REPORT

IRSN
INSTITUT
DE RADIOPROTECTION
ET DE SÛRETÉ NUCLÉAIRE

Faire avancer la sûreté nucléaire

Review of Generation IV Nuclear Energy Systems

REPORT SUMMARY

According to many energy foresight studies carried out in the late 1990s, a shortage of uranium can be expected during the 21st century. This spurred the United States Department of Energy (DOE) to set up the Generation IV International Forum (GIF) in 2000, for the purpose of coordinating research and development work aimed at deploying Generation IV nuclear energy systems (reactors and the related fuel cycle facilities), by the second half of this century.

Member countries, including France, proposed more than a hundred such systems, of which GIF selected six, considered the most promising in light of various criteria based on the following objectives:

- continuation of the progress made by Generation III water reactors in terms of competitiveness and safety;
- more effective use of uranium resources;
- less radioactive waste, especially high-level, long-lived waste;
- greater protection against malicious acts and the diversion or theft of nuclear materials.

The six systems selected by GIF are:

- Sodium-cooled Fast Reactors (SFR);
- Very High Temperature Reactors (VHTR);
- Gas-cooled Fast Reactors (GFR);
- Lead-cooled or Lead-Bismuth Eutectic (LBE) cooled Fast Reactors (LFR);
- Molten Salt Reactors (MSR);
- SuperCritical Water Reactors (SCWR).

GIF selected several fast neutron spectrum systems as they facilitate the transmutation of fertile material into fissile material and, in some configurations, are capable of breeding fissile material. SFR, LFR, GFR and MSR systems are in this category. These characteristics could lead to improved use of energy resources.

Furthermore, high-temperature coolants can be used to improve the energy efficiency of nuclear power plants as well as nuclear facilities, such as the VHTR, that could be used to generate process heat for industry.

French industry has opted for the development of a Generation IV reactor prototype based on the SFR concept by 2020, via ASTRID, the Advanced Sodium Technological Reactor for Industrial Demonstration project. ASN, the French Nuclear Safety Authority, stressed that the choice of one system over another selected by GIF should be based on demonstrated safety and radiation protection performance. In this respect, it considers that the system selected for the development of Generation IV reactors in France should provide significantly more protection for the interests mentioned in Article L593-1 of the French Environmental Code[1] than currently offered by

[1] Public security, health and safety, protection of the nature and the environment

Generation III reactors[2]. ASN therefore wished to obtain the opinion of the Advisory Committee for Reactors (GPR) on the safety and radiation protection characteristics of the six systems studied by GIF.

IRSN would first like to draw attention to the difficulties of performing a "balanced" safety and radiation protection review of the nuclear systems selected by GIF, as some concepts have already been partially tried and tested, while others are still in the early stages of development. Consequently, some technologies benefit from substantial operating experience feedback and in-depth studies, while others are only at the project stage.

The safety of a nuclear facility depends both on the intrinsic characteristics of the facility in question and the design and operating provisions implemented. For that reason, it is impossible to make a comprehensive assessment of the different systems at this point in time and designs currently under consideration may not necessarily be those ultimately selected for Generation IV reactors and the related fuel cycle. IRSN's assessment thus seeks to measure the "safety potential" of the systems in question as far as can be determined on the basis of current knowledge. To this end, IRSN focused on reviewing the intrinsic characteristics of each system and the related design and operating constraints, including aspects specific to fuel cycle facilities. In particular, it considers that the future generation of nuclear facilities would benefit from the development of "tolerant" concepts that are not highly sensitive to events liable to occur inside or outside the facility.

Sodium-cooled Fast Reactors (SFR)

Based on its assessment, IRSN considers the SFR system to be the only one of the various nuclear systems considered to date to have reached a degree of maturity compatible with the construction of a Generation IV reactor prototype during the first half of the 21st century. Furthermore, the scenario in which PWRs would be phased out and replaced by SFRs by the end of the century is plausible given the ready supply of plutonium in the initial stages of the deployment scenario and the use of a closed nuclear fuel cycle with proven oxide fuel. Before this scenario can be rolled out, however, further progress must be made in research and technological development in areas that are, for the most part, already identified. Metal fuel is another option that could be considered, albeit it in the longer term as its use would involve extensive changes to current reprocessing methods and require a significant R&D effort. Investment on this scale would only appear justified if the use of metal fuel brought about a significant improvement in reactor safety.

The main advantage of SFR technology in terms of safety is the use of low-pressure liquid coolant. The normal operating temperature of this coolant is significantly lower than its boiling point (margin of about 300°C), allowing a grace period of several hours during loss-of-cooling events. The advantage gained from the high boiling point of sodium, however, must be weighed against the fact that the structural integrity of the reactor cannot be guaranteed near this temperature. The use of sodium also comes with a number of drawbacks due to its high reactivity not only with water and air, but also with MOX fuel. Design provisions must be made to mitigate the related risks.

While it seems possible for SFR technology to guarantee a safety level at least equivalent to that targeted for EPR-type pressurised-water reactors, IRSN is unable to determine whether it could significantly exceed this level, in view of design differences and the current state of knowledge and research. The nuclear system associated with the SFR has been the focus of considerable R&D, both in France and elsewhere. One example is the ASTRID reactor

[2] In other words, meeting safety objectives on a par with those defined for the Flamanville 3 EPR currently under construction.

should lead to a more accurate assessment of the various technological solutions studied in terms of feasibility and safety.

Very High Temperature Reactors (VHTR)

The VHTR benefits from the operating experience feedback obtained from High Temperature Reactors (HTR). Although it aims at higher technical performance levels than the HTR, it does not offer any progress in terms of safety. Nonetheless, the safety level of the last HTRs developed in the 1980s is already an improvement compared with the other systems selected by GIF. In particular, this technology is intrinsically safe with respect to loss of cooling, which means that it could be used to design a reactor that does not require an active decay heat removal system. The VHTR system could therefore bring about significant safety improvements compared with Generation III reactors, especially regarding core melt prevention. The feasibility of the system, however, has yet to be determined and will chiefly depend on the development of fuels and materials capable of withstanding high temperatures; the currently considered operating temperature of around 1000°C is close to the transformation temperature of materials commonly used in the nuclear industry. A precise assessment of risks relating to graphite dust would also be required if this option was selected.

As it stands, this system does not optimise natural resource and waste management in the long term; storing structural waste and spent fuel without conditioning is not a viable permanent solution. Alternative management solutions are being explored, however, to limit the quantities of graphite for disposal. Although it was contemplated in the past, the use of a closed fuel cycle for the VHTR system is not feasible at present. Demonstrating that VHTR waste can be safely managed and determining the long-term behaviour of waste after disposal (and defining how it should be conditioned) are therefore two key factors in decision-making. The closed cycle option could, however, become more credible in the longer term if thorium fuel (^{233}U-Th) were to be used, although substantial R&D would be required to demonstrate feasibility of the concept. Whatever the case, managing this type of fuel cycle would be a long and complex task. It is clear that VHTR safety performance can only be guaranteed by significantly limiting unit power, which makes it very unlikely that VHTRs will one day replace French nuclear power plants. In view of the above, IRSN sees the VHTR more as a way of consuming plutonium to back up the PWR fleet, for example, and reducing plutonium inventory in the fuel cycle - even significantly so within the context of a nuclear phase-out.

No operating experience feedback from the other four systems studied can be put to direct use. The technological difficulties involved rule out any industrial deployment of these systems within the time frame considered. Nevertheless, a distinction can be made between the LFR and GFR systems on the one hand, for which small reactors could be built during the first half of this century, and the MSR and SCWR systems on the other, where it seems hard to imagine any reactor being built before the end of the century.

Lead-cooled Fast Reactors (LFR)

Lead has some useful neutron properties and, unlike sodium, does not react violently with water or air. The thermal inertia associated with the large volume of lead used and its very high density results in long grace periods in the event of loss of cooling. In addition, the high boiling point at atmospheric pressure is a guarantee of high margins under normal operating conditions and rules out the risk of coolant boiling. This in turn limits the risk of reactivity insertion by void effect that might result from coolant boiling during a loss-of-flow transient in the reactor core without control rod drop. As for the SFR, however, structural collapse would occur at far lower temperatures.

The main drawback of lead-cooled (or LBE-cooled) reactors is that the coolant tends to corrode and erode stainless steel structures. The process currently under consideration to overcome this problem involves creating an iron oxide layer on the surface of these structures. However, mastering this process seems complicated owing to considerable constraints in terms of operating temperature and reactor coolant purification. The reactor operating temperature range is also limited by the risk of the lead freezing. LFR safety is thus highly reliant on operating procedures, which does not seem desirable in a Generation IV reactor. The highly toxic nature of lead and its related products, especially polonium isotope 210 (^{210}Po), produced when lead-bismuth is used, raises the problem of acceptability owing to the potential environmental impact of the facility.

Based on current knowledge and research, IRSN is unable to determine whether the LFR system could guarantee a significantly higher safety level than Generation III reactors. It also draws attention to the fact that various technical hurdles need to be overcome before a reactor of this type could be considered.

Gas-cooled Fast Reactors (GFR)

Given the current state of GFR development, construction of an industrial prototype reactor would not be technically feasible. GFR specifications are highly ambitious, especially regarding the required operating temperatures, and raise a number of technological problems that are still a long way from being solved. However, a low-power experimental reactor could be built, which would be a vital step towards further development of this system. From the safety point of view, the GFR does not display any intrinsic quality likely to lead to a significant improvement over Generation III reactors. The major drawback of this type of reactor is the relatively high power density of the core compared with the low thermal inertia of the coolant. Research is under way to develop a refractory fuel capable of withstanding temperatures in excess of 1600°C in an attempt to overcome this problem, but the feasibility of this type of fuel is far from certain in light of current data in the literature. Short-term core cooling in this type of system therefore requires active devices and would result in shorter grace periods than those obtained with Generation III reactors. The GFR appears less effective in mitigating the impact of severe accidents than the other systems selected by GIF, as the coolant it uses is not capable of retaining radioactive materials. The safety demonstration would rely almost exclusively on the reliability and performance of protection and safeguard systems. In view of the above, IRSN considers that the GFR design, as it is currently perceived, must make much more progress before it can meet the safety levels required for Generation IV reactors.

As with the SFR, several types of fuel could be used in GFR and LFR systems, with carbide and nitride fuels respectively being considered for these two systems in current GIF projects. The feasibility of industrial-scale fabrication and reprocessing of these types of fuel has yet to be demonstrated and several solutions, including pyrochemistry, have yet to leave the laboratory. Furthermore, these fuels are pyrophoric, which means that the

design and safety of fabrication and reprocessing plants would have to be rethought if these systems were selected.

There is no likelihood of even an experimental or prototype MSR or SCWR being built during the first half of this century. The feasibility of these systems has not been demonstrated, nor will it be in the near future, especially for the MSR.

Molten Salt Reactors (MSR)

The MSR differs considerably from the other systems proposed by GIF. The main differences are that the coolant and fuel are mixed in some models and that liquid fuel is used. A particular example of this system is the CNRS-developed Molten Salt Fast Reactor (MSFR), which is the reference model selected by GIF. Its intrinsic neutron properties could be put to good use as, in theory, they should allow highly stable reactor operation. The very low thermal inertia of salt and very high operating temperatures of the system, however, call for the use of fuel salt drainage devices. System safety depends mainly on the reliability and performance of these devices. Furthermore, salt has some drawbacks; it is corrosive and has a relatively high crystallisation temperature. The MSRE reactor built in the United States in the 1950s has provided some operating experience feedback.

The reactor must also be coupled to a salt processing unit and the system safety analysis must take into account the coupling of the two facilities. Lastly, attention must be drawn to the high toxicity of some salts and substances generated by the processes used in the salt processing unit, as the environmental impact of the system could raise problems of acceptability.

The MSR nevertheless has several advantages, including its burning, breeding and actinide-recycling capabilities and the fact that it saves natural resources. The feasibility of fuel salt processing, however, remains to be demonstrated. At present, MSFR deployment scenarios are still being studied. The only scenarios available are simply forecasts aimed at testing the deployment capability of this system under conditions where fissile material resources are in short supply and are therefore not representative of the energy situation in France.

SuperCritical-Water-cooled Reactors (SCWR)

Lastly, the SCWR, a reactor operating in the thermal neutron spectrum, is seen as a further development of existing water reactors and thus benefits from operating experience feedback, especially from boiling water reactors (BWR). Its chief advantage is economic as the operating temperatures considered are consistent with a target efficiency in the region of 45%.

While the use of supercritical water during operation at power avoids problems relating to the phase change from liquid to vapour, such as departure from nucleate boiling or dry-out (which are limiting factors for PWRs and BWRs), it does not present any intrinsic advantage in terms of safety. Thermal inertia is very low, for example, when the reactor is shut down. Furthermore, the use of supercritical water in a nuclear reactor raises many questions, in particular its behaviour under neutron flux (radiolysis). The highly specific behaviour of water in the pseudo-critical region, with thermodynamic properties varying significantly with heat flux in the fuel and with mass flow rate, will also require considerable research. In this respect, close attention must be given to depressurisation accidents, which would result in separation of the water and steam phases and significant heat exchange variations according to the quality of the mixture; much more work is needed to understand the phenomena to be modelled. The SCWR is the only system selected by GIF that uses water as a coolant. At the

current stage of development, it is impossible to ascertain whether the system will eventually become significantly safer than Generation III reactors.

So far, IRSN has not encountered any particular problem regarding fuel fabrication. Nevertheless, constraints relating to fuel reprocessing will need to be taken into account at the fuel element design stage.

Lastly, it would seem more difficult to perform a robust safety demonstration for the SCWR and MSR systems than for the other designs selected by GIF because of close coupling between neutron, thermal-hydraulic and, in the case of the MSR, thermochemical phenomena.

Conclusion

IRSN draws attention to the fact that the review of the Generation IV nuclear systems selected by GIF primarily focused on their intrinsic qualities in order to determine their "safety potential". The safety of these facilities will ultimately depend on the design and operating provisions implemented. Whatever the case, the assessments made will have to be reviewed once the systems have made further progress and new knowledge has been acquired. This is particularly true if the deployment of Generation IV reactors is delayed and postponed until the end of the century. Similarly, current assessments of these systems could be reconsidered due to the emergence of new, more realistic nuclear power scenarios that take into account industrial conditions and make allowance for the decommissioning of present and future reactors. A well-balanced safety and radiation protection assessment of these systems is impossible at present owing to deployment time frames, which can be very different, significant gaps in degrees of maturity, and the fact that the state of knowledge varies considerably according to the system. Consequently, the indications given in the IRSN report should be viewed with caution.

At the present stage of development, IRSN does not have all the necessary data to determine whether the systems under review are likely to offer a significantly improved level of safety compared with Generation III reactors, except perhaps for the VHTR, which is a low-power reactor. For this reason, the VHTR does not seem compatible with the objective - if confirmed - of renewing the existing nuclear power reactor fleet and, in any case, could not be constructed in the short term given the temperatures involved.

Much more research is still required to corroborate this general standpoint. For example, few studies are available on the behaviour of these systems under severe accident conditions.

IRSN also stresses that the Generation IV systems selected by GIF are intended for different national conditions. The selected systems can be associated with different fuel management modes (e.g. open or closed cycles, plutonium breeding or burning) and are therefore not all suited to the energy context in France. Some criteria such as sustainable and optimised management of natural resources and waste, which are particularly associated with fast reactors, are not necessarily compatible with a significant improvement in reactor safety. This is largely because of high operating temperatures and the toxicity and corrosive nature of most coolants considered.

Regarding SFRs, and possibly GFRs and LFRs, IRSN restates its position on research into minor actinide transmutation, namely that this option offers only a very slight advantage in terms of inventory reduction and geological waste repository footprint when set against the induced safety and radiation protection constraints for fuel cycle facilities, reactors and transport. On this point, ASN has recently announced that minor actinide transmutation would not be a deciding factor in the choice of a future reactor system.

Lastly, it should be borne in mind that any industrial deployment of a Generation IV reactor system in France will be linked to its advantages, not only regarding reactor fleet operation and safety, but also in terms of the coherence and performance of the associated fuel cycle. This concerns all aspects relating to safety, radiation protection, material management and efforts made to minimise the quantities of radioactive waste generated, without overlooking the overall economic competitiveness of the nuclear system. Ultimately, the choice of system must be made as part of an integrated approach, based on studies that cover multiple criteria and all the aspects mentioned above.

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List of Abbreviations

ADS	Accelerator-Driven System
ALFRED	Advanced Lead Fast Reactor European Demonstrator
ARE	Aircraft Reactor Experiment
ASN	Autorité de Sûreté Nucléaire (Nuclear Safety Authority, France)
ASTRID	Advanced Sodium Technological Reactor for Industrial Demonstration
BISO	BIstructural iSOtropic
CANDU	CANadian Deuterium Uranium
CCAM	Minor-Actinide-Bearing Blankets
CCI	Corium-Concrete Interaction
CEA	Commissariat à l'énergie atomique et aux énergies alternatives (Alternative Energies and Atomic Energy Commission, France)
Cp	Specific heat
CPPF	Comité de Programme Préparation du Futur
DHR	Decay Heat Removal (emergency cooling system)
DOE	Department of Energy (USA)
dpa	Displacement Per Atom
EC	European Commission
ENEA	Italian national agency for new technologies, energy and sustainable economic development
ESNII	European Sustainable Nuclear Industrial Initiative
FBTR	Fast Breeder Test Reactor
FP	Fission Products
FP (+number)	Research and Development Framework Programme of the European Commission
FR	Fast Reactor
GFR	Gas-cooled Fast Reactor
GIF	Generation IV International Forum
HLW-LL	High-Level Long-Lived Waste
HPLWR	High-Performance Light-Water Reactor
IAEA	International Atomic Energy Agency
IHX	Intermediate Heat Exchangers

INL	Idaho National Laboratory
INPRO	International Project on Innovative Nuclear Reactors and Fuel Cycles
IRSN	Institut de Radioprotection et de Sûreté Nucléaire (French Institute for Radiological Protection and Nuclear Safety)
ISIR	In-Service Inspection and Repair
JAEA	Japan Atomic Energy Agency
LBE	Lead-Bismuth Eutectic
LEADER	Lead-cooled European Advanced DEmonstration Reactor
LFR	Lead-cooled Fast Reactor
LOCA	LOss-of-Coolant Accident
LOFC	LOss of Forced Coolant Flow
LWR	Light-Water Reactor
MA	Minor Actinides
MC	Mixed uranium-plutonium Carbide
MN	Mixed uranium-plutonium Nitride
MOSART	MOlten Salt Actinide Recycler and Transmuter
MSBR	Molten Salt Breeder Reactor
MSFR	Molten Salt Fast Reactor
MSR	Molten Salt Reactor
MSRE	Molten Salt Reactor Experiment
MYRRHA	Multipurpose hYbrid Research Reactor for High-tech Application
NCII	Nuclear Cogeneration Industrial Initiative
NEA	Nuclear Energy Agency
NGNP	Next-Generation Nuclear Plant
NRA	Nuclear Regulation Authority (Japan)
ODS	Oxide Dispersed Strengthened
ORNL	Oak Ridge National Laboratory
PBMR	Pebble Bed Modular Reactor
PFBR	Prototype Fast Breeder Reactor (Indian SFR)
PFR	Prototype Fast Reactor (British SFR)
RHWG	Reactor Harmonisation Working Group WENRA)

SCWR	SuperCritical-Water Reactor
SFR	Sodium-cooled Fast Reactor
SG	Steam Generator
SNETP	Sustainable Nuclear Energy Technology Platform
SWR	Sodium-Water Reaction
TRISO	TRistructural ISOtropic (VHTR fuel)
VHTR	Very High Temperature Reactor
WENRA	Western European Nuclear Regulators Association

DEFINITIONS

The following definitions apply for the purposes of this report:

Accident sequence: Series of events occurring while the facility is in operation and liable to result in the degradation of one or more barriers.

On-site emergency capacity: Period during which safety functions are ensured by on-site resources in the event of an accident, before external resources are required (fuel oil makeup, mobile power sources, etc.).

Generation III reactors: Name given to reactors that provide improved safety over those commissioned up to the 1990s and primarily designed using light- or heavy-water technologies. Particular safety features of Generation III reactors include consideration of severe accidents at the design stage and improved protection against external hazards.

Generation IV reactors: Name given to future reactors that meet a number of objectives: saving uranium resources, gains in competitiveness, improved safety over reactors of the previous generation, reduced waste, and protection against malicious acts and the risks of diversion or theft of nuclear materials. The construction of these reactors is not expected before the second half of the 21st century at the earliest.

Grace period: Period during which a safety function is ensured during an event, without requiring any action by personnel.

Passive systems: Systems that require no AC power source to operate.

Postulated initiating event: Degraded state of facility or consequences of an accident sequence to be avoided (core melt in the case of PWRs, potential hazards, etc.).

Severe accident: In a nuclear reactor, an accident is considered “severe” when the confinement function designed to prevent or mitigate the release of radioactive elements from the nuclear fuel is significantly degraded, whether the fuel is in the reactor, being handled or in a storage area.

1. INTRODUCTION

1.1 BACKGROUND

1.1.1 *INTERNATIONAL SITUATION*

According to many energy foresight studies carried out in the late 1990s, a shortage of uranium can be expected during the 21st century. This spurred the United States Department of Energy (DOE) to set up the Generation IV International Forum (GIF) in 2000, for the purpose of coordinating research and development work aimed at deploying Generation IV nuclear energy systems by the middle of this century. The GIF now brings together 13 countries, including France.

The member countries proposed more than a hundred such systems, of which GIF selected six, considered the most promising in light of various criteria based on the following objectives:

- continuation of the progress made by Generation III water reactors in terms of competitiveness and safety;
- more effective use of uranium resources;
- less radioactive waste, especially high-level, long-lived waste;
- greater protection against malicious acts and the diversion or theft of nuclear materials.

GIF selected several fast neutron spectrum systems as they facilitate the transmutation of fertile material into fissile material and, in some configurations, are capable of breeding fissile material. These characteristics contribute to improved use of energy resources. The use of high-temperature coolants also improves energy efficiency and limits the amount of thermal energy discharged to the atmosphere, both for nuclear power plants and nuclear facilities used to generate process heat for industry.

The systems selected by GIF thus include systems with high energy efficiency, others designed for minor actinide transmutation and one (the VHTR) that can generate process heat for industry. The choice also includes systems that can be associated with different fuel cycles, such as open or closed cycles or symbiotic cycles that combine thermal- and fast-spectrum reactors to handle transition periods. GIF's selection takes into account different national situations and meets different requirements.

The six systems selected by GIF (reference 1) are:

- Sodium-cooled Fast Reactors (SFR);
- Very High Temperature Reactors (VHTR);
- Gas-cooled Fast Reactors (GFR);
- Lead-cooled or Lead-Bismuth Eutectic (LBE) cooled Fast Reactors (LFR);
- Molten Salt Reactors (MSR);
- SuperCritical-Water-cooled Reactors (SCWR).

GIF then prepared a development plan for these technologies (reference 1), published in 2002, which sets out the principles for international cooperation in this area. The plan was recently reviewed and the new edition published in March 2014 (reference 2). Within this context, and taking into account initial experience in operating experimental, demonstrator or prototype reactors (reference 1), prospects for industrial deployment, initially planned for the second half of the 21st century, have changed for some systems and GIF now admits that a further 10 to 15 years of research will be needed to determine their feasibility.

At roughly the same time IAEA, the International Atomic Energy Agency, launched the International Project on Innovative Nuclear Reactors and Fuel Cycles, or INPRO, to promote the development of innovative nuclear systems in order to meet future energy needs, while meeting requirements relating to economic competitiveness, safety, environmental concerns, non-proliferation and acceptability to the public. Unlike GIF, INPRO does not set out to coordinate technical R&D action. Its first aim is to determine national energy requirements and the related objectives regarding the development of nuclear energy, as situations vary from one country to another, and then to specify how innovative systems can help meet them. Another of its ambitions is to define criteria and methodologies for analysing and comparing different innovative reactor concepts.

In Europe, the Sustainable Nuclear Energy Technology Platform (SNETP) was set up in 2009 to drive research, development and demonstration work on present and future light-water reactors, the fuel cycle, three fast-neutron concepts and cogeneration systems (compatible with the use of VHTRs).

A special SNETP team, known as the European Sustainable Nuclear Industrial Initiative, or ESNII, has focused on three types of fast-neutron reactor: the SFR, LFR and GFR (reference 3) with:

- the ASTRID prototype SFR as the reference solution (ASTRID = Advanced Sodium Technological Reactor for Industrial Demonstration);
- two demonstrators as alternative solutions, namely ALLEGRO for the GFR and ALFRED (Advanced Lead Fast Reactor European Demonstrator) for the LFR;
- the MYRRHA demonstrator (Multipurpose hYbrid Research Reactor for High-tech Application) based on the Accelerator-Driven System (ADS) concept, combining a proton accelerator and a nuclear fission reactor in a subcritical state. Although accelerator-driven systems are not among those proposed by GIF, if construction of MYRRHA were to go ahead, useful data could be obtained for LFR systems, as MYRRHA will use a lead-bismuth eutectic (LBE) coolant. It could also be used for fast-neutron irradiation of structural materials and fuel.

Lastly, a number of projects funded by the European Commission under its Research and Development Framework Programmes concern concepts selected by GIF, some of which back up ESNII's activities.

1.1.2 *SITUATION IN FRANCE*

Meanwhile, French commitment to developing Generation IV reactors, was reflected in particular in the objective announced by the French President in January 2006 to commission a Generation IV prototype reactor by 2020, in line with the French Energy Policy Act 2005-781. This objective is closely tied to those defined in French Act 2006-739, dated 28 June 2006, on the sustainable management of radioactive materials and waste, under which the industrial prospects for new generations of reactors, including accelerator-driven systems, were to be assessed in 2012, with a focus on the partitioning and transmutation of long-lived radioactive elements. This legislative request was incorporated in the Decree dated 16 April 2008 which sets out the requirements of the French National Radioactive Materials and Waste Management Plan. Under the terms of this plan, CEA, the French Alternative Energies and Atomic Energy Commission, is responsible for coordinating research into partitioning and transmutation of long-lived radioactive elements.

