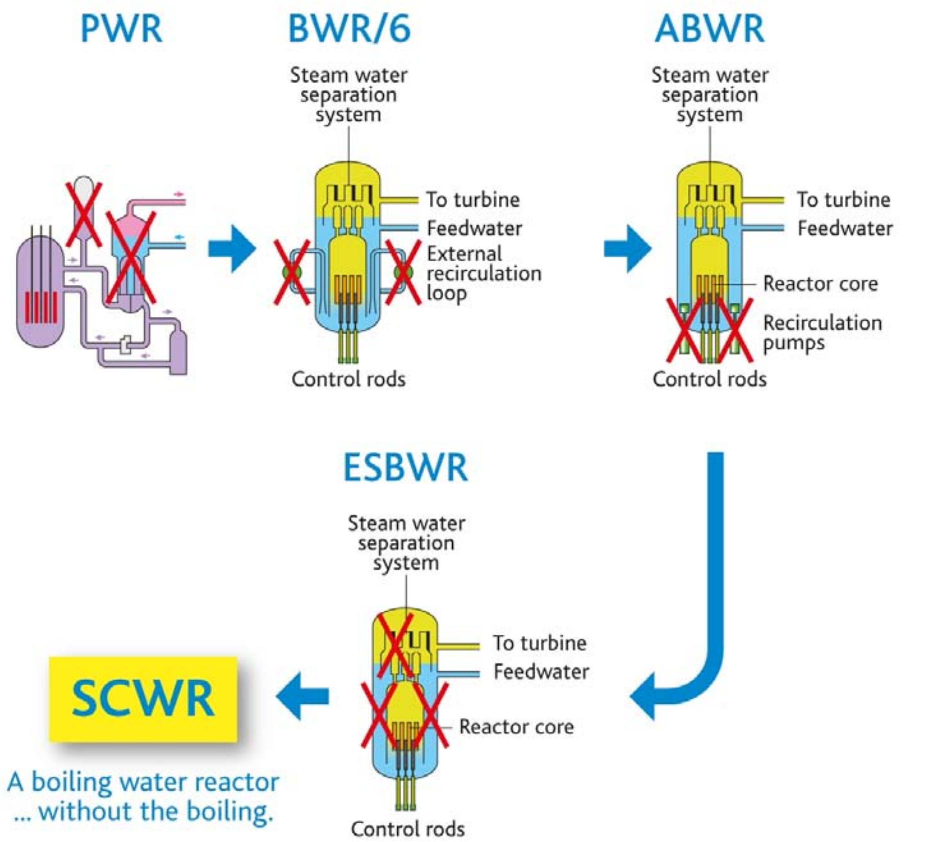

Figure 16.
The SCWR in the context of the various light water reactor designs.

6/ WENRA statement on safety objectives for new nuclear power plants

// should be reviewed no later than 2020

(see Reference Document [2])

Objective O1
Normal operation, abnormal events and prevention of accidents

- Reducing the frequencies of abnormal events by enhancing plant capability to stay within normal operation.
- Reducing the potential for escalation to accident situations by enhancing plant capability to control abnormal events.

Objective O2
Accidents without core melt

- Ensuring that accidents without core melt induce[42] no off-site radiological impact or only minor radiological impact (in particular, no necessity of iodine prophylaxis, sheltering nor evacuation[43]).

[42] In a deterministic and conservative approach with respect to the evaluation of radiological consequences.

[43] However, restriction of food consumption could be needed in some scenarios.

- Reducing, as far as reasonably achievable:
  - the core damage frequency taking into account all types of credible hazards and failures and credible combinations of events;
  - the releases of radioactive material from all sources.
- Providing due consideration to siting and design to reduce the impact of external hazards and malevolent acts.

Objective O3
Accidents with core melt

- Reducing potential radioactive releases to the environment from accidents with core melt[44], also in the long term[45], by following the qualitative criteria below:
  - accidents with core melt which would lead to early[46] or large[47] releases have to be "practically eliminated"[48];
  - for accidents with core melt that have not been practically eliminated, design provisions have to be taken so that only limited protective measures in area and time are needed for the public (no permanent relocation, no need for emergency evacuation outside the immediate vicinity of the plant, limited sheltering, no long term restrictions in food consumption) and that sufficient time is available to implement these measures.

Objective O4
Independence between all levels of defence-in-depth

- Enhancing the effectiveness of the independence between all levels of defence-in-depth, in particular through diversity provisions (in addition to the strengthening of each of these levels separately as addressed in the previous three objectives), to provide as far as reasonably achievable an overall reinforcement of defence-in-depth.

Objective O5
Safety and security interfaces

- Ensuring that safety measures and security measures are designed and implemented in an integrated manner. Synergies between safety and security enhancements should be sought.

44
For new plants, the scope of the safety demonstration has to cover all risks induced by the nuclear fuel, even when stored in the fuel pool. Hence, core melt accidents (severe accidents) have to be considered when the core is in the reactor, but also when the whole core or a large part of the core is unloaded and stored in the fuel pool. It has to be shown that such accident scenarios are either practically eliminated or prevented and mitigated.

45
Long term: considering the time over which the safety functions need to be maintained. It could be months or years, depending on the accident scenario.

46
Early releases: situations that would require off-site emergency measures but with insufficient time to implement them.

47
Large releases: situations that would require protective measures for the public that could not be limited in area or time.

48
In this context, the possibility of certain conditions occurring is considered to have been practically eliminated if it is physically impossible for the conditions to occur or if the conditions can be considered with a high degree of confidence to be extremely unlikely to arise (from IAEA NSG1.10).

Objective O6
Radiological protection and waste management

- Reducing as far as reasonably achievable by design provisions, for all operating states, decommissioning and dismantling activities:
  - individual and collective doses for workers;
  - radioactive discharges to the environment;
  - quantity and activity of radioactive waste.

Objective O7
Leadership and management for safety

- Ensuring effective management for safety from the design stage. This implies that the licenses:
  - establishes effective leaderships and management for safety over the entire new plant project and has sufficient in house technical and financial resources to fulfil its prime responsibility in safety;
  - ensures that all other organizations involved in siting, design, construction, commissioning, operation and decommissioning of news plants demonstrate awareness among the staff of the nuclear safety issues associated with their work and their role in ensuring safety.