During its sessions on 20 December 2006 and 20 May 2008, the French Atomic Energy Committee directed French industry towards sodium- or gas-cooled fast reactors, particularly because of the need to save uranium resources and reduce waste (by burning plutonium or producing it from uranium-238 and through transmutation of minor actinides such as americium and curium). In 2009, this work refocused on SFRs, a choice that seems to have been largely based on the maturity of the concept, available know-how and consistency with the strategic national objectives of closed fuel cycles and long-lived-waste management. Within this context, French designers and operators (CEA, AREVA and EDF) set up the joint "RNR-Na" project to define and conduct the necessary R&D for the development of a future SFR system.

CEA is currently heading studies on the ASTRID SFR prototype mentioned earlier, in collaboration with AREVA and EDF[3]. Commissioning is currently scheduled for 2025.

1.2 AIMS OF THE ADVISORY COMMITTEE FOR REACTORS MEETING AND ORGANISATION OF THE REVIEW

1.2.1 *ASN REQUEST*

ASN wishes the Advisory Committee for Reactors to give an opinion on the safety and radiation protection characteristics of the six systems studied by GIF (reference 4) in connection with the development of Generation IV reactors in France. ASN's request can be found in Appendix 1 of this report. In the request ASN stressed that the choice of one system rather than another proposed by GIF should be based on demonstrated safety and radiation protection performance. In this respect, it considered that the system selected for the development of Generation IV reactors in France should provide significantly more protection for the interests mentioned in Article L593-1 of the French Environmental Code[4] than offered by Generation III reactors[5].

[3] The ASTRID project safety orientations were examined by the Advisory Committee for Reactors on 27 June 2013.

[4] Public security, health and safety, protection of the nature and the environment

[5] In other words, that meet the safety objectives defined for the Flamanville 3 EPR currently under construction.

ASN specified that although the focus of the analysis would be on reactors, aspects relating to fuel cycles would also be considered. The agency also announced that the performance of the different systems regarding the transmutation of long-lived radioactive elements should be presented.

ASN's Director-General stated that he wished for the opinion of the Advisory Committee for Reactors on each of the reactor concepts, as might be built around 2050, with particular attention to the following points:

- general characteristics of the different concepts in terms of safety and radiation protection;
- maturity of the concepts and R&D requirements regarding nuclear safety and radiation protection;
- related specific risks;
- main accident sequences;
- details of any operating experience feedback available;
- any particular difficulties or technological obstacles to be overcome before the construction of a prototype or industrial reactor can be considered.

It was also stipulated that there was no need, in principle, for a classification of the concepts.

Regarding aspects relating to fuel cycles, the Director-General wished to have the opinion of the Advisory Committee for Reactors on:

- the various fuel cycle options that could be considered (open cycle, closed cycle, symbiotic cycles, deployment conditions, etc.), particularly those included in the technical-economic scenarios studied by CEA under Act No. 2006-739 of 28 June 2006 on sustainable management of radioactive materials and waste (in particular, these scenarios are associated with the deployment of a fleet of sodium-cooled fast reactors);
- current R&D relating to the processes associated with fuel cycle operations (degree of maturity, technological obstacles, safety and radiation protection issues already determined, etc.);
- management of waste and other materials resulting from fuel cycle operations (inventory/flow, characteristics, storage requirements, impact on fuel cycle facilities, and disposal, etc.).

With regard to transmutation options for long-lived radioactive elements, the Advisory Committee for Reactors will take into consideration the IRSN assessment (No. 2012-00363 dated 3 August 2012) of the various options in terms of safety, radiation protection, material and waste management and the feasibility and advantages of deployment.

Lastly, ASN wishes to obtain the opinion of the Advisory Committee for Reactors on the Generation IV nuclear systems selected by GIF by mid-2014, focusing in particular on greater protection for the interests mentioned in Article L593-1 of the French Environmental Code than offered by EPR-type Generation III reactors.

1.2.2 *ORGANISATION OF THE EXAMINATION AND SUPPORTING DOCUMENTS*

The CPPF has designated the "RNR-Na" Project team as IRSN's contact for the purpose of this examination.

The IRSN review of the six nuclear systems was based on the documentation submitted by the "RNR-Na" Project team, including references 6 to 12.

As the comparative data presented in these documents was lacking in detail and dated back several years, IRSN based its assessment of the advantages and drawbacks of the reactor systems studied by GIF on other documents, in particular references 13 and 14, various IAEA reports and a number of articles published in scientific journals.

Lastly, in order to obtain more accurate technical data, IRSN contacted specialists in three of the six reactor systems under review, namely:

- the Laboratory for Subatomic Physics and Cosmology (LPSC), part of the CNRS National Institute of Nuclear Physics and Particle Physics (IN2P3), which is developing the Molten Salt Fast Reactor (MSFR) project in France;
- the Karlsruhe Institute of Technology (KIT), the German organisation involved in European projects to develop an SCWR;
- and TÜV Rheinland, the German organisation that supported the German Safety Authority in its safety review of HTRs built in Germany.

The details mentioned above were consolidated or completed during technical discussions with the "RNR-Na" Project team organised respectively on 27 November 2013, and 14 and 15 January 2014.

1.3 CONTENTS OF REPORT

The purpose of this report is to provide guidance for the assessment of the Generation IV nuclear systems studied by GIF, based on currently available data.

Chapters 2 to 7 set out the main characteristics of the six reactor concepts associated with the systems selected by GIF. Each chapter gives the history and prospects for development of each type of reactor, together with the theory of operation and general design features, barriers, control of safety functions, special risks inherent in the concept and the main accident sequences studied. This is followed by consideration of the environmental impact, the advantages and drawbacks regarding radiation protection and, for some systems, decommissioning. Lastly, operating experience feedback is described and the main R&D requirements specified. IRSN based its analysis on development projects already under way and incorporated the data gathered in the report to illustrate designs currently under consideration and assess the degree of maturity of the different concepts. The chapters end with a summary of the main advantages and drawbacks in terms of safety and radiation protection. Chapter 8 collates all the above information according to different themes.

Chapter 9 provides further safety information about the fuel cycles that could be considered for the various reactors and the feasibility of certain fuel management options. In this regard, a very brief overview of current R&D on fuel cycle processes has been provided to assess the exact degree of industrial feasibility and detect any difficulties or technical problems that might be raised by, or even call into question, the deployment of certain reactor systems.

Chapter 10 gives the conclusions of the IRSN assessment, mentioned in reference 5, regarding studies of the industrial prospects of SFRs for use in minor actinide partitioning and transmutation and on whether this option could be usefully employed for the management of high-level long-lived waste (use in connection with disposal, impact on fuel cycle facility safety). It also mentions very briefly the actinide transmutation capabilities of the other systems.

Chapter 11 gives a general conclusion.

Note: IRSN stresses the difficulties of performing a "balanced" safety and radiation protection review of the nuclear systems selected by GIF owing to disparities in operating experience feedback, current knowledge and available studies concerning the different systems. This can lead to errors of appraisal.

The safety of a nuclear facility depends both on the intrinsic characteristics of the facility in question and the design and operating provisions implemented. For that reason, it is impossible to make a comprehensive assessment of the different systems at this point in time as designs currently under consideration may not necessarily be those ultimately selected for Generation IV reactors and the related fuel cycle. That is why IRSN's assessment seeks to measure the "safety potential" of the systems in question as far as can be determined on the basis of current knowledge. To this end, IRSN focused on examining the intrinsic characteristics of each system and the related design and operating constraints and incorporating aspects specific to fuel cycle facilities. It considers that the future generation of nuclear facilities could benefit from the development of "tolerant" concepts that are not very sensitive to events liable to occur inside or outside the facility.

2. SODIUM-COOLED FAST REACTORS (SFR)

2.1 HISTORY AND PROSPECTS FOR DEVELOPMENT

2.1.1 *BACKGROUND*

Since the dawn of the era of civilian nuclear power, fast reactors have shown that they enable the transmutation of fertile materials to fissile materials and, in some configurations, are capable of breeding. This possibility was put forward in the light of the scarcity of enriched uranium and the limited resources of natural uranium. Sodium very quickly became the choice of coolant.

Experimental reactors, industrial prototypes and industrial-sized reactors were built and operated in a number of countries. The first nuclear power reactor of this kind was the EBR-I at the Idaho National Laboratory (INL) in the USA (1.4 MWth[6], 0.2 MWe). The EBR-I operated between 1951 and 1963.

A number of SFRs have been put into operation since then:

- France's Rapsodie (25 then 40 MWth, non-power) operated from 1967 to 1982, Phenix reactor (565 MWth, 250 MWe) operated from 1973 to 2009 and the Superphenix reactor (3000 MWth, 1240 MWe) operated from 1985 to 1998.
- In the UK, the Prototype Fast Reactor PFR (650 MWth, 250 MWe), operated from 1974 to 1994.
- Germany's KNK-1 reactor (60 MWth, non-power) was put into service in 1972, retrofitted and renamed KNK-2 in 1977, and permanently shut down in 1992. Construction of the SNR-300 reactor (330 MWe), which began in 1972, was abandoned in 1991.
- Kazakhstan's BN 350 reactor (750 MWth, 250 MWe) operated from 1972 to 1998.
- Russia's BOR-60 (60 MWth, 12 MWe) and BN-600 (1470 MWth, 550 MWe) reactors have been in operation since 1968 and 1980, respectively.
- In the US, the EBR-II reactor (62 MWth, 20 MWe) operated from 1963 to 1993 and the FFTF reactor (400 MWth, non-power) operated from 1980 to 1993.
- In Japan, the Jōyō experimental reactor (140 MWth, non-power) achieved criticality in 1997. Operations at the reactor have been suspended since 2007 after an experimental device located above the core was severely damaged during a handling operation.
- Also in Japan, the Monju prototype reactor (714 MWth, 280 MWe) achieved criticality in 1994. After being forced to shut down in 1995 following a sodium fire, the reactor was restarted in 2010. Shortly after Monju was restarted, an in-vessel transfer machine fell onto the primary circuit. In May 2013, the Nuclear Regulation Authority (NRA) refused to allow the reactor to be restarted following quality assurance failings (many components had not been inspected) on the part of the reactors licensee, JAEA (Japan Atomic Energy Agency).
- In India, the Fast Breeder Test Reactor (FBTR), an experimental SFR (40 MWth, 13 MWe), has been in operation since 1985.

[6] In this report, MWth refers to thermal power and MWe refers to electrical power.

- In China, the China Experimental Fast Reactor (CEFR, 60 MWth, 25 MWe) was connected to the grid in 2011 before being shut down for a safety review. The CEFR is the result of close collaboration between China and various nuclear institutes in Russia. IRSN has no information on a possible resumption of operation of the CEFR.

Altogether, SFRs account for a total of around 400 reactor years of operation.

In June 2013, only three reactors were still in operation (BOR-60, BN-600 and FBTR).

Following the commissioning of the Superphenix reactor, studies were conducted in France in the 1980s regarding a project to build a 1500 MWe SFR known as RNR 1500. The GPR reviewed the preliminary safety analysis report for the project. These studies were then continued in Europe and brought the UK, Germany and France together around a project known as EFR (European Fast Reactor). In 1998, the project was ultimately shelved after it began losing momentum when Germany's electrical utilities pulled out from it in 1993 due to the absence of a licence to operate the SNR-300 reactor.

2.1.2 *ONGOING PROJECTS AND PROSPECTS FOR DEVELOPMENT*

Several projects are under way, with varying degrees of advancement:

- In India, the 500 MWe Prototype Fast Breeder Reactor (PFBR) is expected to achieve its first criticality at the end of 2014. This prototype is expected to be followed by several 1000 MWe SFRs.
- In Russia, construction on the BN-800 reactor (800 MWe) has resumed after being halted by the dissolution of the USSR. The core is being fuelled and the reactor is expected to begin producing electricity in 2014. Russia is also considering developing a commercial version with a power output of 1200 MWe (BN-1200) based on the BN-800 reactor. An SFR research reactor known as MBIR is also in development. The originality of the MBIR lies in the test loops, which are inside the primary circuit and can be cooled by various types of coolant (lead, LBE, gaseous coolant or molten salts).
- In Japan, the preliminary design of the 1500 MWe Japan Sodium cooled Fast Reactor (JSFR) is scheduled for 2015. However, the Fukushima-Daiichi accident has put the project on shaky ground.
- The China Demonstration Fast Reactor (CDFR) project, which has a generating capacity of between 600 and 900 MWe, is under way. It foreshadows the commercial stage of the 1000-1500 MWe Chinese Commercial Fast Reactor (CCFR).
- South Korea is considering building a 390 MWth/150 MWe prototype SFR sometime around 2030 (reference 24).
- In the US, the TerraPower company is developing a 1200 MWth/500 MWe SFR design able to operate for 40 years with the same core (reference 25) while GE Hitachi Nuclear Energy has developed a 310 MWe reactor known as PRISM (Power Reactor Innovative Small Module).
- Lastly, in France, the 600 MWe ASTRID reactor is expected to begin operation in 2025. This reactor is described by its vendors as a prototype of Generation IV SFRs designed to provide experimental qualifications, industrial validations and the maturity and operating experience feedback needed for the growth of the industry.

2.2 MAIN CHARACTERISTICS

2.2.1 *GENERAL DESIGN*

There are two designs of SFR:

- In the pool design, the entire primary sodium coolant is located inside a vessel in which the primary pumps and heat exchangers (intermediate exchangers) are immersed. This design is used for an overwhelming majority of SFR projects under way.
- In the loop design, the primary sodium flows through loops connecting a main vessel (containing, like in PWRs, just the core) to the other main components (primary pumps, intermediate heat exchangers and steam generators). The FBTR in India and the JSFR in Japan are two examples of this type of design.

In pool-type SFRs already in use or in the design stage, the sodium in the primary circuit does not directly exchange its heat with the water in the electricity generation circuit. Instead, it does so with the sodium contained in an intermediate circuit consisting of several independent loops. Each of these loops has at least one intermediate heat exchanger located in the vessel and one or more steam generators (SG) that drive the turbine.

The parts of the primary and intermediate circuits that do not contain sodium are filled with an inert gas that is generally argon (case of the volume or plenum located above the sodium in the main vessel, known as the cover gas space, or the gas space located above the free spaces of the steam generators). The figures on the following pages show schematic diagrams of a pool-type SFR are shown in the figures on the following pages. Figure 1 shows the primary circuit and Figure 2 shows an intermediate circuit that is connected to the primary circuit by two intermediate heat exchangers.

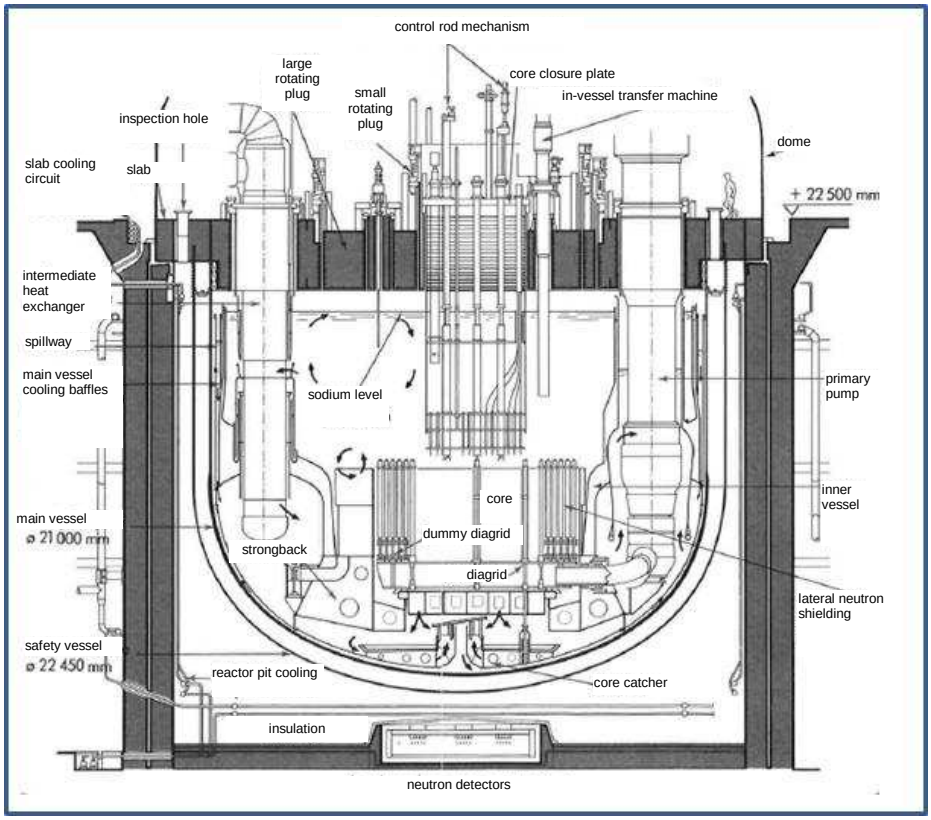

Figure 1: Diagram of a pool-type SFR - primary circuit

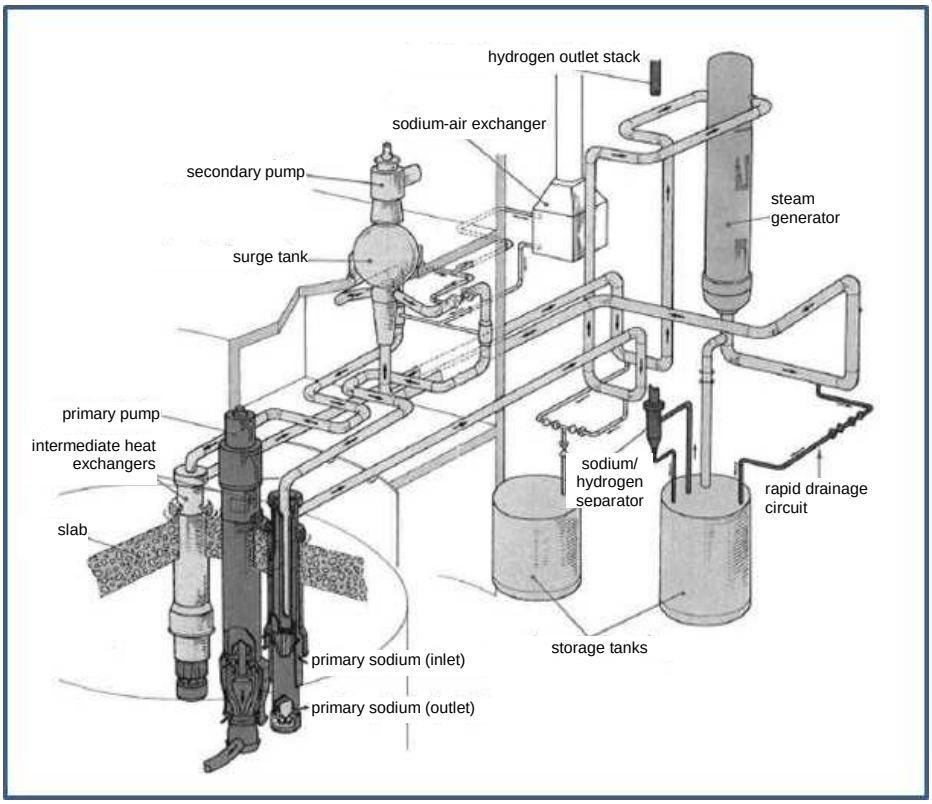

Figure 2: Diagram of a pool-type SFR - intermediate circuit

Figure 3 shows a loop-type SFR.

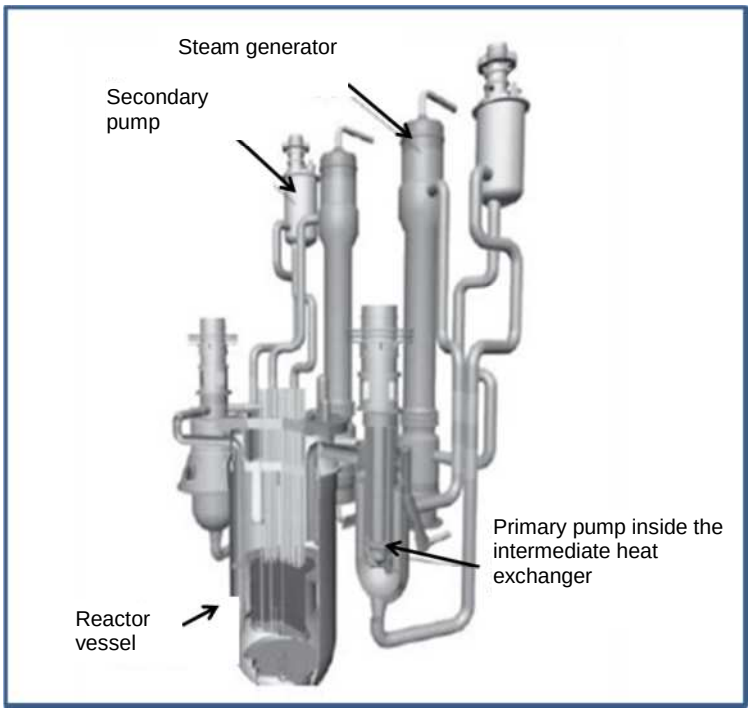

Figure 3: Diagram of a loop-type SFR

In both cases, the reactor is cooled by pumps that are immersed in the sodium coolant. These pumps supply sodium to the diagrid, which is located at the core inlet. The core inlet temperature is imposed by the intermediate sodium circuit and the secondary water circuit. Generally speaking, the temperatures in the primary and intermediate circuits are between 400°C and 550°C.

2.2.2 *GENERAL SAFETY OPTIONS*

SFR technology was initially developed to use natural uranium and for breeding purposes whilst ensuring the viability of the fuel cycle. Accordingly, the cores of fast reactors generally have a volumetric fraction of fuel and a high power density[7]. To ensure such cooling, liquid metal designs use a single-phase primary fluid under little or no pressure.

While liquid metals are highly effective coolants, the cores of liquid metal cooled reactors are sensitive to local cooling (blockages) and power disturbances. SFR safety is also highly influenced by the chemically reactive nature of sodium (including vis-à-vis fuel) and by the increase in potential power that would result from sodium drainage. Lastly, whatever the type of liquid metal used in a reactor, it is very difficult to provide significant amounts of makeup coolant to compensate for leaks in the primary circuit. The aforementioned characteristics have led to the following general safety options:

- Loss of primary coolant is made highly unlikely by the implementation of a dual wall that surrounds the primary circuit and is equipped with a leak detection system.
- The volumes located above the free levels of the sodium circuits are rendered permanently inert and are protected from any ingress of air (positive pressure and inert gas sweeping).
- The assembly cladding temperatures are monitored during operation by a system that reads the assembly outlet temperatures and which quickly detects local blockages.
- The core is equipped with a cladding failure detection system connected to the protection system and with a leak localisation system.
- If energy is converted using the Rankine cycle, the water/steam circuit is isolated from the activated primary sodium circuit by an intermediate sodium circuit.
- The risks associated with potential sodium leaks and fires are addressed by a defence-in-depth strategy that makes use of the detection, isolation and rapid drainage systems (drainage-dedicated circuits).
- The reactor building provides dynamic confinement (ventilation/filtration system) and is generally designed to manage small primary sodium fires.

2.2.3 *FUEL*

The fuel used in SFRs is generally uranium oxide and plutonium (MOX – $(U,Pu)O_2$) with a plutonium content that is substantially greater (15% to 20%) than that of the MOX used in PWRs. This fuel is of relatively small density and slightly moderates the neutron spectrum.

The US has extensive operating experience feedback on the use of metal fuels in SFRs, but less compared to its operating experience feedback on the use of oxide fuels. Metal fuels are very compact, allowing for the design of relatively small reactors. As they do not moderate, the neutron spectrum is harder than with oxide fuels and the Doppler coefficient and efficiency of control rods are reduced. India is operating its FBTR experimental reactor with carbide fuels. The characteristics of these fuels and the operating experience feedback are briefly set out in Appendix III.

[7] In countries with plutonium reserves, SFRs can be used in the medium term to reduce these reserves (see Chapter 9).

2.2.4 *COOLANT*

Liquid sodium is used as a coolant because of its good neutron properties (low moderation effect and low neutron absorbing capability). Sodium rapidly became the coolant of choice following the use, in all of the very first reactors, mercury or a eutectic sodium-potassium alloy (NaK).

In addition to its neutron properties, sodium offers the advantage of having a high boiling point (880°C at 0.1 MPa). This makes it possible to design reactors with:

- a core outlet coolant temperature of around 550°C, the temperature required to aim for a high thermal efficiency of the facility[8] (40% to 45%) and have a significant margin over the sodium boiling point during accident transients. However, this boiling temperature is incompatible with stainless steel;
- a primary circuit that is either unpressurised (pool configuration) or slightly pressurised (loop configuration).

The high thermal conductivity of sodium (around 70 times that of water) also ensures a high transfer coefficient between the cladding and the sodium.

Although sodium is stable when exposed to neutron flux, it produces two radioactive isotopes: sodium-22, which has a half-life of 15 hours, and sodium-24, which has a half-life of 2.6 years. However, the quantities of sodium-24 are significantly less than those of sodium-22, allowing work in the primary circuit after only a few days of shutdown.

2.2.5 *BARRIERS*

There are generally three barriers.

2.2.5.1 First barrier

The first barrier is fuel cladding.

The pressure of the fission gases in the cladding increases throughout the life of the fuel pins because most of the fission gases produced in the fuel are released into special expansion chambers located at the top or bottom ends of the pins.

There are two types of cladding failure: “gas” failures and “open” failures. In the first, only gaseous fission products escape from the cladding. In the second, sodium comes into contact with the fuel.

In the case of a “gas” failure, gaseous fission products migrate to the cover gas space. If many failures occur, the pressure may increase and gaseous fission products may be released by the safety valves on the argon circuit of the cover gas space. Moreover, the occurrence of a large number of “gas” failures while a reactor is operating may, depending on the axial position of the failures, cause an increase in power due to the injection of a large amount of gas in the core (void effect - see Section 2.3.1).

An “open” failure in a pin containing MOX fuel causes the formation of a compound whose thermomechanical characteristics can lead to an increase in the size of the failure and allow fuel to leak into the primary sodium.

[8] In the rest of this report, thermal efficiency shall be understood to mean the ratio between the electrical power delivered by the alternator and the thermal power produced by the nuclear reactions inside the reactor core.

Monitoring of the first barrier

SFRs in France are equipped with systems that detect delayed neutrons released by certain fission products (including some isotopes of bromine). The detection threshold is set to avoid worsening the fault and triggers an automatic reactor trip if it is exceeded. Operating experience feedback has shown that this system, known as DND (Delayed Neutron Detection) detects small "open" failures.

An additional system is then used to detect the assembly on which the failure occurred and remove it from the core. Licensees abstain from operating their facilities with "open" cladding failures (clean reactor concept).

Furthermore, the outlet temperatures of the assemblies are measured in order to estimate the temperature of the hottest cladding. France's Phenix and Superphenix SFRs were equipped with an automatic shutdown system that would trip if the temperature threshold value of the hottest cladding was exceeded.

2.2.5.2 Second barrier

In the case of pool-type SFRs, the second reactor barrier is relatively complex and comprises several items of equipment including:

- the main reactor vessel and the safety vessel. The latter is designed to catch any primary sodium that might leak out of the main vessel;
- the reactor top closure;
- the auxiliary circuits, which carry primary sodium or cover gas (argon) away from the primary circuit;
- the pipes of the intermediate heat exchangers (IHX) separating the primary sodium from the intermediate sodium;
- the pipes of the heat exchangers in the decay heat removal system (DHRS), which are immersed in the primary circuit.

This barrier is not leaktight. Argon seeps through the top closure of the primary circuit and immediate or delayed leaks of argon into the environment occur when the valves used to regulate the cover gas space are opened.
Proven materials are used to ensure that the equipment making up this barrier withstand these conditions. For example, the main vessel is made of low-carbon austenitic steel with a controlled nitrogen content.

Monitoring of the second barrier

Monitoring of the second barrier consists in tracking a number of parameters such as the temperatures of the structures (core support structures, vessels, core cover plug), the sodium level in the primary circuit, the activity measured in the gas filling the inter-vessel gap and the cover gas space, and the detection of sodium in the inter-vessel gap.
In-service inspection of the second barrier is, however, a weak spot in this type of reactor because sodium is "optically opaque" and at high temperature (approx. 200°C when the reactor may be inspected).

It nevertheless should be remembered that it was possible to inspect the main vessel at Superphenix thanks to its low thickness. Such inspections were carried out using an inspection module (known as MIR) that could be placed in the gap between the main vessel and the safety vessel. However, there was no way to inspect the main vessel from the inside. Likewise, the reactor top closure cannot be inspected from inside the tank.

The aim of the design of the ASTRID prototype is to minimise the need for inspections (for example, by eliminating welds in sensitive areas) and facilitate access to areas where failures could have safety consequences. IRSN notes that R&D measures are being taken to apply new periodic inspection technologies that are compatible with sodium. It emphasises that the possibility of conducting in-service inspections of circuits important to safety at a facility is a prerequisite for a Generation IV reactor. Research has been undertaken in India on a main vessel inspection device similar to the MIR used at Superphenix (see reference 33). Sodium-compatible ultrasound systems are also proposed in Japan. The effectiveness of these techniques remains to be demonstrated.

2.2.5.3 Third barrier

The third barrier ensures the confinement of radioactive materials and primary sodium (a toxic substance) during accident situations leading to failure of the first two barriers.

It comprises the reactor building, which may also contain handling and storage facilities for new and spent fuel assemblies. In most cases, the steam generators are located outside the reactor building. However, they may also be installed inside it, as is the case with the JSFR.

The third barrier serves as a protection against external hazards, particularly aircraft crashes.

Monitoring of the third barrier

The pressure inside the reactor building is continuously monitored to ensure that it remains lower than the outdoor pressure.

2.2.6 *FUEL ASSEMBLY HANDLING AND STORAGE*

The handling and storage of new and spent fuel assemblies in SFRs has a number of particular features. The related handling operations, from on-site storage of new assemblies to storage of spent assemblies, are described on the following pages and are based on the design decisions taken for Phenix and Superphenix (see Figure 4 extracted from reference 12) and proposed for the EFR.

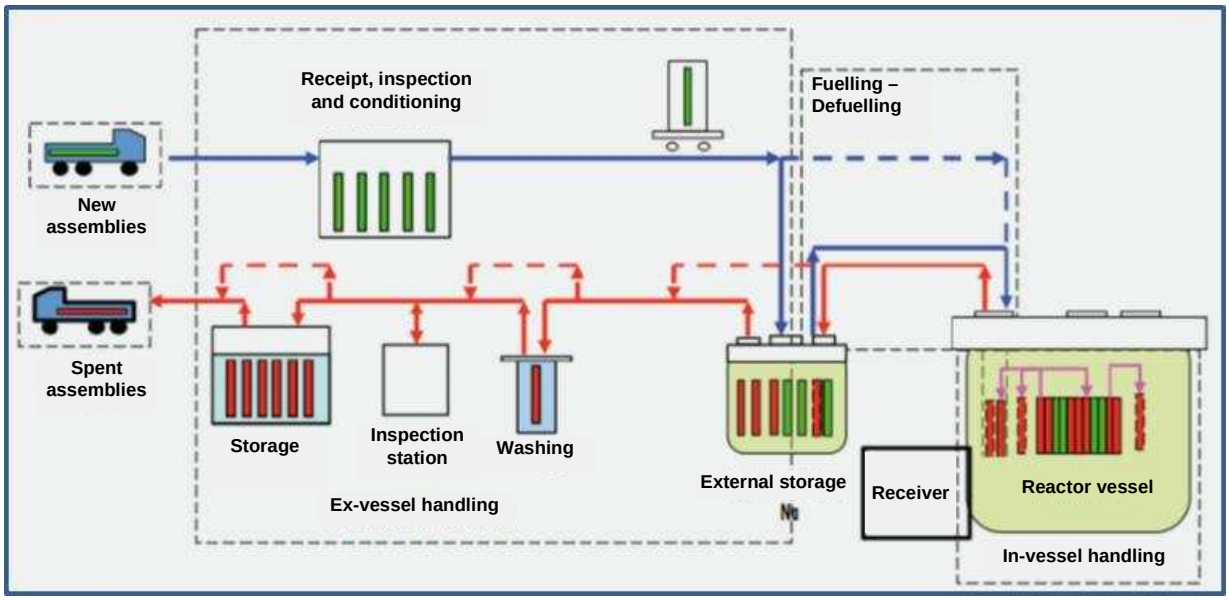

Figure 4: Diagram of handling operations in an SFR

New assemblies are first stored in a dedicated pit in the "receipt, inspection and conditioning" area shown in Figure 4. Due to their low plutonium content, the initial power of these assemblies is non-zero (between 300 and 400 W per assembly for Superphenix). They are cooled by a ventilation system. Because the reactor is shut down, the subsequent operations are carried out "blind", i.e., without opening the various vessels that will house the fuel in turn.

The new assemblies are then placed in an "external storage" area (see Figure 4) consisting of a tank filled with sodium (storage drums at Phenix and Superphenix). The assemblies are then placed inside the reactor vessel via a system of airlocks. There, they are transferred one by one into a sodium pot up to the receiver inside the reactor vessel. The arm of the reactor handling system takes the assemblies from their pot and deposits them in channels inside the core. This operation requires disconnecting the control rods from their mechanisms to allow the rotating plugs to turn. At Phenix and Superphenix, a system known as "Visus" was used to detect any obstacles preventing these plugs from rotating[9]. The EFR project had no external storage area. Instead, there was an internal storage area located along the edges of the reactor vessel.

Spent assemblies are removed to the storage tank in a symmetrical manner.

Their decay heat is reduced during their time in the external or internal storage area. A limited amount of decay heat is required to clean the assemblies of the residual sodium film on their structures (washing area in Figure 4). Proven methods exist for removing this film. Spent assemblies can be transferred to the washing area in a gaseous atmosphere by means of a built-in blower, such as the one used at Superphenix. Washed assemblies can be stored in a water-filled pool (storage area in Figure 4) until they are transferred to the reprocessing unit. The assembly storage tank and its top closure serve as the second barrier.

[9] Japan's Jōyō SFR and India's FBTR SFR are not equipped with such a system. Both have suffered accidents that have led to lengthy shutdowns and complex repairs but without any radiological consequences.

2.3 CONTROL OF SAFETY FUNCTIONS

2.3.1 *REACTIVITY CONTROL*

The reactor is designed so that, under normal operating conditions as well as incident conditions, a slight increase in power or in the temperatures of the sodium at the core inlet or outlet lead to a decrease in reactivity (negative total feedback coefficients – see Appendix II). This is to ensure stable reactor operation. The large amount of sodium provides significant thermal inertia to the primary circuit, particularly in the case of pool-type SFRs, allowing feedback to work.

However, under emergency operating conditions, it is possible that the reactivity temporarily increases either everywhere or in some parts of the core. This is particularly true in the case of sodium draining or the ingress of gas bubbles (void effect).

The void effect is influenced by the size and design of the reactor and varies with the location of gases inside the core. The total void effect may be positive and its value may be all the higher if the core is large (for example, it was close to zero for Phenix and amounted to several dollars for Superphenix and the EFR). Nevertheless, it can be reduced by design provisions such as the insertion of fertile material into fissile material. However, it should be remembered that even if the total void effect is negative, local feedback can be positive and the gain made on the void effect degrade other neutron parameters. Overall optimisation is therefore necessary.

It is worth pointing out that significant efforts are being made by CEA as part of the ASTRID project to develop cores with a low total void effect.

Reactivity can also be increased by core compaction, which occurs when the fuel assemblies are positioned close together. This is because the core is not in its most reactive configuration under nominal conditions. Designers involved in the ASTRID project are looking into the possibility of mitigating this effect in the event of energy loading by designing a core that is relatively compact under nominal operating conditions and which would be associated with core geometry monitoring provisions and, if necessary, compaction prevention systems.

Control rods

The reactivity control system in an SFR consists solely of control rods that, in most cases, contain boron carbide (B_4C).

Automatic reactor trip must be ensured by redundant systems that are diversified to achieve a high level of reliability (for the EFR, a probability of failure of less than 10^{-7} per demand was targeted for total loss of automatic reactor trip).

Innovative rod-drop systems requiring no intervention on the part of the protection system are being explored in order to increase the reliability of the automatic trip systems, for example:

- a system consisting of an absorber mass held over the core by a solid link (metal rod). The link would be broken by differential thermal expansion mechanism if a temperature threshold is exceeded. The absorber mass would then fall by gravity into the core. Such a system can be activated by an increase in the core outlet temperature following loss of the primary pumps combined with failure of the emergency trip systems (CEA's

innovative negative reactivity insertion system known as SEPIA);

- a system that, upon receiving a signal indicating a high sodium temperature at the outlets of the assemblies as well, would cut power to electromagnets holding the control rods, causing them to drop. A system known as SASS (Safe-Actuated Shutdown System) was presented by the JAEA in reference 26.

Conclusion: reactivity control in SFRs requires optimised core parameters and highly reliable automatic shutdown systems. Much research is being done on both of these aspects as part of SFR development projects.

2.3.2 *DECAY HEAT REMOVAL*

Both the high specific heat capacity and large amount of sodium give SFRs a high thermal inertia that slowly increases the sodium temperature in the event of loss of cooling and provides long grace periods for starting up the DHRSs.

However, the temperatures of the sodium and the structures, including those supporting the reactor core and vessels, will rise should the absence of cooling last for more than a few hours. This could ultimately cause the structures to break under the influence of temperature creep in the case of temperatures below the sodium boiling point. The core, most likely damaged by many cladding failures, will no longer be cooled and will collapse into the reactor pit. The radiological consequences of such an accident have yet to be assessed in France.

Various technological solutions are being considered to reinforce provisions for preventing these situations. These solutions entail the use of a combination of systems, such as:

- The normal heat removal circuits (SG water circuits). This solution is used in India by the designers of the PFBR (reference 32).
- Circuits connected to the intermediate loops, such as the sodium-air exchangers (BPR) at Superphenix or at the BN-800 reactor (reference 30).
- Circuits connected to the primary circuit. This solution was implemented for Superphenix (emergency cooling system [RUR]) and had been considered for the EFR as well as the BN-1200 in Russia (reference 29), the PBFR (references 30 and 32) and the JSFR (loop-type reactor) in Japan (see Figure 5 extracted from reference 31 – DRACS and PRACS). Although this solution is widely preferred in SFR projects, particularly ASTRID, it is IRSN's opinion that it has potential common modes of failure of the circuits (sodium use; circuits immersed in the primary circuit, requiring maintaining the sodium inventory in the reactor vessel; stacks making up the heat sink). This matter has been identified by the RNR-Na project and is a major research subject.
- Circuits that remove heat through the main and safety vessels (emergency circuits [US] at Phenix and reactor pit cooling circuits [RUS] at Superphenix).

Diversification can also be achieved by using a coolant other than sodium for one of the DHRSs (such as oil or NaK). This would eliminate the risk of the coolant freezing up in the event of excessive cooling of the circuits using sodium.

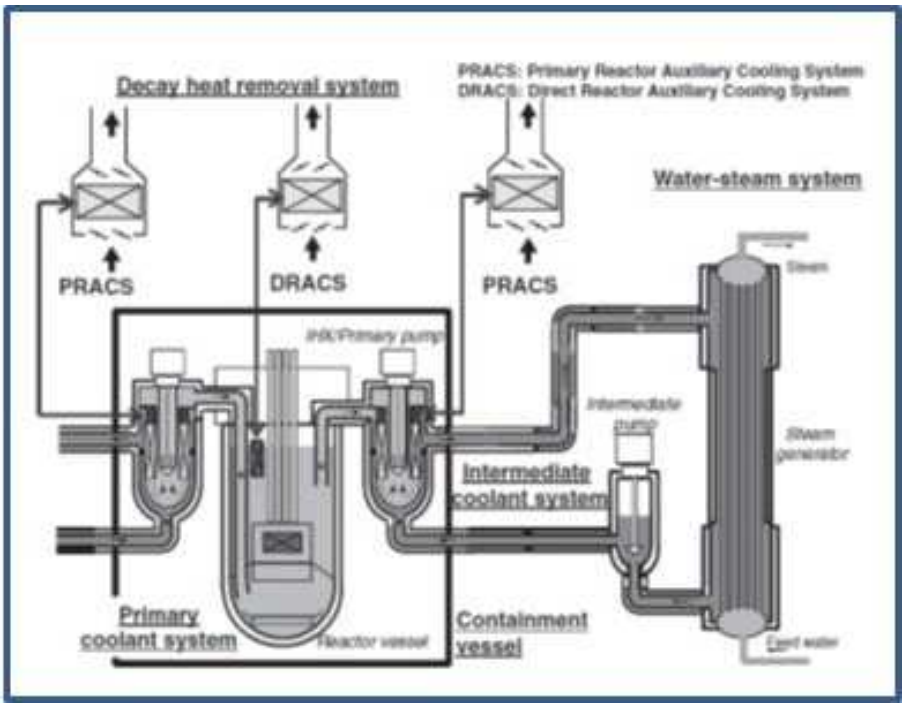

Figure 5: DHRS of the loop-type SFR (JSFR design)

Figure 6, which is of Superphenix, shows the various decay heat removal systems listed above and found in a pool-type reactor.

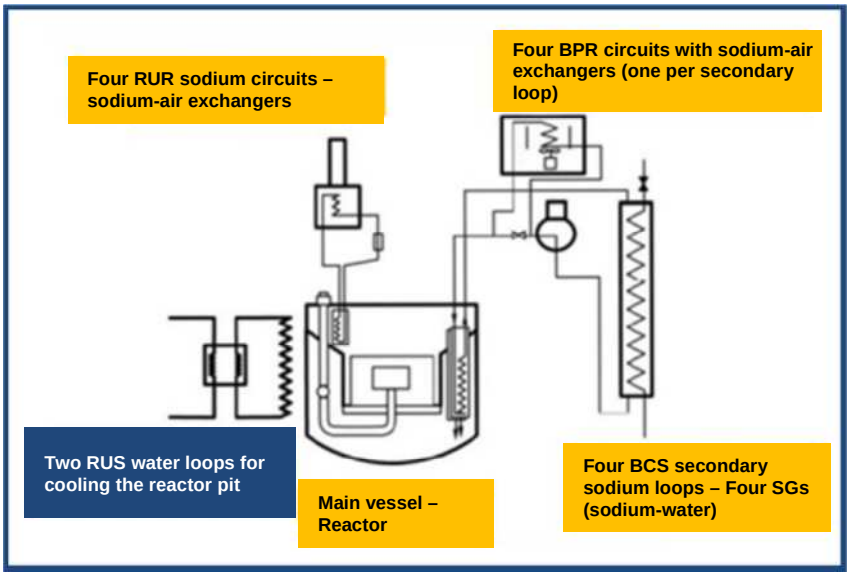

Figure 6: Various technological solutions for the SFR decay heat removal systems illustrated by the case of Superphenix

The high expansion coefficient of sodium and the temperature range where it remains in liquid form also make it possible to establish flows able to cool the core by means of natural convection. This intrinsic feature of the design must also be able to be enhanced by adequate circuit design. Thus, it should also be possible to demonstrate, in the event the normal and emergency power sources are lost, the adequacy of cooling provided by the flow of sodium via natural convection inside the core and the DHRSs. It should be recalled that, for Phenix and Superphenix, the possibility of natural convection was demonstrated by testing some circuits individually but never could be checked comprehensively and simultaneously for all the circuits called upon. In the case of Phenix, a

natural convection test both in the primary circuit and a loop of the intermediate circuit had been planned but could not be carried out.

The high thermal inertia of an SFR makes it possible to significantly limit the kinetics of the increase in coolant temperature in the event of loss of cooling and thus have substantial time (several hours) to implement emergency decay heat removal systems before the temperatures rise to levels that could result in damage to the vessel and the structures. A study of the behaviour of the primary circuit up to the collapse of the core and vessel support structures in the event of sustained total loss of DHR thus seems essential in order to specify the times available for restoring a means of cooling.

Given the consequences of a sustained loss of cooling, several diversified DHRSs must be used to ensure a high degree of reliability of the function. Studies are being conducted as part of various projects to develop systems with few or no common modes. The design must also aim to facilitate cooling of the primary circuit via natural convection. The demonstration of the effectiveness of cooling via natural convection will then have to be established.

2.3.3 *CONFINEMENT OF RADIOACTIVE MATERIALS*

The first two barriers presented in Section 2.2.5 provide static confinement of radioactive materials and primary sodium. Should these barriers fail, ultimate confinement of radioactive materials during accident conditions is provided by the reactor building. A means of dynamic confinement maintains the atmosphere inside the building below the outdoor pressure.

Pool-type SFRs have an advantage in terms of confinement in that the primary sodium and argon in the primary circuit are under low pressure and remain, for the most part, contained inside the second barrier (main vessel). Unlike with PWRs, primary coolant does not flow outside the reactor building during normal or accident conditions. However, there is a risk of the second barrier being bypassed:

- A few auxiliary circuits (such as the purification circuit) may carry a small proportion of primary sodium outside the main vessel.
- The reactor top closure is not completely impervious to argon. The argon inside the reactor building is taken up by the ventilation system of the building.
- The argon in the cover gas space “breathes” in order to maintain a pressure level compatible with the strength of the second barrier. The second barrier is protected from overpressure by valves that, if opened, could vent argon to the stack after a potential passage through decay tanks. A large number of cladding failures can also sweep releases through the argon circuit due to the high amount of fission gases contained in the cladding (see Section 2.2.5.1).

Furthermore, the strategy for the confinement of radioactive materials must take into account the possibility of the release of primary sodium above the reactor slab in the event of an energetic core melt accident.

Lastly, adequate provisions for mitigating the consequences of releases of intermediate sodium must also be implemented.

2.4 RISK ANALYSIS

2.4.1 *RISKS INHERENT TO THE USE OF SODIUM*

2.4.1.1 Sodium-air interaction

Sodium leaks and fires

Like all alkali metals, sodium has a high avidity for oxygen. For example, at temperatures greater than around 120°C, it instantaneously burns in contact with oxygen when the proportion of oxygen in the surrounding medium is between 1% and 2%.

Depending on the degree of atomisation of sodium, its combustion can lead to a rapid increase in pressure (a matter of a few seconds) and an increase in temperature (a matter of a few minutes) in the surrounding area. Rooms containing sodium pipes must therefore be designed to withstand these increases in pressure and temperature, assuming large breaks. A sodium fire can be worsened by contact between the sodium and concrete surfaces. This can lead to exothermic reactions with some components of the concrete and degrade its mechanical strength. Lastly, water released by the concrete can lead to hydrogen combustion or explosion.

The design must therefore aim to prevent the risk of leaks by limiting potential sources of pipe damage and improve early detection of leaks.

Increase in the level of impurities in sodium following ingress of air in the circuits

If air enters the sodium circuits, the amount of hydride and oxide increases, raising the risks of blockage of the assemblies or of rod drop being slowed or even prevented from occurring altogether.

A detection system in which the sodium flow was monitored through an orifice maintained at low temperature ("blockage indicator") had been installed at Phenix and Superphenix in order to detect any abnormal increases in the amount of hydride and oxide in the sodium.

2.4.1.2 Sodium-water reaction (SWR)

Sodium also reacts with water, triggering a series of chemical reactions. The first chemical reaction – the production of sodium hydroxide and hydrogen – is instantaneous, irreversible and exothermic. Although other reactions follow (such as hydration of sodium hydroxide), this first reaction is the one mainly addressed in safety studies. In SFRs, sodium-water reactions primarily occur in steam generator (SG) tubes, where water/steam flows at much higher pressure than sodium (difference of around 18 MPa). In most current designs, water flows inside SG tubes that are generally 2-4 mm thick and a few centimetres in diameter. SWRs induced by water tube leaks is an unstable phenomenon. Initial cracks grow and the neighbouring metal walls (SG tubes or shell) degrade under the action of heat and the reaction products. The neighbouring walls may be pitted by an effect known as "wastage" or by plastic instability (bursting under the effect of temperature creep induced by the very high temperature in the reaction zone). A domino effect can occur, causing the rupture of a large number of SG tubes.

The safety consequences of an SWR in a steam generator are:

- an increase in pressure inside the intermediate circuit following the generation of hydrogen which could damage the tubes of the intermediate heat exchangers (second barrier);

- a risk of the outer shell of the SG rupturing and allowing water and sodium to spill into the building housing the SG and the sodium to react with the air.

In the end, the production of hydrogen in the intermediate circuit and the resulting excess pressure inside the tubes of the intermediate heat exchangers (IHX) create a potential risk of gas being injected into the primary circuit (reactivity effect) and the propagation of a pressure wave (shaking of the core, which also leads to a potential reactivity effect).

A large-scale sodium-water-air reaction in the SG building would create significant excess pressure and possibly a hydrogen explosion that could damage the reactor building (pressure wave, projectiles).

The other areas exposed to SWR risks are:

- the area where spent assemblies and large primary components are washed;
- areas where primary sodium could come into contact with the concrete of the structures;
- the sodium capacities in the event of a sodium leak and induced or concomitant flooding.

When dissolved in water, sodium hydroxide produces caustic soda, which is very corrosive to the structures. This corrosion could affect a loop of the intermediate circuit following the spilling of a large amount of water in the sodium from one or more SG tube ruptures.

The consequences of an SWR first led SFR designers to place an intermediate sodium circuit between the primary sodium having passed through the core and the water in order to avoid the effects of an SWR on the core (pressure effects, ingress of gas bubbles in particular).

In addition, in order to mitigate the risk of SG tube rupture and any consequences of such rupture, efforts in terms of tube design (material and limitation of the number of welds in particular) and rapid detection of water leaks are being made so that a sequence to isolate and lower the pressure in an affected steam generator can be initiated as quickly as possible.

It is worth noting that there are two types of SG design in SFRs already in operation:

- Modular SGs, such as at Phenix, for which each module has a limited number of tubes. This solution mitigates the consequences of an accident and allows for the replacement of just the module in which the SWR occurred.
- “Integral” or “one-piece” SGs, such as at Superphenix, which necessitate in-situ repairs or replacement of affected SGs in the event of an SWR.

A few examples of innovative solutions that mitigate the impact of SWRs in facilities or even eliminate the risk of SWRs in SGs are listed below:

- Modular SGs designed such that the shell of the intermediate circuit could withstand breakage of all the module tubes are planned for ASTRID.
- Another solution consists in eliminating the risk of SWR in SGs by replacing the secondary water circuit by an inert gas circuit.
- In Russia, reverse steam generators (RSG) have been tested in the BOR-60 reactors. In these RSGs, sodium flows inside the tubes while water flows in the space between them. If cracks form, the water will seep into

the tubes but the risk of propagation to the nearby tubes will be significantly limited.

- The Indira Gandhi Centre for Atomic Research (IGCAR) in India has developed an electrochemical meter able to detect SWRs. This process is much simpler than that used to date in French SFRs and offers shorter response times. This meter was successfully tested at Phenix in 2008 and 2009.

2.4.1.3 Reaction between sodium and MOX fuel

When a fuel cladding failure occurs, sodium may come into contact with the solid fuel (once the gaseous fission products have been released into the sodium), allowing the fuel to be released into the primary sodium. This subject has already been addressed in Section 2.2.5.1 on the first barrier.

Likewise, there is a risk during severe accidents of the molten fuel reacting with the sodium. This is addressed in Section 2.4.2.4.

2.4.1.4 Sodium freezing

Undiluted sodium solidifies at 98°C regardless of the pressure. During an extended reactor shutdown, primary sodium can freeze because thermal leaks are no longer offset by decay heat and the heat removed by the primary pumps. Extended shutdowns may take place at operating sites for purposes such as inspection, retrofitting or the replacement of large components. Additional means of heating, such as heat tracing cables, are therefore necessary. The risk associated with sodium freezing is cracking of the structures caused by the contraction or expansion of the sodium as it changes phases (the risk seems greater when sodium thaws because, unlike water, sodium expands when it melts).

Measures must also be taken to ensure that pipes used to drain some components, such as the intermediate loops, are not blocked by slugs of solid sodium. The SWR that occurred in 1987 on an SG at the PFR in the UK was caused by faulty heat tracing cables on a sodium drain pipe. A slug formed, slowing down the drainage of sodium and sustaining the SWR.

2.4.1.5 Sodium-induced embrittlement of steels

R&D has shown that some steels can become embrittled when in contact with liquid sodium at typical SFR operating temperatures. While the austenitic stainless steels used at Phenix and Superphenix (316 and 15-15 Ti steels known for their good ductility) seem to be slightly or not sensitive to embrittlement in the presence of sodium, it appears that the influence of the concentration of non-metal impurities in sodium (oxygen, hydrogen) requires further exploration. However, harder steels, such as T91 (a martensitic steel containing 9% chromium and 1% molybdenum) are being considered for some EFR structures (SG tubes, intermediate circuit and hexagonal fuel assemblies). It should be noted that T91 steels are subject to embrittlement even in contact with undiluted sodium.

2.4.2 *ACCIDENT SEQUENCES*

This section presents the main accident sequences that can occur at an SFR.

2.4.2.1 Inadvertent control rod withdrawal

Inadvertent control rod withdrawal can lead to an increase in power in the assemblies adjacent to the control rod and to fuel melting, particularly during unprotected transients, i.e., without rod drop.

This event has already been the subject of many studies for Superphenix for two reasons: the high linear power density and the internal temperature of fuel during normal operation, and the difficulty in detecting rod withdrawal with neutron measurements and temperature measurements at the outlet of assemblies adjacent to the rod. The RNR-Na project has stated that this difficulty is specific to the design of the reactor at the Creys-Malville plant and that R&D is being conducted to develop more advanced means of detection.

Two solutions are being considered to mitigate the consequences of rod withdrawal: limiting the linear power density and limiting the drop in reactivity during the cycle.

2.4.2.2 Handling errors

Control rods are the only planned means of inserting negative reactivity in SFRs.

At least for SFRs under development in France, the negative reactivity is inserted by control rods is designed to ensure core subcriticality in the event of single or double handling errors (e.g., placement of a fuel assembly instead of an absorber assembly or successive withdrawal of two absorber assemblies). In addition, in-core neutron flux monitoring must make it possible to detect the approach to criticality in the event of errors beyond double handling errors.

2.4.2.3 Local blockages

The high power density released in the fuel assemblies of an SFR and the design of closed assemblies (pin bundle encased in a hexagonal wrapper tube) make these assemblies sensitive to local cooling faults, including those induced by blockages. Cladding failure and even fuel or cladding melt cannot be ruled out in the event of blockage. As stated above, impurities in sodium can promote the formation of blockages.

Blockages are referred to as external or internal depending on whether they occur outside or inside the pin bundle. An external blockage will restrict the flow of sodium through the assembly and raise the temperature of the sodium at the assembly outlet. An internal blockage will restrict heating in and immediately around the blockage (such as in the triangular channel formed by three pins). Regarding internal blockages, it should be noted that assemblies with spacer wires wound around the pins and the presence of an expansion vessel at the bottom part of the pins help to mitigate the risk of particle build-up in the fissile zone.

Detection of blockages

A safeguard based on measurements of the temperature registered by the thermocouples placed at the assembly outlets must make it possible to detect external blockages before unacceptable temperatures are reached in a faulty assembly. The effectiveness of such a system was demonstrated at Superphenix in 1986, when the protection system detected an assembly that was partially blocked at its bottom end. Internal blockages are a very

different case. Depending on the extent of the blockage, there is no certainty that the temperature measured at the assembly outlet can be used to detect local temperature increases even if such increases lead to fuel melt or melting of the cladding around some fuel pins.

Blockages inside pin bundles are potential initiators of generalised core melt in an SFR. Advanced means of mitigating the propagation of local melt to nearby assemblies are necessary.

2.4.2.4 Severe accidents

Loss-of-cooling transients or reactivity insertion transients can lead to core melt or structural collapse (core and vessel support structures).

The phenomenology of reactivity accidents in fast-spectrum reactors is complex because reactivity accidents do not necessarily culminate with power surge-induced fuel dispersion. The relatively high enrichment level of the fuel makes it possible to redistribute materials in critical configurations. This referred to as "secondary excursion", which is in addition to initial reactivity excursion (sometimes referred to as "primary excursion"). This secondary phase poses special challenges to the designers of these reactors, particularly because it is very difficult to reliably predict the geometry of fuel at the end of a primary excursion.

SFR designers are currently attempting to "eliminate" the risk of secondary excursion by designing channels to carry molten fuel out of the core and into a catcher designed to prevent a return to criticality.

Most of the reactors under construction (PFBR in India and BN-800 in Russia) or in the pipeline (BN-1200 in Russia, projects in Korea and Japan, ASTRID) are equipped with a core catcher, which is designed to cool molten fuel and maintain its subcriticality. Specific provisions are designed to prevent the risk of secondary excursion while molten fuel is flowing out of the vessel.

In-vessel core catchers prevent corium from coming into direct contact with the vessel and leading to its rapid collapse. Nevertheless, core catchers pose geometric design problems. For example, the core catcher in the PFBR is designed to collect fuel from only seven out of a total of 181 assemblies. This assumption most probably stems from the factoring-in of a blockage accident on an assembly with limited melt propagation to the six adjacent assemblies. This design thus implicitly amounts to:

- "practically eliminating" loss of residual-heat removal accidents (if only for a short period that does not lead to core melt);
- demonstrating that melt events initiated by assembly blockages will not spread beyond these six adjacent assemblies.

It will also be necessary to demonstrate that corium relocates well into the catcher and that the design ensures that corium will be in a configuration that rules out all risk of criticality. Some Eagle tests conducted by Japanese teams in Kazakhstan showed that molten fuel was entrained towards the top of the vessel. Although such fuel dispersion may be favourable in terms of the risk of return to criticality, the same is not necessarily so in terms of the thermal loading of the vessel. There appears however to be a certain grace period. Restoration, even delayed, of the means of cooling will then prevent vessel melt-through even in the event of relocation of significant masses of corium. Nevertheless, research must continue.

When located outside the main vessel, core catchers can be placed either between the main vessel and the safety vessel or outside the safety vessel. In the event of rupture of both vessels, the latter option would prevent molten core-concrete interaction (MCCI) and, in turn, the production of flammable gases.

In a similar manner as to what could occur in a PWR in the event of high-pressure fuel melt, contact between sodium and molten fuel can cause a thermodynamic interaction with sudden sodium vaporisation. It should be noted, however, that the high thermal conductivity and boiling point values of sodium can cause interactions different from those that occur during a core melt accident in a PWR. Additional research is thus needed to design the reactor building.

In addition to a chemical reaction, contact between sodium and water could also cause a thermodynamic reaction and sudden water vaporisation (steam explosion) accompanied by overpressure effects. Such a situation could occur in the exchangers located between the intermediate loops and the secondary circuit if it is charged with water.

Main release pathways

Core melt accidents are accompanied by a sharp rise in the pressure and temperature in the cover gas space. This rise can:

- blow off the top closure of the primary circuit, allowing primary circuit to flow out and creating a risk of sodium fire;
- transfer fuel fission products and aerosols from the cover gas space to the containment via the leaks through the top closure of the primary circuit;
- transfer fuel fission products and aerosols via the argon circuit in the cover gas space.

The radiological consequences of the last two points are assessed by determining the source term in the cover gas space based on an analysis of the mechanism of transfer of the fission products and actinides from the core.

The release and trapping kinetics of fission products from molten fuel up to the environment is a matter for which R&D measures are necessary.

MCCI causes the containment pressure to rise and ultimately leads to a risk of basemat melt-through in the event of core catcher absence or failure. The effects of such molten streams melting into the ground and contaminating groundwater are particularly detrimental and must be emphasised. The RNR-Na project considers that the risk of molten metal spills is highly unlikely due to the rapid transformation of sodium into oxide, hydroxide and other compounds.

2.5 ENVIRONMENTAL IMPACT, RADIATION PROTECTION AND DECOMMISSIONING

The environmental impact during normal operation is directly related to the source terms available in the facility and depends in particular on:

- the fuel used and its management;
- the materials in the primary circuit and the auxiliary circuits;
- the surface treatments used to mitigate corrosion or erosion products;

- how the circuits are designed to avoid areas allowing the accumulation of corrosion products;
- measures for the confinement of toxic substances;
- the processes used to purify the primary coolant;
- the effluent treatment processes used.

Primary coolant composition

In its original form, sodium contains a few mineral impurities (nickel, potassium, bromine, sulphur, chlorine, etc.) at levels ranging from one to a few hundred mg per kg of sodium. Two other impurities are oxygen and water. These impurities may be carried in by sweeping gases or during handling operations. Hydrogen can also contaminate the sodium by permeation from the secondary circuit via the intermediate heat exchangers.

On top of these impurities is the formation over time of corrosion and erosion products. Selective dissolution of some steel elements (particularly iron, chromium, nickel and manganese) has indeed been observed in liquid metal. These phenomena depend on the primary coolant temperature and the dissolved oxygen content (reference 27). The formation of $Na_xM_yO_z$ compounds was also observed at Superphenix. The behaviour of these impurities in sodium has yet to be studied.

Liquid sodium used to fill the pellet-to-clad gap in absorber control rods also promotes the diffusion of carbon, contained in the boron carbide (B_4C), to the clad. This leads to carburisation of the steel in cladding, embrittling it. This phenomenon affects both the level of impurities in sodium as well as source term composition with, in particular, the formation of carbon-14 (^{14}C).

Lastly, the source term contains tritium, which is formed in fuel (ternary fission) and from activation of the boron in control rods, as well as fission products (FPs) released during the leaching of radionuclides from fuel by sodium or following one or more cladding failures (even if an automatic reactor trip makes it possible to detect them very quickly).

The presence of these various impurities can have repercussions on the contamination of circuits and thus potentially on worker radiation protection, the composition of effluents to be treated before being released to the environment, and the composition of releases during accident situations.

In order to eliminate all these impurities as much as possible, designs must include means for continuously purifying sodium using cold traps. However, the environmental assessment conducted at Phenix in 2009 seems to show that not all radionuclides are trapped. Manganese (^{54}Mn) and cobalt (^{60}Co) make up the greater part of the activity measured in effluent resulting from the washing of spent assemblies (Section 23 of reference 23). Furthermore, this assessment only covers elements that are believed to contribute predominantly to the overall activity. Other short-lived or inactive elements in sodium should be taken into consideration in the impact studies as well as during the pre-decommissioning stages.

Radiation protection

Generally speaking, the doses received by workers during the operation of an SFR are lower by a factor of 10 than those observed in PWRs in operation in France.

The high reactivity of sodium with air and water means that the coolant must be contained to a maximum. This entails carrying out a number of operations with remote handling equipment. Such equipment helps to limit

worker exposure to radiation. The imposed requirements of keeping the primary coolant clean also limit the activity in the primary circuit as well as the decontamination of parts during washing.

The pool-type design contributes to better control of irradiation since the coolant (primary circuit and secondary circuit) provides protection from exposure to γ radiation.

Compared with the pool-type design, the loop-type design makes maintenance easier because it allows for the loops to be drained, thus reducing exposure times during bottom-head inspections.

Both designs thus offer distinct advantages in terms of radiation protection. It may be noted that implementing the option of transmutation of minor actinides would have a negative effect on radiation protection.

Liquid and gaseous waste management

In addition to taking into account sodium, caustic soda and other sodium species for which inhalation is the main exposure pathway, estimations of the consequences of accidents leading to toxic releases (such as fires) must also take into account metal impurities that may also be released and pose potential medium- and long-term health risks, particularly through the ingestion of contaminated foodstuffs.

It is worth noting that no toxicological reference values (TRV) are available for a number of sodium compounds produced from the reaction of sodium with air or water (such as sodium oxide [Na_2O], sodium peroxide [Na_2O_2] or sodium hydride) or with MOX fuel (sodium urano-plutonate, etc.).

Furthermore, recent studies have shown that the capacity of sodium to retain fission products (FPs), which is often highlighted, particularly for volatile FPs (iodine, tellurium, caesium) may be limited (reference 28). As a result, non-negligible amounts of these elements could be released to the atmosphere during accident conditions. Likewise, during these conditions, much less volatile elements could be released mechanically following energetic phenomena (reference 28).

Decommissioning

Unlike with most other designs, operating experience feedback on SFR decommissioning is available both for the three SFRs built in France (Rapsodie, Phenix and Superphenix) and for progress on the decommissioning of SFRs in other countries. This feedback has made it possible to identify difficulties encountered during decommissioning (such as retrofitting of tools for complete core defuelling, difficulties posed by retentions to achieve full vessel draining, presence of sodium aerosol deposits in the slab penetrations, etc.) and to draw lessons to be applied to the design and operation of future SFRs as well as R&D measures. In particular, Areva has concluded that feedback on SFR decommissioning and from studies conducted for decommissioning projects under way shows that the dismantling of SFRs does not pose any technical roadblocks.

After reviewing the available operating experience feedback, and although some operations may be complex, IRSN did not identify any fatal points that could raise doubts about the feasibility of SFR decommissioning. The specific features of SFRs are mainly related to the presence of sodium and its derivatives (NaK), which necessitates implementing processes to treat residual sodium (retentions and films remaining in equipment after sodium draining). Feedback on decommissioning of the Superphenix plant shows good control of the sodium treatment process (TNa facility) and of the residual sodium treatment processes.

2.6 MATURITY OF CONCEPT AND R&D REQUIREMENTS

Maturity of concept depends primarily on design and operating experience feedback as well as on the extent of available R&D and studies.

The reader is reminded that the SFR design benefits from operating experience feedback amounting to more than around 400 reactor years of operation and that it has been the subject of many studies.

2.6.1 *OPERATING EXPERIENCE FEEDBACK*

The lessons learned from feedback on SFR design and operation have been reviewed on several occasions by the Advisory Committee for Nuclear Reactors, first during the committee meetings on the review of the RNR-1500 project, and most recently as part of the review of the ASTRID project safety orientations. In particular, the analysis of the SFR operating experience highlighted areas where innovations to improve the safety of the design were needed. These areas are listed on the following pages.

Fuel elements

The aim of breeding continued as part of the SFR projects made it possible to acquire the knowledge and techniques needed to fabricate, irradiate and reprocess fuel containing a high proportion of plutonium (10-30%). France possesses solid operating experience feedback in the fabrication of MOX pellets. It also has extensive operating experience feedback on their behaviour under irradiation. Around 430,000 pins have been irradiated in French SFRs and in the PFR in the UK (reference 23) and 54,000 pins have been irradiated in Japan (reference 21) with burn-up rates as high as 150,000 MWd/tonne of oxide. India has also irradiated MOX pins in the FBTR at rates of as much as 112,000 MWd/tonne. Cladding materials, such as cold-worked 15-15 Ti steel, which ensure good fuel element behaviour under rapid flux conditions at doses of as much as 100 dpa (displacements per atom) are currently available.

It therefore stands to reason that MOX fuels are the preferred choice of French designers. IRSN considers that metal fuels could have some advantages, particularly during core degradation, and that an analysis of the respective advantages and drawbacks of metal fuels[10] and MOX fuel would be useful for the development of an SFR system. Although they lead to a harder spectrum, metal fuels do not chemically interact with sodium, unlike MOX. Furthermore, metal fuels have a melting temperature that is a hundred degrees or so below the boiling point of sodium. With a suitable design, this could help to promote the removal of molten fuel from areas where sodium may boil. This feature has prompted the United States to choose this fuel for their research on SFRs (references 34, 35 and 36). However, there remains less available operating experience feedback on the use of metal fuel than on the use of oxide fuels. Likewise, the use of metal fuels requires a number of developments for the fuel cycle (see Section 10).

The RNR-Na project considers that metal fuel has major safety drawbacks, particularly a void effect that can be very high in some situations. It announced during its meeting on 14 January 2014 that it had conducted studies on the matter and that these studies concluded that MOX, which benefits from extensive operating experience feedback, is useful.

[10] Metal fuels have been studied primarily in the United States and irradiations of up to 20 at% have been carried out in the EBR-II and FFTF reactors.

IRSN considers that, if the SFR design is selected for industrial deployment throughout France, identifying the respective advantages and drawbacks of MOX fuel and metal fuels in terms of safety, particularly regarding reactivity and behaviour control during severe accidents, would be useful for verifying the appropriateness of the final choice.

Structural materials (ex-cladding)

Several materials have been found to be inadequate during the operation of France's SFR. Two examples are 15D3 steel, which had been used for the spent fuel assembly storage drum at Superphenix that leaked, and 321 steel, which had been used in the intermediate loops at Phenix and is susceptible to relaxation cracking. On the other hand, 316L steel, used for the main vessels at Phenix and Superphenix, exhibited satisfactory behaviour.

Reactivity control

The four negative reactivity shutdowns that occurred at Phenix in 1989 and 1990 remain unexplained. The most likely explanation is that a change in the core geometry, induced by mechanical vibrations, led to the decrease in reactivity at the start of the transient. Likewise, the causes of the decreases in reactivity that occurred at the EBR II (USA) in 1974 and at Rapsodie (France) in 1978 have not been identified. Furthermore, positive reactivity transients occurred at the FBTR in 1994 after assemblies were rearranged during some power ramp-up levels. However, the cause has been identified and corrective measures have been taken.

IRSN considers that, as part of the projects in development, the cores must be instrumented to facilitate the identification of the causes of the phenomena should they ever recur.

Lastly, operating experience feedback has highlighted problems with control rod drop times or rod drop failure due to sodium aerosol deposits in the cold sections of the rod mechanisms. This risk must also be taken into account during the design stage.

Fuel assembly handling

Assemblies were damaged during in-core handling at the FBTR in 1987 and at the Jōyō reactor in 2007. These reactors were not equipped with monitoring systems that could check for the presence of obstacles above the assemblies before the start of handling operations. The incident at Jōyō occurred when instrumentation cables, which were located outside the vessel and were connected to an experimental capsule located inside the reactor (through the sealing slab), were improperly disconnected before the rotating plug was turned. Although the incident had no radiological consequences, the rotating plug was damaged, rendering the reactor unusable (shut down since 2007).

At the time of the incident, measures had already been taken at Phenix and Superphenix to eliminate this risk.

Sodium leaks and fires

In the vast majority of cases, detection provisions have made it possible to quickly detect past sodium leaks and limit the extent of their resulting sodium fires. That said, sodium leaks and fires have severely hampered the availability of some facilities. For example, the leak that occurred at the Monju reactor in 1995 forced the facility to be shut down for 15 years. On a shorter scale, after the storage drum leak at Superphenix in 1987, three months were necessary between the detection of the leak and its confirmation.

IRSN considers that the detection system must unequivocally detect leaks. Leak localisation remains a challenge.

Sodium-water reactions

As stated above, the SG tubes in SFRs are sensitive items of equipment for two reasons: the water and sodium are separated only by a single wall and the tubes cover a large surface area.

One of the most famous SWRs is the one that occurred at the BN-350 reactor in the former USSR. What little information is available shows that although the reaction had been detected by the reading of the hydrogen concentration in the sodium, the SG isolation-decompression sequence was not initiated. The products released by the SWR were dispersed into the sodium, causing the sodium pipe to rupture and a large sodium fire to ignite (the fire was apparently uncontaminated).

The event with the most important lessons regarding SWRs occurred in February 1987 at the PFR in the United Kingdom. Local wear on the tubes of an SG induced by friction with an internal shell, in the absence of detection of small water leaks in the sodium, to the failure of 40 tubes and damage to a hundred others. The system for protecting the affected SG (isolation-decompression) was finally activated by excess pressure in the surge tank on the secondary loop. Once this system was activated, drainage of the sodium was slowed down by a slug of frozen sodium in a pipe that had not been preheated. However, the portion of the second barrier comprising IHX tubes was not affected and no toxic substances were released. This event highlighted the need for early detection of SWRs.

2.6.2 *R&D REQUIREMENTS*

The following SFR research topics, drawn primarily from the summary (reference 37) seem to be of particular importance for safety:

- change in the structure and thermomechanical properties of innovative pins (high burn-up rate, oxide dispersed strengthened [ODS] cladding, etc.) under irradiation;
- study of the behaviour of irradiated fuel pins during power transients, particularly high-temperature fuel creep and the impact of fission gases on fuel behaviour (swelling and ejection of fuel induced by gas pressure);
- study of the behaviour under transient conditions of innovative pins (axial heterogeneous fuel, high burn-up fuel, new cladding materials such as ODS and new (U-Pu)O_2) fuel fabrication methods: determination of the margin to cladding failure and to fuel melt;
- study of the impact of assembly and pin geometry on primary excursion phenomenology and verification of the effectiveness of devices designed to remove molten materials;
- mechanisms of molten fuel relocation, solidification and formation of debris beds in the catcher.
- behaviour of and change in mixed molten pools (UO_2-steel) after the primary accident phase (hydrodynamics, stratification);
- phenomenology and consequences of the thermodynamic interactions: sodium inflows into a molten pool; contact between molten steel and sodium; reaction between sodium and high-temperature fuel, particularly for sodium temperatures near the saturation threshold and which may promote steam explosions (dispersion of fuel beyond the fissile zone or accumulation and associated risk of return to criticality);
- evaluation of source terms and associated toxic and radiological releases: releases from fuel; FP transport and retention, particularly during severe accidents; chemical speciation and partitioning in the primary sodium;

behaviour of sodium aerosols (containing FPs) and physical and chemical changes in the environment, particularly in the case of sodium fires; behaviour of FPs following sodium-concrete interactions (entrainment by the gases released during the interaction: H_20, H_2, CO_2,CO);

- topics related to the high reactivity of sodium (with oxygen and, to a lesser extent, concrete) and induced consequences (hydrogen production/accumulation, kinetics of sodium aerosol production).

2.7 CONCLUSION

With more than 20 reactors built and in operation around the world and combining nearly 400 reactor years of operation, SFRs benefit from extensive design and operating experience feedback.

They have a certain degree of sensitivity to the reactivity control function, as the core is not in its most reactive configuration under nominal conditions. Although they offer good stability under small power oscillations (normal and incident operating conditions), a significant increase in reactivity can occur during accident conditions in all or part of the core, particularly in the event of sodium draining (void effect). Studies of accidents involving the ingress of gas bubbles in some parts of the core should make it possible to determine design tolerance to local void effects given the performance levels of the detection systems used to initiate automatic reactor trips.

The main advantage of SFRs lies in the use of a liquid coolant that is low pressure and whose temperature under normal operating conditions offers a significant margin (300°C) over its boiling point, creating significant grace periods of several hours in the event of loss of cooling.

However, the benefit associated with a high boiling point must be modulated since the integrity of the structures cannot be maintained at these temperatures.

Sodium also has a number of drawbacks that strongly impact facility design:

- Sodium reacts violently with water and air. Efforts are being made to prevent sodium fires and sodium-water reactions and, where appropriate, mitigate their effects throughout facilities.
- Sodium reacts with MOX fuel.
- The combustion of sodium in the air creates toxic aerosols. It should be noted, however, that the effects in the environment are not persistent.

That said, both pool-type and loop-type SFRs benefit from highly favourable operating experience feedback on radiation protection. Worker exposure in pool-type SFRs is lower by a factor of 10 than that observed in PWRs currently in operation in France.

Defining the confinement strategy is an important issue in the safety of Generation IV SFRs, particularly because of the possibilities of primary sodium being expelled outside the main vessel in the event of a severe accident and bypass possibilities.

Lastly, IRSN considers that the state of knowledge and the operating experience feedback obtained on SFR design and operation make it possible to consider building a demonstrator in the short term, i.e., sometime in the first half of the 21st century. This is because proven materials are available both for assembly structures and for vessels and piping. Special efforts will have to be made in terms of demonstrating the safety of reactivity control during accident situations, behaviour during short-term, medium-term and long-term severe accidents, the

assessment of the associated consequences, and provisions for preventing situations such as the collapse of the support structures of the core or the main and safety vessels.

As regards in-service monitoring, more advanced means of monitoring and detection must be developed in many fields (core monitoring, leak detection of sodium leaks and sodium-water reactions, etc.) in order to prevent accidents. Likewise, developments must be made in the instrumentation required to monitor core behaviour during accidents and corium changes during serious accidents. IRSN considers that in-service inspection of these structures is a weakness of the design.

Given the differences in design and the state of the relevant studies and research, IRSN cannot issue an opinion on the possibility of SFRs reaching a safety level significantly higher than that aimed by the Generation III reactors under construction. The safety analysis of the ASTRID prototype and the results of R&D conducted in France and abroad should make it possible to better understand the safety level that could be reached for these reactors. The review of the safety options dossier for this reactor should make it possible to more accurately assess the feasibility and safety of the various technological solutions under consideration.

3. VERY HIGH TEMPERATURE REACTORS (VHTR)

3.1 HISTORY AND PROSPECTS FOR DEVELOPMENT

3.1.1 *BACKGROUND*

The high-temperature gas-cooled graphite-moderated reactor is one of the very first projects on the use of nuclear fission for civilian purposes. As early as 1947, a project for a helium-cooled reactor fuelled with uranium dispersed in beryllium oxide or graphite pebbles was designed in the USA (reference 38).

In the early 1950s, the United Kingdom developed the Magnox[11] power reactor, which used natural uranium metal cooled by carbon dioxide. The first of the four Magnox reactors built at Calder Hall, in the UK, was connected to the grid in August 1956 (60 MW electrical power). A noteworthy feature of the Magnox reactors is that they were naturally safe during loss-of-coolant conditions, even if unprotected, provided the CO_2 inventory was maintained.

The British Dragon reactor was the first helium-cooled high-temperature reactor to reach initial criticality (1966). The Peach Bottom HTR, developed by General Atomics, reached criticality soon after, in May 1967. Because it used helium, an inert gas, Peach Bottom allowed temperatures at least twice those possible with the Magnox reactors. During the same period, Germany used the technology invented in the USA in the 1940s to develop a design consisting of a helium-cooled bed of graphite pebbles. This culminated in the AVR experimental reactor, which was connected to the grid in December 1967.

France developed the UNGG, a graphite-moderated reactor cooled by carbon dioxide and fuelled with natural uranium. The first UNGG was built at Chinon and was connected to the grid in 1963.

The success of the first helium-cooled HTRs prompted the designers to look into industrial models of higher capacities. As a result, the first three HTRs were followed by the construction of two prototype plants: the Fort Saint Vrain Generating Station, in Colorado (USA), and the THTR-300 (Thorium-Hoch-Temperatur-Reaktor), in the German state of North Rhine Westphalia. The Fort Saint Vrain reactor initially had a capacity of 330 MW electrical power. Its core consisted of fuel in the form of spherical particles coated and contained within graphite rods (compacts) that were in turn embedded in prismatic graphite blocks. The THTR-300 had a capacity of 300 MW electrical power and a core consisting of pebbles (see Figure 7). The shift in output from 15 MW (AVR) to 300 MW was achieved by increasing the core diameter and the power density.

The Fort Saint Vrain reactor was connected to the grid in 1976. After 13 years of operation, it was closed in 1989 due to the prohibitive cost of progressively upgrading the prototype and correcting its design deficiencies. As for the THTR-300, after a 14-year construction period (1971-1985) marked by the tightening of safety requirements following the Three Mile Island accident, it operated for just four years. The THTR-300 also suffered from various technical problems that significantly shortened its service life.

[11] Magnox is short for **Mag**nesium **n**on-**ox**idising, which describes the magnesium alloy used to clad the uranium metal fuel.

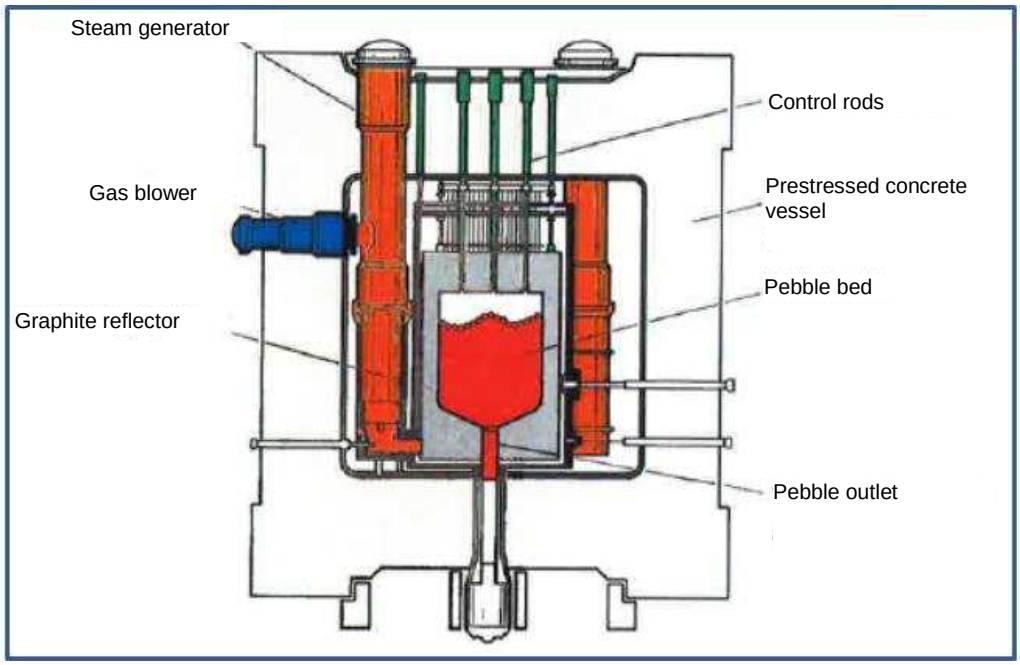

Figure 7: Diagram of the THTR-300

During the 1980s, the German government funded a number of studies to design HTRs even larger than the THTR-300 and with capacities of as much as 3000 MWth. However, this increase in capacity meant the loss of the passive safety features demonstrated with the AVR. In 1984, Siemens/KWU proposed a 200 MWth reactor with a steel vessel through which decay heat could be removed. This reactor would be capable of withstanding reactivity insertion accidents or water ingress without damage. Certification of the reactor, called HTR-MODUL, lasted from April 1987 to August 1989. However, it was subsequently abandoned for economic and political reasons. Although the HTR-MODUL reactor was designed to generate either electricity or heat for industrial processes (cogeneration), the designers' efforts to penetrate the market failed.

In 1993, after Germany and the USA abandoned the HTR concept, South Africa used the advances made in Germany to undertake the development the Pebble Bed Modular Reactor (PBMR), which had advanced features (higher temperatures than the HTR, direct cycle, etc.). Similar to the VHTR concept, the PBMR underwent several major changes during its development. It was ultimately abandoned in 2011, after the South African government withdrew its funding and for want of industrial clients.

In the 1990s the HTR design was seen as a way to reduce Russia's inventory of weapons-grade plutonium. It was thus that, in 1993, the Russian-American Gas Turbine Modular Helium Reactor (GT-MHR)[12] (600 MWth; "direct" energy conversion cycle) was launched. Although the project has not officially been shelved, since 2000 Russia has based its strategy to reduce its surplus weapons-grade plutonium on the fabrication of MOX fuel for use in PWRs and SFRs.

More recently, Japan has considered designing an HTR able to supply heat to thermochemical processes used to produce hydrogen. As part of this project, JAEA built the HTTR experimental reactor at Ōarai, in the Prefecture of Ibaraki. This 30 MWth reactor is designed to achieve a core outlet temperature of 950°C. It went critical in 1998.

[12] MinAtom (Russia), DOE (USA) and General Atomics and Framatome are involved in the project.

The most recent HTR is the HTR-10 (10 MWth), which was built at Tsinghua University, near Beijing (Figure 8). It went critical in 2000 and is currently in operation.

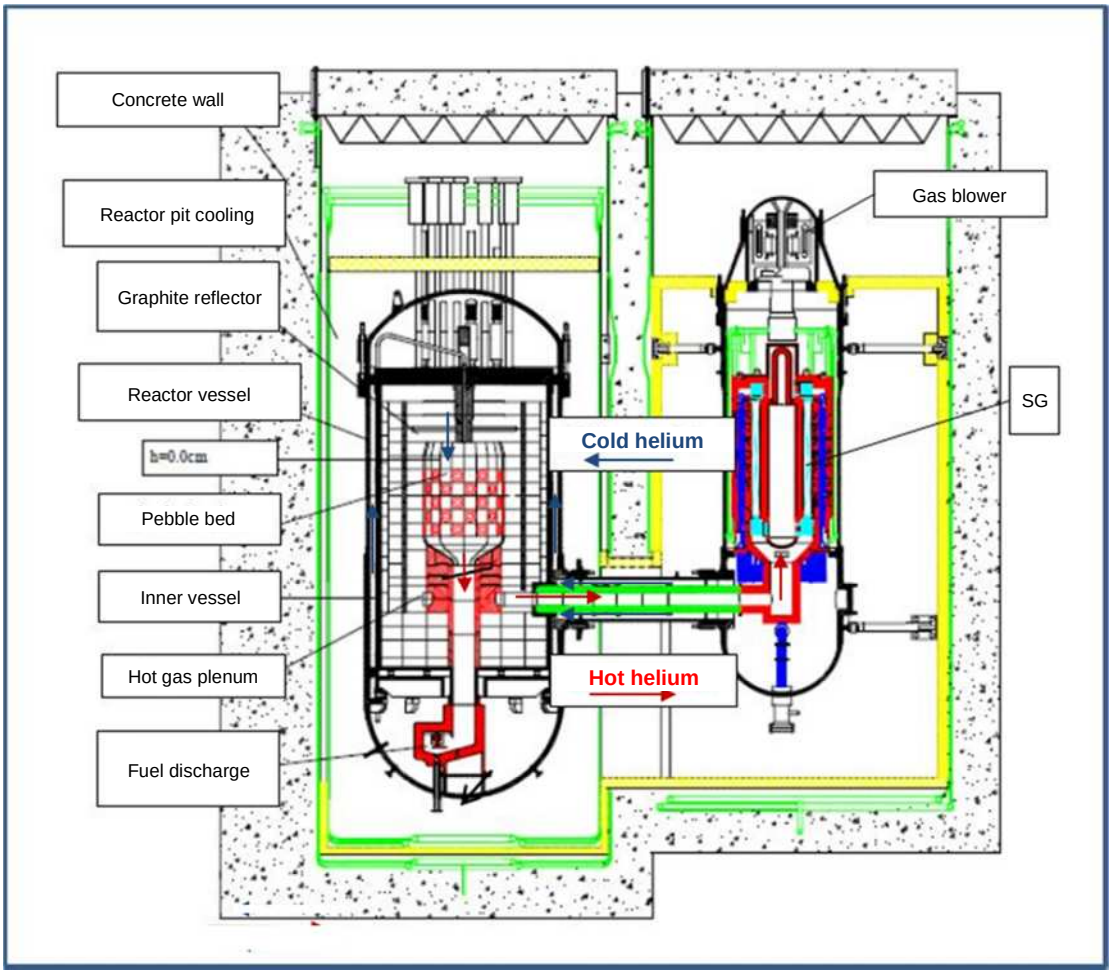

Figure 8: Sectional view of the HTR-10 reactor

3.1.2 *ONGOING PROJECTS AND PROSPECTS FOR DEVELOPMENT*

Studies on helium-cooled high-temperature reactors were resumed in 2001 when the VHTR design was selected by the Generation IV International Forum (GIF). The key players behind this revival were Euratom, via the Framework Programmes for Research and Technological Development, and the US DOE. In 2005 DOE launched the Next Generation Nuclear Plant project (NGNP), which was aimed at building a prototype reactor coupled to a hydrogen production plant located at the Idaho National Laboratory (reference 39). However, DOE recently withdrew from the project and NRC has suspended the certification procedure. It should be noted that the ANTARES project (600 MWth) was developed by Areva partly to respond to call for tenders issued by DOE.

Lastly, since the shelving of South Africa's PBMR project, China is the only country currently developing an industrial prototype reactor. Known as the HTR-PM, this reactor consists of two 250 MWth modules that are coupled to a steam turbine and generate 210 MW of electricity. Modelled after the German HTR-MODUL, the helium temperature inside the HTR-PM will be limited to 750°C. Other projects under consideration in Japan and South Korea do not seem likely to lead to the development of a prototype in the near future.

A European project worthy of mention is the NC2I-R initiative (EC's FP7) led by Poland and which aims to exploit European R&D results and promote the construction of a prototype HTR for cogeneration purposes.

Recently, the members of GIF revised the technical specifications of the VHTR (reference 2) by lowering the core-outlet helium temperature (from 950°C to 750°C), at least for the medium-term development of the system. It should be noted that industrial heating, a potential market for this type of reactor, mainly requires steam at temperatures below 500°C. However, the hoped-for developments in the sulphur-iodine cycle for hydrogen production, which requires temperatures greater than 950°C, did not materialise. Nevertheless, a target of 1000°C has been set for the long term.

3.2 MAIN CHARACTERISTICS

3.2.1 *GENERAL DESIGN*

The Very High Temperature Reactor (VHTR) is a graphite-moderated, helium-cooled nuclear reactor with thermal spectrum. Graphite serves both as a moderator and a structural material for the core.

The core of a VHTR has a power density in the range of 4 to 10 MW/m^3. The heat output generally targeted by designers is between 250 and 600 MW for a thermodynamic efficiency of around 45%. Like the HTR, the VHTR is a modular design.

The neutron properties of the graphite and the small size of the particles allow a large degree of flexibility in terms of fuel composition whilst maintaining a negative power coefficient under all operating conditions.

VHTR fuel is finely divided into spherical particles (Φ = 1mm) dispersed in the graphite moderator. There are two main types of reactor core, the first being a bed of pebbles packed in a cylindrical cavity (Figure 7), and the second being an assembly of prismatic graphite blocks (Figure 9). Annular configurations, which increase the output whilst maintaining the thermal inertia, are also possible for both reactor core types (PBMR or GT-MHR).

The fissile assemblies or pebbles are surrounded by graphite reflectors and insulating carbon bricks.

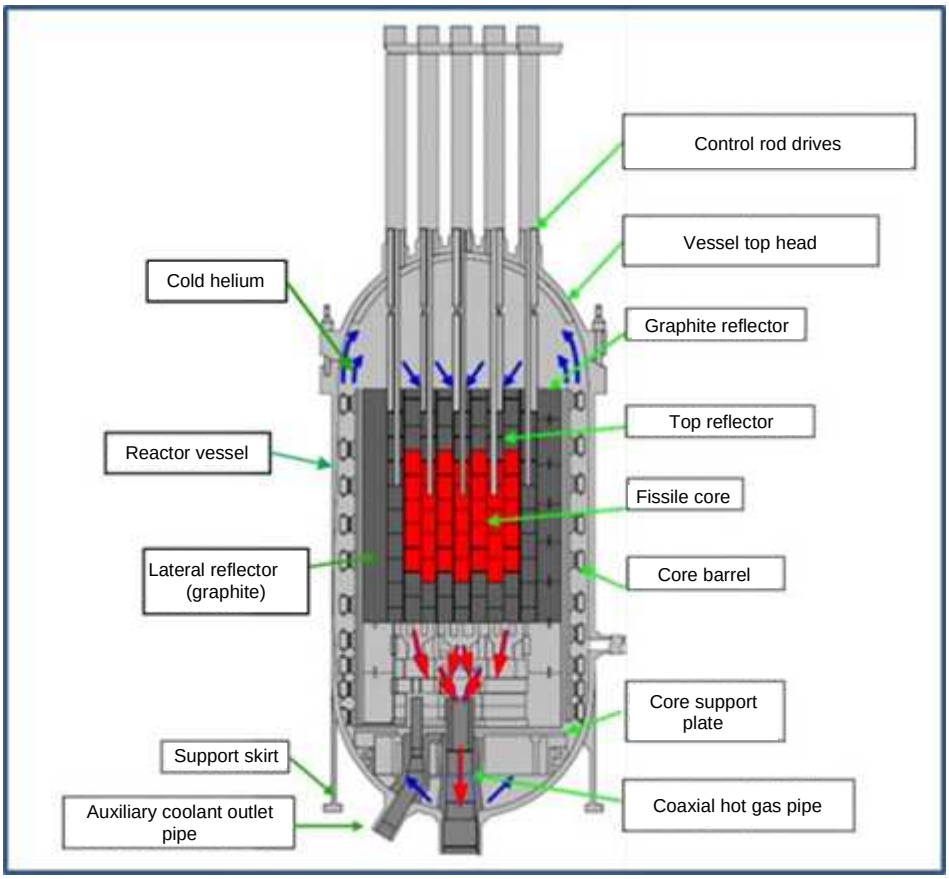

Figure 9: Sectional view of the HTTR vessel

It should be noted that, in a VHTR, the fuel and the helium are separated by more than 1 cm of graphite, whereas the kernel of a particle has a radius of around ¼ millimetre. **This arrangement eliminates all risk of direct contact between the coolant and the fuel apart from hypothetical situations where the core might be exposed to significant oxidation (see Section 3.4.1.1).**

The VHTR design selected by GIF is characterised by a direct cycle that makes use of the expansion of the primary helium in a turbine (Brayton or Joule cycle) coupled to the compressor. This cycle was considered in particular for the GT-MHR and PBMR projects.

However, to get around having to develop an uncooled helium turbine – a technology that has yet to be mastered – Rankine cycle designs were also considered. Thus, upon exiting the core, the hot helium (750°C to 1000°C for the VHTR) is carried to the steam generators (SG), which are connected to the secondary water/steam circuits. The primary helium is circulated by a gas blower.

The pressure housing is maintained at a temperature significantly below the creep limit during normal operation by the circulation of helium at the cold-leg temperature (Figure 9). Internals that convey the hot helium are thermally insulated on their inner walls and cooled on their outer walls by the helium from the cold leg.

The SGs or the turbine (direct cycle) are installed in a vessel that is separated from the primary vessel but connected to it by a coaxial pipe. Because its diameter is greater than its length, this pipe is known as a “cross vessel” (No. 3 in Figure 10). There are also designs wherein the reactor (core, SG and gas blowers) is embedded in a pre-stressed concrete vessel.

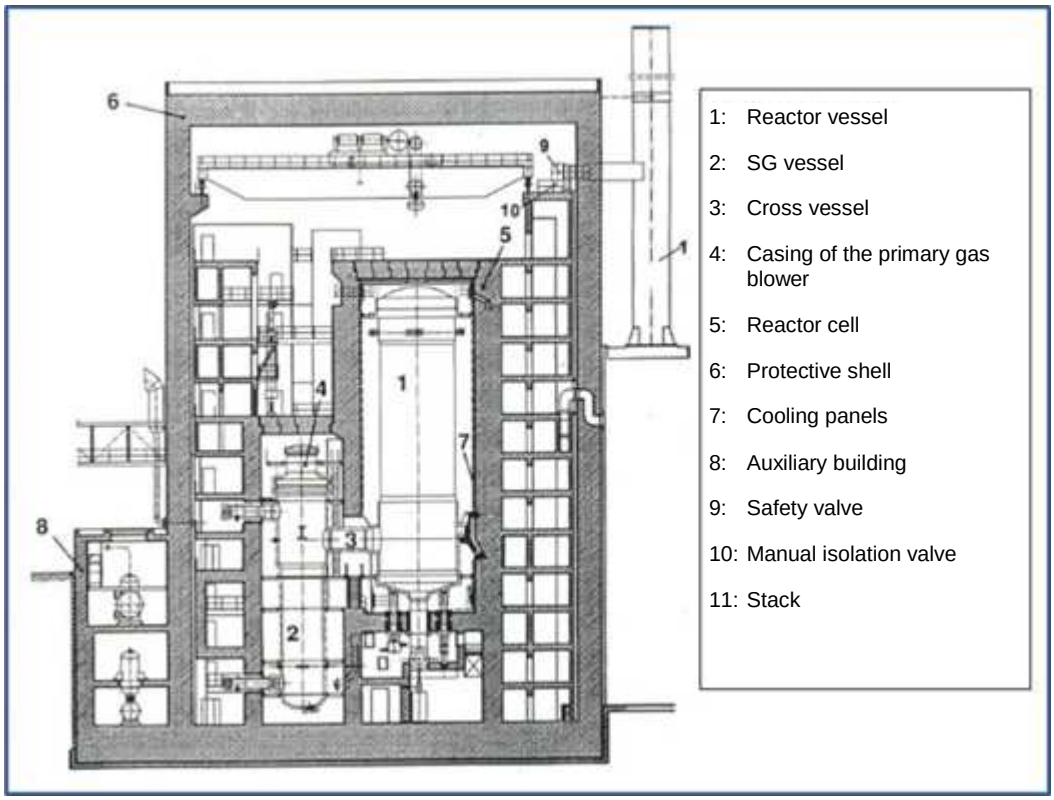

Figure 10: HTR-MODUL – Sectional view of the reactor building (one module)

For reactors that use the Rankine cycle, an alternative to installing steam generators on the primary circuit is to place an "intermediate" gas circuit between the primary circuit and the energy conversion circuit (ANTARES design). The primary circuit is cooled by intermediate heat exchangers (IHX), as in the SFR design. This configuration protects the primary circuit from potential ingress of water/steam in the event of failure of the steam generators tube(s).

3.2.2 *GENERAL SAFETY OPTIONS*

As in the case of an HTR, the main objective in the design of a VHTR is to prevent any accidents that may occur at the facility from leading to significant releases of radioactivity. In particular, the design must physically eliminate any heating of the fuel above the temperature limit guaranteeing its integrity (1600°C for an HTR). This objective leads to the following main safety options:

- designing the core (size, power density, built-in reactivity) so that the fuel cladding and the stability of the core structures are protected during all accident transients used for the safety demonstration;
- determining neutron feedback coefficients that reduce the reactor power in the event of loss of cooling (loss of heat sink or depressurisation) without having to trigger an emergency shutdown;
- designing the core to prevent fuel damage in the event of water ingress (increased moderation) and inadvertent withdrawal of the control rods;
- cooling of the reactor pit and the vessel (second barrier) by an external circuit able to operate in natural convection mode (especially in case of loss of electrical power supply);
- continuous high-efficiency purification and monitoring of primary helium activity levels;

- in the case of a secondary water/steam circuit: limiting, by design, the amount of water that could be injected into the core in the event of an accident and mitigating the induced reactivity insertion (core optimisation).

These options apply to both pebble-bed and prismatic-block reactors.

With regard to the first point listed above, the reactor design approach tends to limit the core power density in zones where peak temperatures could be reached in accident situations (various transients to be considered). Next, the total core power is imposed by the axial power distribution. Thus, in the event of failure of normal cooling, the heat is stored first in the core itself and then in the graphite reflector before reaching the vessel. The core heats up until the radial heat flux balances the decay heat. The heat flux exiting the vessel depends primarily on the radial temperature gradient. The maximum diameter of the core is thus determined by observance of the maximum fuel temperature criterion.

3.2.3 *COOLANT*

The main advantages of helium are its negligible neutron absorption and activation cross sections, and the fact that it does not react chemically with reactor materials. The use of a gas coolant makes it possible to achieve very high operating temperatures.

Compared with an equal volume of liquid coolants, helium makes only a small contribution to the thermal energy that may be stored in the primary circuit during cooling failure. Thus, the product of its density multiplied by its constant-pressure heat capacity ($\rho \cdot Cp$) is 25.8 $kJ \cdot^{-3} \cdot K^{-1}$ for helium (average core-inlet conditions) versus 3976 $kJ \cdot m^{-3} \cdot K^{-1}$ for water under the primary-circuit conditions in a PWR (15 MPa, 300°C).

At high temperature, the impurities present in the helium in trace amounts (O_2, CO, H_2O, FP, etc.) can chemically react with the graphite and the surfaces of the primary circuit, particularly since a passivation layer does not form on the metal surfaces. The selected option is to continuously purify the helium in order to control its moisture, hydrocarbon and carbon dioxide levels and remove tritium and certain fission products from it.

3.2.4 *MODERATOR*

Nuclear-grade synthetic graphite (reference 40) is made from a mixture of petroleum or coal coke, carbon and a binder (resin). In the case of prismatic blocks, this mixture is moulded and converted into graphite by heating it to more than 2000°C (ideally 2800°C). Pebbles cannot be heated to such high temperatures due to the presence of fuel particles. This is why, after treatment, they have a residual carbonaceous phase with properties different from those of graphite (lower mechanical strength, higher porosity, etc.).

The main advantages to using graphite as a moderator are:

- its abundance as a raw material;
- its low neutron absorption cross-section in the thermal spectrum (especially influenced by impurities);
- its resistance to very high temperatures (higher than 3000°C);
- its good thermal conductivity.

Because graphite is a weaker moderator than hydrogenated materials, it is used in large amounts in reactor cores (around 800 metric tonnes for a 600 MWth core fuelled with enriched UO_2). Although HTR cores are large, this gives them a high thermal inertia ($\rho \cdot Cp$ = 3200 $kJ.m^3.K^{-1}$ at 1000°C).

The raw materials and fabrication process have a considerable impact on the behaviour of irradiated graphite elements. Irradiation causes core elements to contract rapidly. This contraction, which is more or less anisotropic[13] depending on the fabrication process used (extrusion or isotropic static compression), is followed by expansion when the fluence exceeds a certain threshold (Figure 11). Core elements must be replaced when their swelling induces excessive mechanical stresses in the core. This phenomenon is less pronounced when the irradiation temperature increases. In addition, an "annealing" phenomenon occurs when temperatures rise above 1200°C, partially nullifying the irradiation effects. Graphite is also subject to irradiation creep, which adds to the swelling.

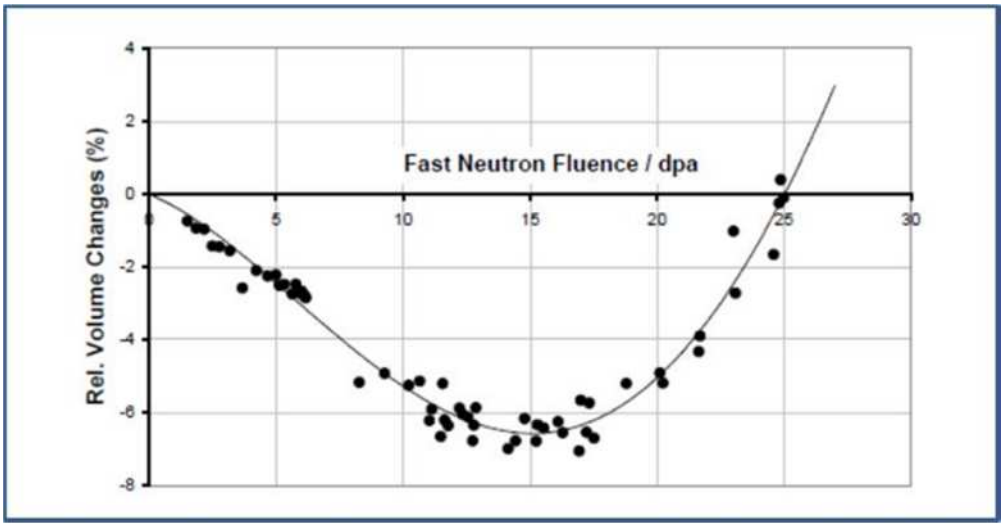

Figure 11: Relative volume changes for samples of ATR-2E graphite
at a 600°C irradiation temperature (as per reference 41)

Irradiation also induces a rapid decrease in the thermal conductivity of graphite. This change in conductivity is therefore the essential reactor design parameter in terms of loss-of-coolant accidents.

Finally, the Young's modulus (elastic modulus) of graphite increases under irradiation, being multiplied on average by a factor of 2 to 2.5 during the lifetime of the blocks or pebbles.

[13] Graphite is an anisotropic material consisting of stacked layers of hexagonal crystals. Synthetic graphite, on the other hand, is made of more or less randomly oriented layers of particles, giving it a relatively isotropic structure at macroscopic scale.

In conclusion, purity and isotropy are the main properties of interest in the graphite used in VHTRs. The characteristics of this graphite change significantly under irradiation. However, the high temperatures of the graphite used in these reactors limit the magnitude of some phenomena (decrease in thermal conductivity and swelling).

Graphite dust

A particularity of graphite-moderated reactors is the production of carbonaceous dust in the primary circuit. This dust is a potential contamination pathway that is of particular consequence in the event of primary circuit depressurisation (resuspension).

The amount of dust liable to be produced in a VHTR is difficult to estimate. A review of the available studies on HTRs was conducted in 2005 as part of the European RAPHAEL project. For example, the density of dust deposited on the surface of the primary circuit THTR-300 (pebble bed) – measured after 480 EFPD (Equivalent Full Power Days) – was 1 mg/cm^2. However, there are dust accumulation areas, particularly in the heat exchangers and the steam generators. The mass of dust produced annually in an HTR is therefore estimated at between a few kilogrammes and 100 kilogrammes. By contrast, the helium circulated contains little dust.

It would appear that prismatic-block cores generate less dust than pebble beds. The reasons for this are twofold: the pebbles are continuously subjected to friction and their incomplete graphitisation makes them more sensitive to abrasion. Furthermore, the pebbles must be placed back in the core more than ten or so times during their life in order to ensure that they are homogeneously irradiated. The pebble transfer system must therefore be specially designed to mitigate the formation of dust and to allow for its potential recovery.

3.2.5 *FUEL*

The TRISO fuel in HTRs consists of a spherical kernel of fissile material coated with four successive layers that form a barrier to fission products. This fissile kernel has a diameter of around 0.5 mm and the TRISO particle a diameter of 1 mm. The TRISO particles (Figure 12) are dispersed in a graphite matrix to form the fuel elements, which are in the form of pebbles or compacts. TRISO fuel is also considered for VHTRs.

The particle fabrication process starts with the formation of fuel kernels obtained, for example, from resin droplets gelated in an ammonia solution (sol-gel process). These droplets are then sintered to obtain solid particles. These particles are then sorted mechanically to keep only those whose shape most closely resembles a sphere. The buffer, the layers of pyrolitic carbon and the layer of silicon carbide are obtained by vapour deposition in a thermochemical reactor.

The porous carbon buffer limits the pressure in the particle, which is estimated at nearly 70 MPa at end-of-life. It protects the subsequent layers by retaining part of the fission products and absorbing their kinetic energy. Lastly, it forms a buffer that protects the pyrolitic carbon from reactions with the oxygen released by fissions in the fuel. This function is very important, especially for kernels composed solely of oxides.

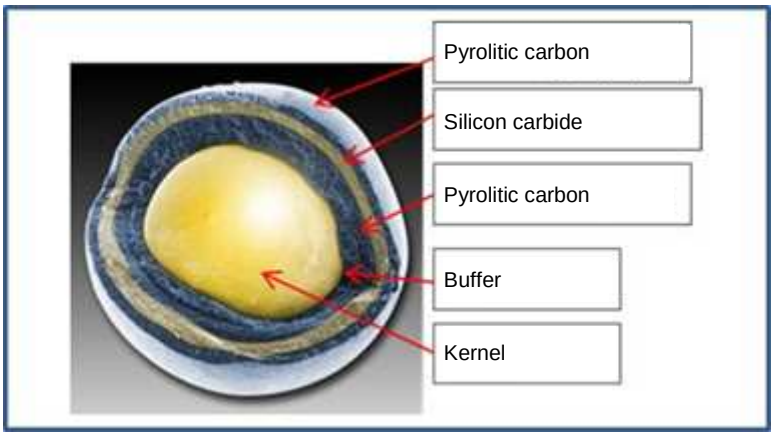

Figure 12: TRISO particle

The layers of pyrolitic carbon help to keep noble gases and halogens inside the particle.

The layer of silicon carbide (SiC) provides high mechanical and thermal resistance and is leaktight to fission products (most notably caesium-134, caesium-137 and strontium-90) and tritium.

Qualification of TRISO fuel for an HTR

An essential step in TRISO fuel qualification consists in linking the fabrication parameters to fuel quality and performance. The approach developed is semi-empirical.

Firstly, it is based on various particle inspection techniques, which include density measurements, sphericity measurements and destructive testing. For example, good particle sphericity is important to prevent the concentration of stresses in the silicon carbide coating. The isotropy of the pyrolytic carbon layers is also assessed. **It should be noted that, due to the large number of particles fabricated and the (current) impossibility of inspecting all of them, faulty particles are placed inside the core with fresh fuel.** Designers have thus determined, by means of sampling, an expected particle failure rate. In the case of the last pebbles fabricated in Germany, this rate amounted to a particle failure fraction of 3.10^{-5} in fresh fuel. A fraction of 6.10^{-5} had thus been adopted for fresh fuel in the HTR-MODUL safety analysis.

Secondly, fuel qualification conventionally includes particle irradiation tests that were conducted under normal HTR operating conditions up to maximum burn-up rates of between 9 at% and 11 at%. Based on the irradiation results for the HTR-MODUL, the particle failure fraction during irradiation was estimated at 2.10^{-5}.

Because external contamination of the particles and the graphite is very low, faulty particles before and during irradiation account for the bulk of sources of radioisotopes in the primary circuit. Ensuring very low failure rates is therefore essential to demonstrating compliance with release limits during normal operation.

Lastly, qualification of fuel under accident conditions consists in subjecting irradiated batches to heat-up tests for several hundred hours. The German fuel was qualified by conducting several series of tests at between 1600°C and 1800°C at the KÜFA facility at Jülich. These tests showed that the particles remained virtually leaktight[14] up to a temperature of around 1600°C and that no additional failures were observed at this temperature. Although these tests did not reveal a cliff edge effect in terms of particle integrity, they did reveal progressive degradation of integrity and an increase in the failure rate from 1600°C onward. As a result, in the case of the HTR-MODUL, the statistical analysis of the heat-up tests led to the definition of a particle failure fraction of 6.10^{-5} under accident conditions (a value of 6.10^{-4} was adopted for the safety analysis). This fraction was deemed compatible with the release limits during accident conditions. **The German designers therefore used a temperature of 1600°C as a bounding limit that must not be exceeded under accident conditions in order to ensure the integrity of TRISO particles**. It should be noted that this limit temperature is associated with a specific type of pebble and a maximum burn-up. It is neither a design limit nor a technological limit for TRISO particles.

In conclusion, HTR fuel is currently qualified for a maximum burn-up of 11 at% and maximum temperatures of less than 1250°C and 1600°C respectively, during normal and accident conditions (reference 47). However, regarding particle behaviour modelling, it appears that the complex phenomena involved are not fully explained or identified. This is particularly the case of diffusion of the silver isotope ^{110m}Ag through the SiC layer (reference 45). It should also be noted that no plutonium-based fuel was fabricated.

Qualification of fuel for the VHTR design

It can be stated that, for a VHTR, the objective would be to fabricate fuel that is qualified for a burn-up of 15 at% and a maximum temperature of 1800°C. During normal operation, the fuel would reach a maximum temperature of 1250°C.

Increasing VHTR operating temperatures to levels higher than in an HTR considerably narrows the margin over the maximum TRISO fuel temperature during normal and accident conditions. During normal operation of the HTR-MODUL, the margin was around 330°C[15] for an average core outlet helium temperature of 700°C. This margin would thus appear to be very low for a helium temperature of 950°C, if the admissible limit defined for TRISO particles during normal operation (approx. 1250°C) was retained. Although fuel temperature uncertainties can be reduced, the same is not true for factors of uncertainty such as those associated with the random paths of the pebbles in the core. This is also the case for local changes in the helium flow rate (bypasses, pebble bed density). All in all, these uncertainties are believed to be close to 200°C for a pebble-bed reactor. Many bypasses between the blocks and in the reflector of prismatic-block reactors also create an uncertainty about the fuel temperature, although this uncertainty is less significant overall than for pebble-bed reactors.

[14] That is, the retention rate for significant FPs for the source term (iodine, caesium and strontium) changed little during the test as compared to normal operation.

[15] 1200°C threshold adopted for normal operation, or a maximum nominal fuel temperature of 870°C.

A number of R&D programmes have been developed to address these problems with the initial aim of testing TRISO fuel limits. The European RAPHAEL and ARCHER projects are two examples. Figure 13 presents the results of an irradiation test (HFR-EU1bis) conducted at the Petten HFR facility, in the Netherlands, on five pebbles fabricated in Germany in 1987 and containing a total of 47,800 particles. These pebbles were irradiated at temperatures between 950°C and 1250°C (temperatures representative of a VHTR) and up to a burn-up of 11 at%. The release rates for five isotopes were measured continuously during the irradiation test by analysing the helium coolant. These rates remained at least five times lower than those that would result from the loss of integrity of just one particle. Only partial losses of particle integrity can explain the increase in the releases observed during the irradiation.

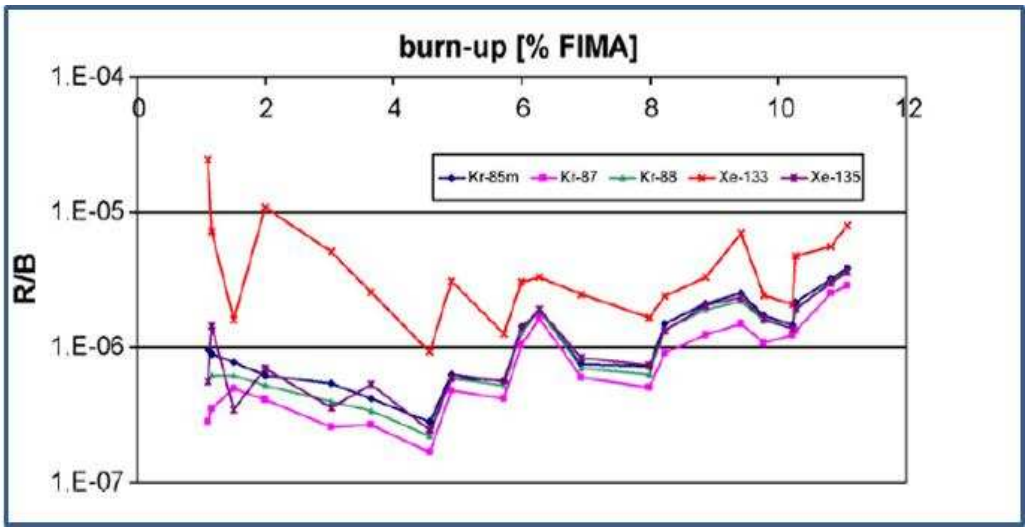

Figure 13: Release rates for five radioactive isotopes of krypton and xenon for irradiation temperatures between 950°C and 1250°C (reference 44)

When the fuel temperature reaches 1250°C, some metal fission products pass through the SiC layer. This is notably the case of the isotope ^{110m}Ag[16] , which was shown in the Dragon and AVR reactors (reference 43). Although produced in small quantities, this isotope can reach the helium and settle on the coldest parts of the primary circuit (heat exchanger or steam generator, blowers). This contamination would make equipment maintenance difficult. The mechanism that allows silver to pass through the SiC layer is not yet fully explained. Whatever the case, the 1250°C limit, minus the uncertainties on fuel temperature, provides a fuel safety criterion during normal operation.

Heat-up tests at 1600°C were conducted in the KÜFA facility with the pebbles from the HFR-EU1bis test. A comparison of these tests with those performed on pebbles irradiated for the purposes of qualification of the HTR-MODUL fuel reveals a significant increase in the caesium released from the particles in the HFR-EU1bis samples (factor of 100). According to reference 54, these results suggest that the HFR irradiation conditions exceeded the acceptable limits for these fuels. Nevertheless, they must be confirmed by other irradiations for the reason that the conditions effectively reproduced in the HFR are not fully representative (accelerated irradiation, temperature excursions during irradiation, etc.).

[16] The metastable isotope ^{110m}Ag is produced by activation of the isotope ^{109}Ag and has a half-life of 253 days. The isotope ^{111}Ag, also produced by activation of the isotope ^{109}Ag, can be released into the primary circuit. However, it has a half-life of just 7.5 days.

Despite the results mentioned above, IRSN considers that TRISO fuel may be the key asset of the VHTR design even though current fuels are not yet qualified for the targeted operating conditions.

Several avenues for fuel performance improvement are being considered, particularly regarding coating materials (use of ZrC instead of SiC) and fuel quality control methods.

3.2.6 *BARRIERS*

According to its designers, the VHTR has the same barriers as a PWR but the importance placed on each is different. Figure 14 illustrates the barriers of a VHTR.

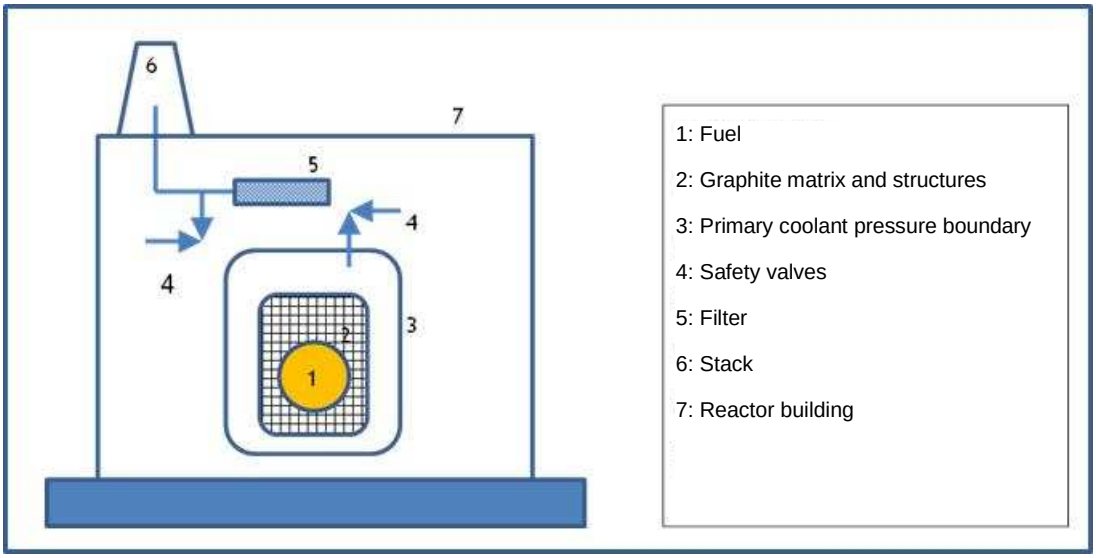

Figure 14: Schematic diagram of the barriers in the HTR-MODUL reactor

In the schematic diagram above, the coating surrounding the fuel kernels in TRISO particles makes up the first barrier (1). The metal primary circuit pressure boundary is the second barrier (3). The primary circuit is protected from potential overpressure by valves that discharge into the reactor containment (4). Lastly, the reactor building is ventilated continuously and is equipped with safety valves (4) and filters (5). This is because, unlike the reactor building of a PWR, it is not designed to withstand internal overpressure. In practical terms, the reactor building of an HTR cannot be described as a barrier. It serves primarily to protect the reactor against external hazards.

3.3 CONTROL OF SAFETY FUNCTIONS

The following subsections describe, using the HTR-MODUL AND HTTR projects as examples, how the safety functions are ensured and the strategy used to control the main accident scenarios. These considerations apply also to the VHTR design.

3.3.1 *REACTIVITY CONTROL*

Inherent characteristics

The core of a VHTR is designed (density of fissile material, enrichment, isotopic vectors) so that the feedback coefficient related to the power is always negative and essentially controlled by the Doppler effect and the moderator heating effect. Whatever the power level, an increase in fuel temperature causes the core power to decrease. Thus, reactor trips are not necessary in the short term in the event of loss of coolant or primary circuit depressurisation. The heating of the core is sufficient to stop the chain reaction and stabilise the core at a low power level. Insertion of a neutron absorbent is required to control the effect of the disappearance of xenon, which led to a rise in reactor power (limited) after a few hours.

The reactivity effects of overcooling, the amplitude of which can be limited by circuit design, are mitigated by the thermal inertia of the graphite.

Lastly, unlike liquid or gas-cooled fast reactors, the mechanically compact core of the VHTR is virtually insensitive to mechanical vibrations. However, disturbances in core stability can lead to power level variations. Such disturbances, which may occur both during earthquakes and normal operating conditions, consist primarily of:

- deformations in the stacking of the graphite blocks induced by irradiation and thermal expansion (feedback from Fort Saint Vrain);
- random variations in the pebble-bed density.

Reactor trip systems

There are two diversified and redundant reactor trip systems: a system of control rods (usually B_4C) supplemented, for example, by tanks containing boron carbide balls. Located at the top of the reactor, these tanks can be gravity drained into cavities built in the reflector. In the event of abnormal core heating, the draining of these tanks can be initiated by a Curie point passive system. Such a system was installed at Fort Saint Vrain[17].

In prismatic-block reactors, the control rods can be inserted in and around the core. However, it is not possible to insert the control rods directly into a pebble bed. Instead, they are inserted into the reflector. Lastly, nitrogen injection can also help to bring the reactor to cold shutdown, such as in the event of depressurisation (AVR).

3.3.2 *DECAY HEAT REMOVAL*

After normal shutdown, the decay heat is removed by the normal circuits (startup/shutdown circuits) and the reactor pit cooling circuit. To reduce the time required for the temperature to drop to the level at which the fuel can be handled, the reactor can be equipped with a dedicated shutdown cooling circuit.

In the event of failure of the normal cooling circuits or depressurisation accidents, the decay heat is removed only by means of conduction in the core and convection/radiation from the main vessel to the reactor pit cooling circuit. This circuit, which runs continuously, is designed to maintain the reactor pit, vessel and concrete at temperatures compatible with their mechanical integrity. It may be designed to operate on natural convection.

[17] It should be noted that, unlike with the HTR-MODUL, the trip system in the event of loss of coolant is required at Fort Saint Vrain.

The reactor pit cooling circuit is designed according to the rules applicable to items important to safety (high level of quality, redundant architecture, backup power supplies, etc.).

The concrete of the reactor pit and its environment (ground, air, etc.) also help to remove the decay heat from the reactor. The inertia of the core and the surrounding structures limits the rise in temperature of the second barrier. For example, the designer of the HTR-MODUL evaluated the temperatures of the fuel and structures in the event of reactor depressurisation and failure of all the cooling systems. The temperature of the reactor vessel would rise to 600°C but the temperature of its support structures would not exceed 300°C. This would not compromise their strength. In addition, failure of the reactor pit cooling circuit would have no effect on the maximum temperature reached by the fuel, i.e., 1620°C. However, the temperature inside the reactor pit would be slightly lower than that of the vessel. This would lead to damage to the surface concrete. Based on these results, the grace period for cooling the reactor pit was estimated at some 15 hours.

3.3.3 *CONFINEMENT OF RADIOACTIVE MATERIALS*

The confinement function is ensured by three successive barriers and the filtered venting system in the reactor building. However, these barriers do not have the same efficiency. Due to the robustness of the first barrier (TRISO), the designers assigned only a relative retention function to the other two barriers.

In practical terms, the second barrier has a significant leakage rate during normal operation. For example, the leakage rate estimated for the HTTR is around 75% by weight per year. In the past, these leakages were considered acceptable given their low radiological consequences. It will nevertheless be noted, for example, that the primary circuit of the THTR was surrounded by a controlled area consisting of rooms maintained at a pressure lower than that of the reactor building in order to reduce operator exposure.

A primary break would cause the circuit to be depressurised in a matter of minutes and the pressure in the reactor building to rise rapidly. The non-condensable nature of helium and its high temperature would cause this pressure to decrease more slowly than in the case of a steam release, for example. However, this helium is supposed to be only slightly contaminated during normal operation and the depressurisation would not lead to particle degradation (at least in the short term). These two specific features of the VHTR explain the particular approach of the designers, who did not look for short-term confinement of potential releases. Thus, in the event of sudden depressurisation, the pressure inside the reactor building would be limited by safety valves (principle of the HTR-MODUL) calibrated at a few tens of millibar. The release would pass unfiltered directly into the stack. Once the safety valves would close, the releases could once again be filtered. Following a depressurisation transient, the pressure in the reactor building is close to atmospheric pressure. There would be no more driving pressure to create a risk of subsequent dispersion of the releases outside the reactor building. Only core heating could cause a slight rise in the reactor building pressure, since the injection of additional helium in the primary circuit is not necessary. The reactor building would thus play an important role in retaining radioactive elements.

The safety approach for confinement is thus based, for the short term, more on preventing contamination of the primary helium, monitoring and purifying it continuously.

IRSN notes that, in the HTR designs developed so far, only the first barrier provides a truly efficient integrity function. This feature is reportedly carried over to the VHTR. **Nonetheless, even if the helium is only slightly contaminated, the HTR confinement strategy should be revised to improve its performance in order to meet the objectives set for Generation IV reactors. Particularly, design provisions for mitigating the environmental consequences of sudden depressurisation should be investigated.** Taking the THTR as an example, the rooms surrounding the primary circuit should be able to serve as buffer zones between the primary circuit and the reactor building.

Similarly, IRSN restates that the risks of containment bypass should be mitigated to prevent unfiltered releases. Following this principle, the helium purification circuit in the HTR-MODUL is located in an auxiliary building but the circuits exiting the reactor building can be isolated (redundant valves). All the primary helium circuits in the future HTR-PM are reportedly located in the reactor building.

3.4 RISK ANALYSIS

3.4.1 *RISKS INHERENT TO DESIGN*

3.4.1.1 Air ingress and graphite fire risk

The ingress of air in the primary circuit triggers the following reactions with the graphite:

- $C+O_2 \rightarrow CO_2$ (formation of carbon dioxide, exothermic reaction);
- $2C+O_2 \rightarrow 2CO$ (formation of carbon monoxide, exothermic reaction);
- $C+CO_2 \Leftrightarrow 2CO$ (Boudouard reaction, endothermic reaction);
- $C + H_20 \rightarrow CO + H_2$ (endothermic reaction[18]).

Note that 1 m^3 of air allows the combustion of 0.1 kg to 0.2 kg of graphite (reference 46).

Graphite fires are a special case for which the aforementioned oxidation reactions are self-sustaining. For graphite oxidation to be maintained in an HTR/VHTR, air would have to be circulated in the core at a rather high rate to provide enough oxygen but without inducing overcooling. In particular, two breaks adequately positioned in the primary circuit to induce a chimney effect in the core and create conditions leading to a graphite fire should be considered. Furthermore, the purity and high density of the graphites used in VHTRs is a limiting factor of oxidation. Lastly, if depressurisation is caused by a single small break (line, pebble discharge tube, etc.) significant core oxidation leading to degradation of particle integrity is very unlikely. These considerations, accompanied by a specific approach for guaranteeing the workmanship of the primary circuit, led the HTR designers to not consider graphite fires.

[18] Reaction due to the air moisture. Also applies to cases of water ingress.

However, following the Windscale and Chernobyl accidents, the risk of graphite fire was investigated in detail, particularly for gas-cooled graphite-moderated reactors and hypothetical situations began to be considered for the HTRs. If the VHTR core were exposed to air, the graphite coating the particles would be gradually converted to gas in the areas where the temperature is above 500°C to 600°C. At core temperatures, oxidation is primarily a surface phenomenon and the thickness of the graphite structures progressively erodes. Once the particles are exposed, the pyrolytic carbon is in turn oxidised and the air reacts with the SiC layer. Several air oxidation tests have been conducted on pebbles, compacts and fresh and irradiated particles, particularly at the KORA facility at Jülich. On average, several days (around 100 hours at 1100°C) are required before the graphite and the pyrolytic carbon in the pebbles are destroyed under natural air convection conditions and at temperatures between 900°C and 1400°C, representative of a depressurisation accident (reference 47). In the presence of air, the SiC layer progressively degrades when the temperature reaches between 1300°C and 1400°C. The FP release rate increases substantially with failure of the SiC layer and oxidation of the particle fuel kernels. However, the mechanism of SiC oxidation depends on various parameters that are less easily identified than for graphite (particularly irradiation).

In conclusion, the mechanisms that lead to graphite oxidation, or even to fire, in the event of air ingress are currently well known and modelled (reference 46). However, once a particle is exposed, bounding approaches must be applied to estimate the degradation of the SiC layer and the number of degraded particles. R&D on the VHTR design is currently focused on means for mitigating the consequences of air ingress, such as inerting and the design of oxidation-resistant pebbles. It should be noted that operators can take advantage of the grace period to intervene locally, provided that the radioactive contamination permits it.

3.4.1.2 Risk of water ingress

Firstly, it should be recalled that the risk of massive water ingress while the reactor is operating at power applies only to VHTRs equipped with steam generators. Diffuse water ingress is supposed to be detected by the purification system used to monitor the moisture in the helium at a very low level during normal operation. Careful monitoring of this parameter is very important for preventing corrosion of metal surfaces. Furthermore, operating experience feedback from Fort Saint Vrain has shown that although water ingress is not highly damaging to graphite (provided the reactor is shut down), it can jeopardise the operation of the control rods.

The ingress of massive amounts of water into the primary circuit of a VHTR could have the following consequences (reference 48):

- rapid increase of reactor power (moderation effect);
- graphite corrosion and the production of flammable gases (CO, H, CH_4);
- washout of the contamination deposited in the primary circuit;
- increase in the primary circuit pressure and in potential releases to the environment (depending on the amount of water injected and the calibration pressure of the safety valves in the reactor building).

It should be noted that water-induced oxidation of graphite is an endothermic reaction and helps to cool the reactor. This is why the consequences of water ingress are considered to be mostly significant in the short term (FP entrainment, production of flammable gases, etc.).

Furthermore, if water ingress is concomitant with a break in the primary circuit, the release of gases and graphite dust into the reactor building might lead to an explosion hazard[19].

The detection of moisture in the primary circuit or variation in core power would automatically trigger actions aimed at mitigating the consequences of water ingress, i.e.:

- reactor trip, shutdown of the blowers and isolation of the steam generator;
- activation of a separator on the primary helium purification circuit (dedicated to this function).

The aim of the aforementioned measures is to limit the amount of water injected into the core so that it does not lead to an uncontrolled power excursion.

The safety approach developed for VHTRs, carried over from that for HTRs, is based as much as possible on preventing and mitigating the consequences of water ingress by the design and size of the primary circuit. For example, the principle applied for the HTR-MODUL (reference 48) consists firstly in limiting the amount of fissile material in each pebble so that the increase in the core power caused by water ingress can be offset by neutron feedback. Secondly, the design of the steam generators and the primary circuit should make it possible to avoid opening the safety valves before isolating the steam generator.

Lastly, the management of the risk of massive water ingress is based firstly on limiting core power and the water flow rates required in the steam generator tubes. This approach must be supplemented by automatic protective measures associated with continuous monitoring of the gases in the helium, especially in cases of diffuse water leaks. **Nevertheless, the risk associated with the production of flammable gases in a VHTR must be investigated in hypothetical situations combining water ingress and primary circuit breaks**. Furthermore, in the event the primary circuit valves would be called on (breaks of several SG tubes, SG isolation failure), IRSN notes that the source term potentially generated by "washout" of the primary circuit is still poorly defined.

In terms of overall management of the risk of water ingress, IRSN notes that use of a secondary water/steam circuit is not an inherent feature of the VHTR design, which is more oriented towards high-temperature gas conversion cycles (Brayton cycle).

[19] R&D measures being taken as part of the EU ARCHER project are intended to define the amounts of fuel gas that are admissible in helium.

3.4.2 *ACCIDENT SEQUENCES*

3.4.2.1 Loss of primary coolant

It should be recalled that loss of primary coolant has no significant consequences on fuel particle integrity. The reactor is shut down by the neutron feedback effect due to core heat-up. Decay heat is removed by conduction up to the vessel wall, then by convection and radiation in the reactor pit. Figure 15 illustrates changes in the maximum temperatures of the main structures of the HTR-MODUL reactor during a depressurisation accident (reference 49).

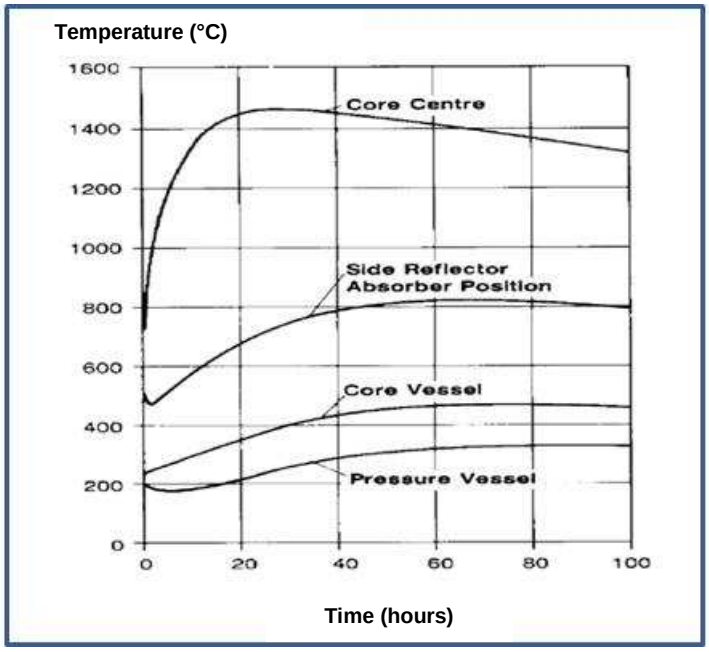

Figure 15: Changes in maximum temperatures during depressurisation (HTR-MODUL)

In the short term, a fraction of the plate-out dust is resuspended and entrained in the reactor containment if a primary circuit break occurs. The matter of the source term associated with a primary break is explained further on, in the section on severe accidents.

The reader is reminded that, depending on the size of the break leading to depressurisation, more or less substantial ingress of air occurs in the core, as described in detail in Section 3.4.1.1.

3.4.2.2 Reactivity insertions

The VHTR may be designed so that reactivity insertions caused by inadvertent moderator ingress in the core remains naturally controllable by neutron feedback. This is accomplished by optimising the margin between the in-operation moderation ratio and the theoretical optimum moderation. The excess reactivity at startup must also be limited (which is easier with pebble-bed reactors).

Likewise, the design may be modified to limit the effect of core overcooling (transient mitigated by the thermal inertia of the graphite).

The bounding accident considered for pebble-bed reactors is the withdrawal at maximum speed of all the control rods, the core being in a penalizing state. Such a scenario would have no consequences for the HTR-MODUL reactor.

For the HTTR, the accident of control rod withdrawal with lift limitation and/or emergency shutdown (reference 55) was adopted for the safety demonstration. The prismatic-block reactor has a lower margin regarding the risk of rod withdrawal on account of the excess reactivity intended to compensate for fuel burn-up during the cycle. It does not seem that the total control rod withdrawal accident has been taken into consideration. It should be noted that, in the case of the HTTR, the initial fuel temperature (1495°C) is already close to the maximum adopted for accident situations (1600°C). However, the core of the MHTGR (reference 56), another prismatic design developed by General Atomics, could in theory safely withstand a total control rod withdrawal accident.

Rod ejection generally is not considered in the reactors already built. Calculations made using the characteristics of the PBMR project show that the power peak induced by such an accident, followed by depressurisation, could be acceptable in terms of temperatures and stresses in the fuel particles (reference 57). That said, some design provisions make it possible to reduce the risk of ejection (mechanisms enclosed in the vessel for the HTR-MODUL).

3.4.2.3 Severe accidents

There is currently no consensus on the definition of a severe accident for the VHTR design.

Nevertheless, IRSN considers that an investigation of hypothetical scenarios of massive air ingress could be a possibility (see Section 3.4.1.1). Such a scenario would lead to substantial and prolonged core oxidation.

In the short term, given the slow kinetics of degradation of the VHTR core during all possible situations, hypothetical accidents with degradation of the second and third barriers would cause only part of the activity accumulated in the primary circuit during operation of the reactor to be released to the environment.

The studies conducted as part of the European RAPHAEL and ARCHER projects show that the analytical models currently available to assess the source term generated during primary circuit depressurisation are not satisfactory. Only empirical assessments are currently achievable. It should be noted, however, that the radioactive inventory that may be released early would consist primarily of relatively small amounts of isotopes of krypton, xenon, caesium, iodine, tellurium, strontium and silver (reference 58). This inventory would be primarily associated with dust from the primary circuit.

An example of an approach to the medium-term phase of a severe accident is given in the safety approach developed for the HTTR (reference 55). In the so-called hypothetical accident scenario, air would be circulated in the core following failure of a coaxial pipe. This would cause the core to oxidise for 30 hours until it sufficiently cooled and the reaction stopped by itself. In this scenario, particles could also be oxidised (with conservative assumptions). JAEA calculated that the consequences of releases from the stack and from the ground are 6.3 mSv whole body and 1.3 mSv for the thyroid (adult), when the same release assumptions as for PWR fuel are considered (100% of noble gases and 50% of iodine).

As JAEA notes in reference 55, this approach does not sufficiently take into account the special features of VHTR fuel. Furthermore, IRSN does not have sufficiently detailed information to assess the approach used for the VHTR. That said, the following general comments may be made:

- Reactor vessel failure was not taken into account in the HTR safety approach. Investigating the consequences of this accident would require taking into account the risk of erosion of the concrete and of the production of flammable gases in contact with the graphite in the core at very high temperature.
- Although there is no risk of an explosive phenomenon that is equivalent, in terms of energy potentially released, to the steam explosions investigated for the PWR, flammable gases and dust could be released into the reactor containment and lead to an explosion hazard.
- Oxidation of the graphite leads to structural erosion (conversion into gas) that, if localised on the support structures, could cause the core to collapse.
- In the medium term, oxidation of the graphite could lead to overheating of the vessel, its support structures and possibly the reactor pit, which could jeopardise their resistance.

As regards the early depressurisation phase, IRSN notes that the HTR-MODUL safety analysis report mentions 1 kg of dust released by failure of a DN 65 line[20] (failure of the pebble discharge tube). This very low value would have to be re-assessed, particularly if larger primary circuit breaks were adopted (situation ruled out for the HTR-MODUL). The study on the cross vessel break is a case in point[21].

These comments show that studies of severe accidents with significant potential releases remain to be developed for the VHTR even if it is highly unlikely that such accidents would have significant short-term consequences.

3.5 ENVIRONMENTAL IMPACT, RADIATION PROTECTION AND DECOMMISSIONING

The VHTR design stands out for the presence in the core of a significant amount of irradiated graphite. Graphite takes up part of the primary circuit contamination and contains activation products from impurities remaining after purification. As stated above, graphite dust is created during reactor operation by corrosion and mechanical friction (handling operations, friction between the pebbles themselves and with the reflector, etc.). This radioactive dust plates out on various parts of the primary circuit, in areas where the helium velocity is low and in the handling circuit (pebble-bed reactors). Dust may also be created by erosion and corrosion of the metal walls of the primary circuit. However, most of this dust is composed of graphite and can bind or adsorb radioactive pollutants.

The presence of dust will have an impact on both radiation protection and radioactive releases during normal and accident conditions.

[20] The pebble discharge tube is the largest line on the primary circuit of the HTR-MODUL. A break on this tube is considered to be the initiator of the bounding depressurisation scenario.

[21] JAEA's assumptions on this matter for the HTTR are not known by IRSN.

Primary coolant composition

When helium contains impurities such as H_2, CO, CO_2 and CH_4, it is corrosive and must be purified. Graphite can also be susceptible to the presence of impurities that can oxidise and thus degrade it. The radiological inventory will contain not only the various corrosion products of the steel structures (nickel, cobalt, molybdenum, chromium, etc.) but carbon as well.

Carbon-14 (^{14}C), which is related to the presence of graphite, and technetium-99m are formed if molybdenum is released from the steel.

Radiation protection

Available information on radiation protection feedback (references 51, 52, 53) describes low collective doses:

- The AVR (15 MWe): apart from at the start 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 approx. 0.2 person-Sv/yr by the end of its life.
- The THTR-300 (300 MWe): the collective dose was approximately 0.1 person-Sv/year at the end of its life.
- The Fort Saint Vrain reactor (330 MWe): the collective doses did not exceed 0.03 person-Sv/yr between 1974 and 1978.

These values are lower than the annual doses encountered in the 900 MWe reactors in operation in France (approx. 0.7 person-Sv/year/reactor). However, they were measured in reactors that have not been operated over long periods. Furthermore, the modularity of the HTRs calls for comparing two fleets with an identical total power (around a factor of 10 for the number of HTRs for an equivalence with a fleet of 900 MWe PWRs).

Maintenance of the equipment on the primary circuit may be the source of most of the dose delivered to workers. The radioactive dust carried through the primary circuit may have a tendency to build up in the cold legs of the circuit or areas where the coolant rate is low. However, exploitation of the German operating experience feedback regarding this matter is insufficient. The isotope ^{110m}Ag, which can diffuse into the metal parts of the equipment on the primary circuit (reference 54) could also pose significant radiation protection problems during maintenance operations.

Lastly, the fuel-bearing particles would not be reprocessed, just disposed of. As a result, there would be no occupational exposure from reprocessing of the irradiated fuel. Such exposure would be limited to waste handling and facility decommissioning operations. However, reprocessing of irradiated graphite could lead to occupational exposure.

Furthermore, as with the SFR, LFR and GFR designs, the possible transmutation of minor actinides would not seem to be conducive to radiation protection.

Liquid and gaseous waste management

As stated above, TRISO particles are not leaktight during normal operation. Gaseous fission products, tritium and other radioactive materials are contained in the primary helium and are released through continuous primary circuit leakage (non-negligible). Another release pathway is composed of the helium purification system, whose filters (molecular sieves and cryogenic filters) are periodically regenerated by fresh helium that is subsequently discharged to the stack (HTR-MODUL strategy).

It seems that, except during pre-maintenance decontamination of the equipment on the primary circuit, the VHTR will produce few liquid effluents.

Lastly, storage of irradiated fuel particles is envisaged. Releases could occur in storage areas if these particles are not perfectly leaktight.

Decommissioning

Although several HTRs are currently being decommissioned (such as the AVR in Germany and the Fort Saint Vrain reactor in the USA), IRSN has no information that would allow the advantages and drawbacks of HTRs to be identified in terms of decommissioning.

3.6 MATURITY OF CONCEPT AND R&D REQUIREMENTS

3.6.1 *OPERATING EXPERIENCE FEEDBACK*

Operating experience feedback from HTRs shows that the two prototypes whose thermal power was close to 1 GW did not prove their industrial viability. On the other hand, smaller reactors operated satisfactorily. A number of technical incidents affected the safety functions at the high-power reactors and in some cases resulted in radioactive releases (THTR). In addition, the Fort Saint Vrain reactor and the THTR did not meet the passive safety criteria set for the latest HTRs and the VHTR.

The HTR prototypes made it possible to test and qualify several types of fuel. As a result, TRISO fuel demonstrated its high safety potential.

Graphite structures

An important point of this feedback relates to the behaviour of the graphite structures in the core. Graphite is a strong, yet relatively brittle material. As noted above (Section 3.2.4), variable internal stresses that are not easily assessed occur in the core structures. An inspection of the internal structures of the AVR during its decommissioning showed that pieces of graphite had come loose from the sections of the bottom reflector adjacent to coolant penetration slits. Cracks also appeared in the blocks, some of which had shifted from their initial positions (Figure 16, left). As a result, around 170 pebbles remained stuck inside the core after it was unloaded.

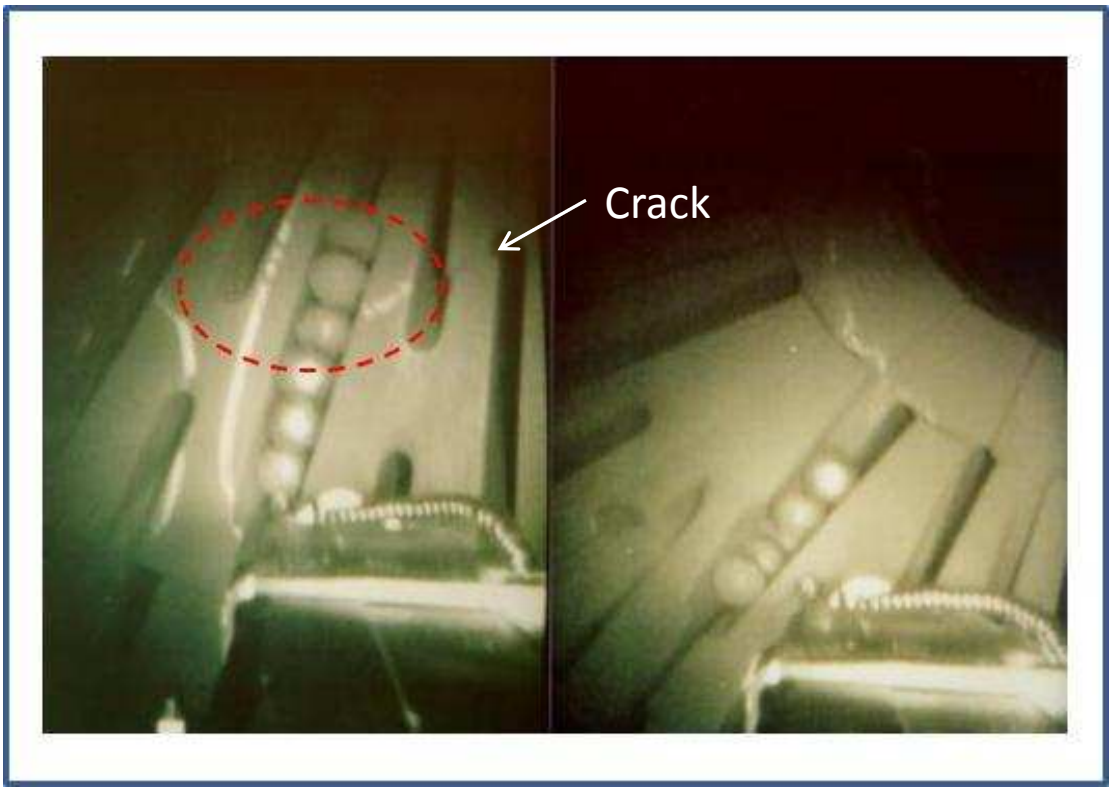

Figure 16: Cracks and pebbles jammed at the bottom of the AVR (reference 42)

These observations show that progress is still needed in the field of design (design rules) and monitoring of the graphite structures.

In conclusion, IRSN considers that the HTR technology is now mature for a maximum helium temperature of around 750°C to 850°C and a limited thermal output power of between 200 MW and 600 MW (prismatic-block reactor). If completed, China's HTR-PM project should make it possible to consolidate the experience acquired in this area by German designers in the 1980s.

3.6.2 *R&D REQUIREMENTS*

Based on the HTR experience, IRSN considers that R&D measures are necessary in order to consolidate the safety demonstration and that technological advances should be made to guarantee the feasibility of the VHTR.

The main developments expected in terms of R&D on safety aspects are thus:

- development of more advanced inspection methods for verifying the quality of fabricated particles;
- development of models for studying the behaviour of particles residing in the core and predicting their failure rate (because the current model is primarily empirical);
- improvement in the definition of the coefficients of diffusion of fission products through the first barrier;
- development of analytical models for assessing the behaviour of fission products released into the helium as well as their interaction with dust;
- source term calculation that takes into account large primary breaks (resuspension and entrainment of dust, behaviour of aerosols, etc.);

- assessment of the production of dust (reference 60) and modelling of their behaviour in the primary circuit and in the event of primary breaks;
- improvement in the models used to describe core behaviour and which involve coupled thermal-hydraulic-neutron calculations.

The Generation IV VHTR design is characterised primarily by a maximum helium temperature of between 950°C[22] and 1000°C and compliance with the objective of passive safety stated for the HTR. The following three technological advances are necessary to achieve these objectives:

- strengthening of the high-temperature resistance of the coating of TRISO particles;
- development of materials that can withstand the corrosion and high temperatures in the steam generator and the control rod systems (reference 59);
- development of robust, effective insulation to protect the parts of the circuit likely to contain hot helium.

3.7 CONCLUSION

The VHTR design has the same safety objectives as the last-generation HTRs, but with higher technical performance levels.

As with the HTR, the safety of the VHTR design is based primarily on the robustness of TRISO fuel. This robustness has been demonstrated under HTR operating conditions by international testing programmes and by operating experience feedback from prototype reactors. The fuel qualification limits during normal operation are 1250°C for the maximum temperature and at least 100 GWd/t for an average burn-up (oxide or oxycarbide fuel). The limit adopted for the maximum fuel temperature during accident situations is around 1600°C. Exceeding this temperature would not lead to a cliff edge effect, at least up to 1800°C. Given the objective of higher operating temperatures for the VHTR (up to 1000°C), this qualification range is insufficient and its extension is currently the subject of R&D.

The feasibility of the VHTR design is also based on the development of high-temperature materials (at least 1000°C) for the equipment on the hot leg of the primary circuit.

The naturally safe behaviour of the VHTR design regarding loss-of-coolant accidents has been demonstrated by the calculation and verified by tests (HTTR). It is thus possible to design a reactor that does not require active decay-heat removal systems. The time-to-rod-drop after an unprotected transient is generally more than several hours and has no significant influence on the maximum temperature reached by the fuel.

Hypothetical accidents of massive ingress of air or water have been identified as the major initiators of possible radioactive releases outside the second barrier. They may induce hazards on the third barrier (production of flammable gases, dust explosion hazard) and releases to the environment. For the VHTR, defence-in-depth against such accidents must be reinforced over that of the HTR by:

- improving their prevention;
- studying situations that are more conservative than those considered in the past (corresponding to current "standards");

[22] Japan's HTTR experimental reactor is designed to reach 950°C. This temperature had to be lowered to 850°C.

- improving understanding and modelling of the phenomena involved in the definition of the source term;
- redefining the confinement objectives and strategy for the third barrier.

Nevertheless, IRSN notes that water or air ingress accidents involve physical phenomena that are easier to model than core melt accidents in a PWR or an SFR. Again, the limited power of the VHTR helps to mitigate the risk associated with graphite oxidation and the robustness of the fuel would appear to eliminate potential cliff edge effects in terms of releases.

The presence of graphite dust could also have a significant impact on worker exposure. Although the operational experience feedback from the HTR reactors is positive, it must be emphasised that the reactors from which this data was obtained have not been operating for long.

Regarding radioactive waste production, the key point is the recycling of core graphite, a technology that has not yet been fully explored (see Section 9.7).

In conclusion, a reactor based on the latest HTRs would be technically feasible by meeting safety objectives that could be markedly higher than those adopted for Generation III reactors, particularly in terms of prevention of core melt, provided the defence-in-depth strategy for water or air ingress accidents is strengthened. The feasibility of a VHTR, which would meet the same safety objectives, must be demonstrated by continuing R&D on TRISO fuel, high-temperature materials and graphite optimisation. However, contrary to the first approach adopted by GIF, studies on the VHTR design have shown that it was not necessary to aim for a maximum helium temperature of 1000°C for the purposes of hydrogen production in particular and of cogeneration in general. This is why current projects are now targeting temperatures below 850°C and building on a mature fuel technology. The fact remains that the safety performance of the HTR or the VHTR can be guaranteed only for a reactor power limited at around 250 MWth (pebble-bed core) to 600 MWth (prismatic-block core).

4. GAS-COOLED FAST REACTORS (GFR)

4.1 HISTORY AND PROSPECTS FOR DEVELOPMENT

4.1.1 *BACKGROUND*

The advantage of the GFR concept is that it harnesses the respective benefits of the fast neutron spectrum and high operating temperatures. The GFR concept was originally based on the SFR and HTR technologies. The goal was to maximise the breeding gain (with minimal neutron capture by the coolant gas), by eliminating the precautions adopted for SFRs relating to the highly reactive nature of sodium, and in particular the need for an intermediate cooling circuit.

No prototype gas-cooled fast reactor has ever been built, although numerous studies have been carried out. In particular, two projects developed between the 1960s and 1980s may be considered as the precursors to the GFR concept: the Gas Cooled Fast Reactor (GCFR) designed by General Atomics in the United States, and the reactors GBR-1 to 4 designed in Europe. These two projects are described below.

GCFR

In 1962, General Atomics designed a prototype reactor with a rated power of 300 MWe as well as a 1000 MWe industrial-scale reactor. The detailed design and certification programme for the 300 MWe prototype (Figure 17) began in 1968, with divergence scheduled for 1983. This GCFR reactor was designed to use fuel consisting of uranium and plutonium, with austenitic stainless steel cladding. Helium entered the reactor core at a temperature of 385°C and exited at 550°C. The helium coolant pressure was 8.5 MPa, and the primary circuit was contained within a reinforced concrete shell. The core of the GCFR was very similar to those of sodium-cooled reactors, apart from the addition of equipment to distribute the helium supply to the base of the fuel assemblies and the adoption of ridged cladding to enhance the heat exchange with the helium.

Development work for the GCFR continued until 1981, when development of fast reactors was abandoned in the United States. Furthermore, it would appear that the results of safety studies conducted at that time were inconclusive (reference 61).

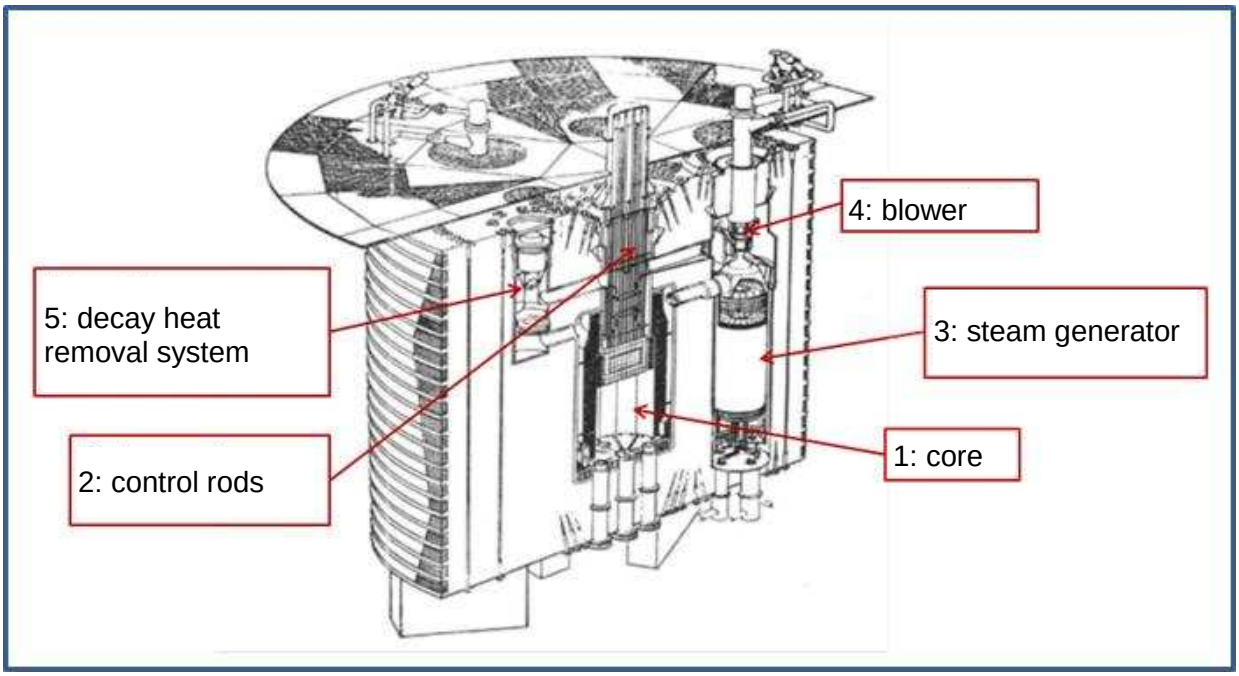

Figure 17: Cross-sectional diagram of the GCFR (reference 62)

GBR

In Europe, a consortium of reactor designers known as the Gas Breeder Reactor Association designed four gas breeder reactor projects between 1970 and 1981: GBR-1 to GBR-4. Figure 18 is a diagram of GBR-4.

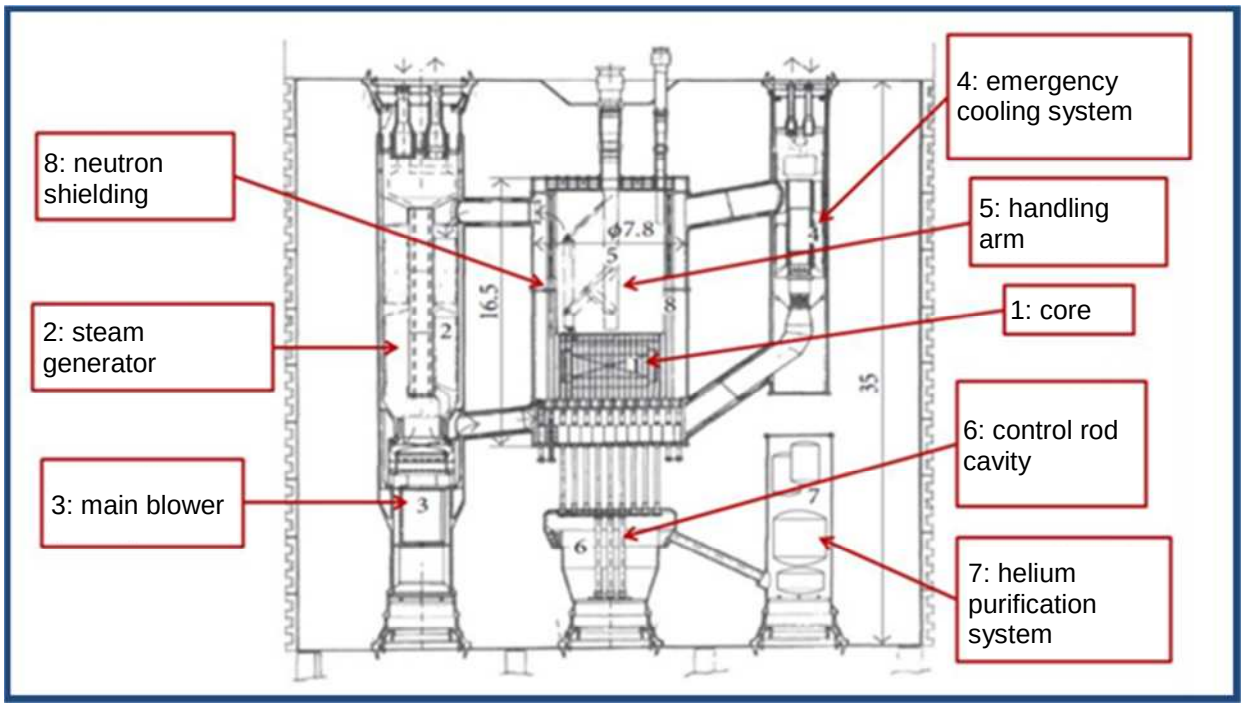

Figure 18: Cross-sectional diagram of the prestressed concrete vessel for the GBR-4 project (dimensions in m) (reference 62)

The fuel envisaged for GBRs was a mixture of plutonium oxide and uranium. These reactors were of a similar design to the General Atomics concept, although the GBR-2 and GBR-3 designs featured innovative assemblies with silicon carbide-coated particle beds, inspired by the HTR. The target core exit temperatures ranged between

550°C (stainless steel-clad fuel) and 700°C (particles). GBR-4 was designed with a rated power of 3540 MWth (reference 62).

Like the General Atomics GCFR, development work for the GBR project was halted in the early 1980s.

Lessons learned

The lessons learned from these projects showed that the potential savings relative to a sodium-cooled system (due to the elimination of the intermediate cooling circuit and a superior neutron balance) were partially negated by the complex design of the decay heat removal system. This was because the nature of helium (see Section 3.2.3) makes the reactor difficult to cool, particularly in the event of accidental depressurisation of the primary circuit, due to the limited thermal inertia of the coolant. Furthermore, adopting the cladding materials already used in the SFR resulted in a proven fuel recycling technology but decreased the safety margins in the event of a loss of coolant. As a result, the operating temperatures could not be higher than with sodium unless a particle bed concept was adopted.

4.1.2 *ONGOING PROJECTS AND PROSPECTS FOR DEVELOPMENT*

The GFR experienced a renaissance in the early 2000s, through the work of GIF. The renewed interest is attributable to the availability of refractory materials suitable for use in fuel cladding (including silicon carbide and other metal alloys arising out of R&D efforts for the ITER project). A goal of operating in a temperature range similar to the HTR (i.e. >700°C at core exit) was set. Furthermore, the GFR concept, like the other fast reactors, benefited from renewed interest in the light of new concerns relating to management of uranium resources.

Until recently, CEA was the main promoter of the GFR, studying a project for an experimental reactor (ALLEGRO) and an industrial-scale concept (GFR 2400). Europe supported the GFR concept by funding two successive R&D projects named GCFR[23] and GoFastR (completed in 2012). The ALLEGRO project (reference 65) launched in 2010 as part of the ESNII initiative aims to build a low-power (approx. 70 MWth) reactor, resuming development of the experimental reactor of the same name originally studied by CEA. This project is backed by Poland, the Czech Republic, Hungary and Slovakia. The ALLEGRO reactor would use SFR-type fuel assemblies with stainless steel cladding, operating at a maximum temperature of 530°C. This reactor would be used to test assemblies with refractory cladding as well as technologies suitable for a helium-based primary circuit (blowers, heat exchangers, purification system, etc.).

4.2 MAIN CHARACTERISTICS

4.2.1 *GENERAL DESIGN*

The GFR concept described here is based on a project developed by CEA until 2009 (reference 63), as this is currently the most representative model of an industrial gas-cooled fast reactor.

[23] The same abbreviation as for the reactor designed by General Atomics.

The fuel assemblies used in the GFR core are similar to those used in the SFR (cylindrical cladding and hexagonal tubes). The main innovation compared with earlier concepts is the use of cladding made of refractory materials.

Figure 19 and Figure 20 respectively show a cross-section of the reactor vessel and of the GFR reference diagram. The reactor features a pressurised main reactor vessel and a second internal vessel containing the core. Three loops provide reactor cooling in normal operation. Each cooling system includes a primary loop through which helium is circulated by a blower. When the helium in the reactor primary loop has passed through the core, it is cooled in an Intermediate Heat eXchanger (IHX) connected to an intermediate gas system. Part of the heat transferred to the intermediate system is converted into electricity via a gas turbine (in a Joule or Brayton cycle), and the remainder is used in a steam generator connected to a tertiary system. This tertiary system also generates electricity using a Rankine cycle. The overall thermodynamic efficiency is in the region of 45%.

The main reactor vessel is pressurised and maintained at a temperature of approximately 400°C in normal operation (by the same principle as for the SFR or VHTR) by helium exiting the IHXs (cold leg), which passes through the inter-vessel gap before entering the core. The internal vessel is protected by thermal insulation on the hot helium-facing surfaces; the same applies to the interior surfaces of the coaxial pipes (or "cross-ducts") that connect the main reactor vessel with the vessels containing the intermediate heat exchangers.

Four Decay Heat Removal (DHR) systems remove decay heat from the core in normal operation and accident conditions. The DHR systems are described in detail in Section 4.3.2.

Lastly, all the systems through which helium flows are contained inside an 11,600 m^3 "close containment" filled with nitrogen at atmospheric pressure. In the event of a leak from the primary circuit, the close containment maintains a back-up helium pressure of approximately 0.5 MPa inside the reactor. The back-up pressure ensures that effective core cooling can be maintained in the event of a primary circuit leak with no need to use high-power blowers[24].

[24] Cooling effectiveness depends on the mass flow rate of the helium injected into the core. At constant blower speed, mass flow rate increases proportionally to helium density and therefore pressure.

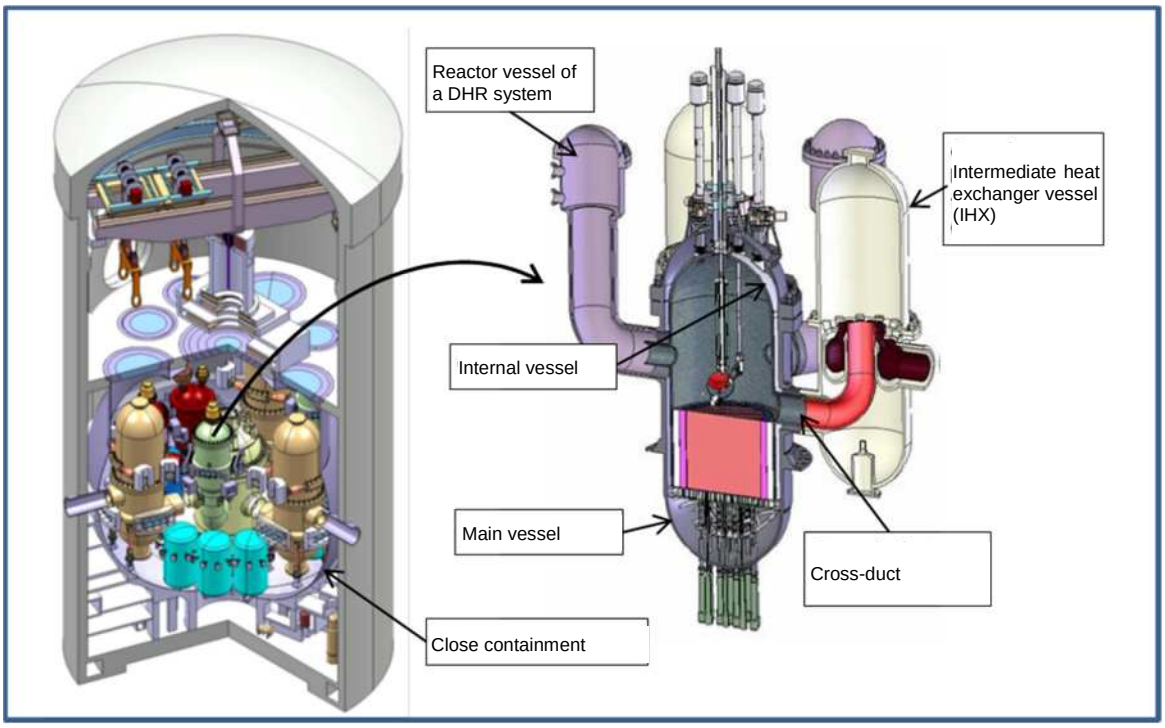

Figure 19: Reactor building and vertical cross-section of the GFR reactor vessel (reference 63)

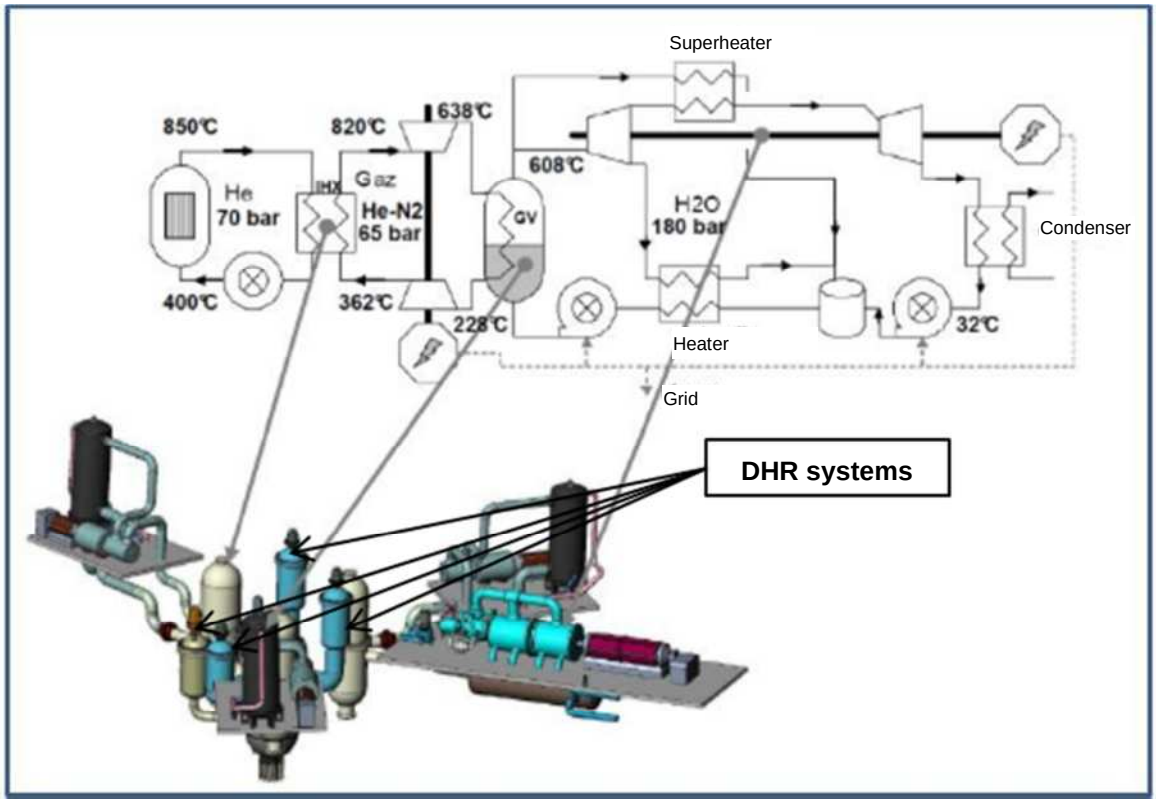

Figure 20: Overview of the GFR with the combined direct cycle (reference 63)

The main characteristics of the reference GFR concept studied via the GoFastR project are described in Table 1.

Table 1: Characteristics of the GFR concept (core version developed in March 2009 – reference 63)

Core	
Thermal power	2400 MWth
Core power density	100 MW/m^3
Fissile core diameter (equivalent)	4.23 m
Fissile core height	1.65 m
Number of fuel assemblies	516
Fuel type	Pellets (U,Pu)C in pins
Mean Pu enrichment at equilibrium	16.3%
Plutonium inventory at equilibrium	10.8 t
Neutron characteristics	
Fraction of delayed neutrons at equilibrium, at end-of-life (ß)	360 pcm (= 1$)
Doppler constant at equilibrium, at end-of-life between 180°C and 990°C	860 pcm
Helium void effect at equilibrium, at end-of-life (7 MPa -> 0.1 MPa)	0.89 $
Reactor characteristics	
Mean core outlet helium temperature	850°C¤
Mean core inlet helium temperature	400°C
Coolant helium pressure	7 MPa
Number of intermediate loops	3

4.2.2 *GENERAL SAFETY OPTIONS*

The GFR is notable for the low efficiency of its coolant fluid and the low thermal inertia of the primary circuit. Appropriate design measures must therefore be adopted to decrease the probability of loss of primary helium to an acceptable level. Like other fast-spectrum reactors, the GFR is sensitive to the unintentional introduction of moderator into the core and the coolant void effect is positive, albeit less than β, the delayed neutron fraction. At the current stage of development of the GFR, the main safety options being considered for the purpose of performing the safety functions are as follows:

- Develop a core made of refractory materials that remain intact in the event of loss-of-normal-cooling transients.

- Design decay heat removal (DHR) systems capable of operating by natural convection when the coolant helium pressure is sufficient.
- In the event that the primary circuit is depressurised, use active decay heat removal means for the first 24 hours, then natural convection.
- Ensure a minimum back-up pressure in the event of a leak from the primary circuit by implementing a close containment and bypass isolation devices on systems that pass through it.
- Preclude by design any risk of control rods ejection from the core.
- Minimise the risk of air or water ingress by design.

When the primary circuit pressure is high enough, harnessing natural convection is the preferred solution, with helium circulating upwards in normal operation, minimising the hydraulic head loss through the core.

IRSN notes that options to ensure the mechanical stability of the core have not been addressed (compaction risk).

4.2.3 *GFR FUEL*

The biggest R&D challenge relating to the GFR core concerns the development of fuel assembly and fuel cladding materials that are able to withstand high temperatures, in order to provide adequate margins with regard to core damage in the event of an accident.

The GFR reference reactor core (Figure 21) features a hexagonal geometry. Each fuel assembly consists of a hexagonal tube containing a bundle of fuel pins arranged in a triangular pitch. The assembly includes two neutron reflectors, positioned to either side of the fuel pin bundle. This geometry reflects the need for fast reactor cores to be compact and for gas flows to be individualised.

The fuel consists of pellets of mixed uranium and plutonium carbide ((U,Pu)C). As helium is a less effective heat transfer fluid than liquid metals, it must occupy a proportionally larger fraction of the reactor core volume, to the detriment of the fuel. The high density of carbide partially offsets this disadvantage. The reactor core has two different enrichment areas, known as the inner core and the outer core.

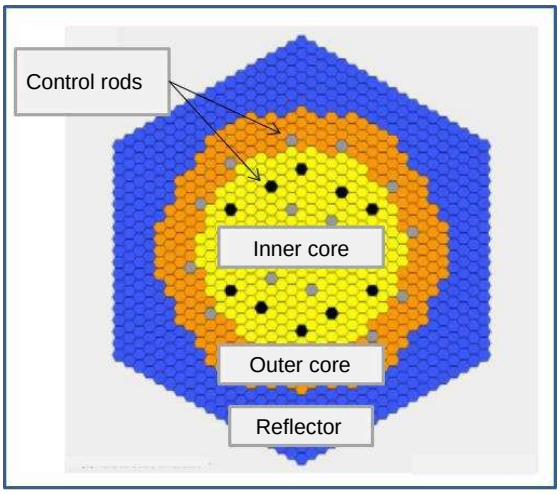

Figure 21: Horizontal cross-section of the GFR core (reference 63)

In the GFR concept, the core must be able to withstand temperatures above 1600°C, or even 2000°C for several minutes. In practice, the choice of materials is fairly restricted, due to irradiation-related constraints. Silicon carbide (SiC) was considered, in particular for use in the first barrier. However, this brittle material must be reinforced if it is to be used in cylindrical cladding. An all-silicon carbide fibre-reinforced composite material was ultimately adopted (SiC/SiC_f composite). Figure 22 shows the most recent steps in the development of a ceramic cladding solution. The cylindrical cladding is made from woven silicon carbide fibre that gives the cladding mechanical strength and a degree of "ductility"[25]. This fabric is then impregnated with silicon carbide. The resulting SiC/SiC_f composite does not provide a sufficiently tight barrier to fission products. This function is performed by a metal liner, in this case made of tantalum. A carbon "buffer" provides the cladding with a degree of protection against fission products (silicon carbide may decompose under the effect of metallic oxides) and limits mechanical interactions as the pellet swells.

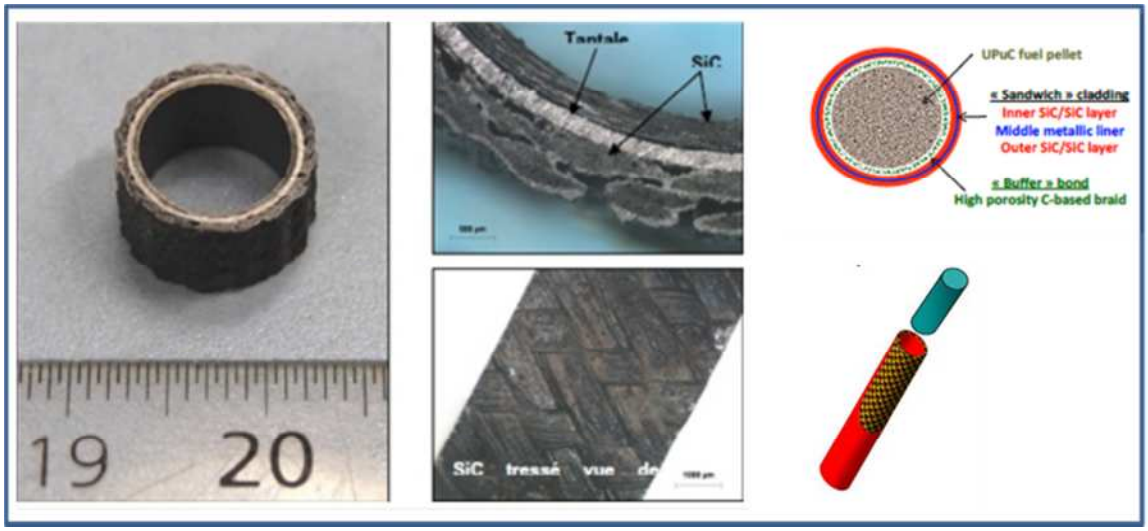

Figure 22: Sample SiC/SiCf sandwich cladding, CEA patent, M. Zabiego – GoFastR Project – 2012

The planned fissile assembly would be a composite assembly made up of metallic and ceramic elements. It is not currently possible to produce a ceramic fuel pin as a single piece extending the full height of the fissile column; the fuel bundle may be made of half-pins mounted one above the other.

Note that the fuel for the GFR is still at the R&D phase. CEA has carried out experiments to assess cladding compatibility with carbide fuel and has conducted tests to define the mechanical specifications of the non-irradiated cladding (information collected as part of the GoFastR project). The performance of the fuel has not yet been tested in a reactor. However, it should be noted that a few recent experiments have been carried out, focusing on irradiation of "solid" silicon carbide and SiC/SiC_f composites. For example, a series of irradiation tests was recently completed by ORNL (a US-Japanese programme); the purpose of these tests was to measure any changes in the pre- and post-irradiation properties of 15 types of SiC/SiC_f composite, all essentially consisting of SiC_β[26] specially developed for nuclear applications relating to PWR cladding, high-temperature reactors and nuclear fusion (reference 66). These tests revealed that the only physical characteristic significantly affected by the

[25] Before it is irradiated, the SiC/SiC_f composite is still a brittle material compared with stainless steel cladding, as its elongation is between 0.1% and 1% (when subjected to tensile loading in the longitudinal direction of the fibres – reference 66), compared with more than 10% in the case of stainless steel.

[26] The beta variety has the same crystalline form as diamond.

irradiation recorded in the reference 66 is material heat conductivity, which decreases significantly (although the actual values are very dispersed, depending on the type of composite). Swelling appears to saturate rapidly around 2%, up to 70 dpa at least (60 dpa envisaged for GFR cladding). Concerning the fuel, the FUTURIX-MI irradiation experiment performed at the Phenix[27] reactor was also expected to yield data relating to matrix behaviour. Lastly, the behaviour of the (U, Pu)C fuel has been researched in the past and a certain volume of physical data and irradiation experiment-related results are available (see Appendix III of this report).

Note that vanadium alloy cladding may be feasible for reactors operating at lower temperatures than those in the project studied by CEA.

In conclusion, the following points should be noted:

- Silicon carbide can be used to produce cladding that in theory would remain intact at temperatures far higher than in normal operation.
- Fibre-reinforced composites are tougher than solid carbide but are not leaktight.
- The carbide fuel must be isolated from the silicon carbide.
- The effects of irradiation on cladding decrease its thermal conductivity and cause moderate swelling, whereas the other physical characteristics appear to be largely unaffected at doses up to those envisaged for a GFR.

Although the development of SiC/SiC_f composite cladding appears promising for high-temperature concepts in general, and even for PWRs, a number of qualification stages remain necessary, to prove that they are compatible with the operating conditions encountered in the core of a GFR.

4.2.4 *COOLANT*

The properties of the coolant are described in Chapter 3.2.3, in the context of the VHTR concept.

Unlike with the VHTR, for safety reasons it is essential to prevent a loss of helium inventory in the primary circuit, as the GFR core must be cooled by convection. The nature of helium makes it hard to keep this system leaktight. The same applies to the close containment, in the event of a leak from the primary circuit.

4.2.5 *BARRIERS*

Radioactive materials in the reactor core are confined by three successive, independent barriers: the fuel cladding, the primary helium circuit envelope and the reactor building. The designs of the GFR project did not assign a barrier role to the close containment.

[27] This experiment was completed in 2009.

4.2.5.1 First barrier

Fuel cladding would reach temperatures approaching 1000°C in normal operation. Furthermore, the core and DHR systems are designed to prevent the temperature from exceeding 1600°C during most accident transients. For the preliminary studies, this limit was adopted due to the similarities with TRISO fuel. For some severe primary circuit depressurisation accidents (involving large or multiple breaks), the designers considered that the cladding may reach a maximum temperature of 2000°C for around 10 minutes and remain intact. In any case, the aforementioned temperature limits are merely exploratory values. They are not treated as safety criteria for the GFR fuel. The scope for comparison with the VHTR fuel is very limited, as the chemical environment of silicon carbide is very different with the GFR, due to its direct contact with helium and water vapour or air in some accident conditions, the presence of a metal liner, etc.

Notwithstanding the uncertainties relating to cladding performance (see previous section), IRSN notes the following points, which are critical for safety:

- Analysing hot spots is more complicated for a gaseous coolant than a liquid coolant (due to more difficult mixing between channels, the effects of compressibility, etc.).
- Analysing the influence of any pre-existing defects in a composite material would appear to be more complex than for metallic cladding (due to anisotropy).
- In-core cladding failure would be harder to detect than in the SFR.

The safety issues posed by the latter point are different in the GFR and SFR because helium does not react with the fuel. However, there is a risk of contamination of the cold parts of the primary circuit by volatile fission products left by the purification system. This could in turn be problematic for maintenance operations or in the event of a helium leak entraining such contaminants into the reactor containment building.

The goal of preserving the mechanical integrity of the cladding in all incident and accident conditions appears to be harder to achieve than with the VHTR, in part due to the intended operating conditions. In this respect, it would be useful to explore the effect of decreasing the reactor core outlet temperature on the estimated "margins" adopted at the preliminary design stage. In addition, IRSN notes that adequate justification for the GFR fuel cladding temperature limit of 1600°C has not yet been provided.

4.2.5.2 Second barrier

As shown in Figure 19, the second barrier of the GFR is relatively complex, consisting of the external envelope of the primary circuit (reactor vessel, cross-ducts, IHX vessels, DHR, etc.) and the IHX tubes. In addition, an isolatable helium purification system is connected to the primary circuit.

The material expected to be used for the second barrier, except for the high-temperature surfaces, is X10CrMoVNb9-1 (P91) martensitic steel, as used in thermal power plants. The material for the intermediate heat exchanger, which operates at 850°C, has not yet been defined (although nickel alloys are under consideration).

Regarding monitoring of the second barrier, IRSN notes that small helium leaks would be difficult to detect and locate (specific technology would need to be developed). Furthermore, the intermediate heat exchangers would include hard-to-monitor plate-type modules. In the event of a leak from the intermediate heat exchangers, helium coolant would enter the intermediate system, which operates at a lower pressure. Accordingly, a leak detection

(activity measurement) system and isolating valves would be installed on the pipes in the intermediate system, as close as possible to the heat exchanger vessels.

In accident operation, there is a potential risk (as there is with the VHTR) of hot gas entering the cold leg. However, as heat exchanges between the internals and helium are less effective than with liquid metals, the thermal shocks impacting the vessel and internals during operating transients should be less severe than with the SFR or LFR design. In any case, the risks of the second barrier and reactor core supports being subjected to aggression by hot gases (at 800°C) will need to be thoroughly examined.

The intermediate heat exchanger tubes are a weak point in the second barrier, owing to their severe operating conditions and their necessarily large heat exchanging surface area. Valves coupled with a leak monitoring system should enable the loop to be isolated if the barrier were to be bypassed at the heat exchangers.

4.2.5.3 Third barrier

IRSN does not have any information about the reactor building envisaged for a GFR. In particular, the role of the reactor building with regard to the confinement function in the event of a primary circuit leak or core melt event has not been clearly defined. Note that the close containment does not form a barrier, but acts a buffer volume between the primary circuit and the reactor building: its leakage rate must be such that the back-up pressure can be maintained for a few tens of hours (depending on the decay heat decrease rate).

4.2.6 *FUEL ASSEMBLY HANDLING AND STORAGE*

As the reactor vessel cannot be opened, the fuel loading and unloading system would use a pressurised airlock. CEA considered using an articulated arm (Figure 23) to remove fuel assemblies from the core and place them in a fuel assembly container. The other steps in the handling sequence remain to be defined.

Despite the lack of available information, IRSN notes the following few safety-related considerations:

- Cooling irradiated fuel assemblies during handling is likely to be a technical challenge: despite their ceramic cladding being designed to withstand high temperatures they will probably require active cooling (like the cladding for the SFR).
- Unlike for the SFR, irradiated assemblies are unlikely to require washing before being stored, thereby reducing the risks of potential thermal shocks involving the first barrier and decreasing liquid waste volumes[28].
- As with the SFR, it will be necessary to examine the risk of clad ruptures due to a fuel assembly falling

[28] A significant proportion of the liquid waste for the SFR is generated during fuel assembly and reactor component washing and decontamination operations.

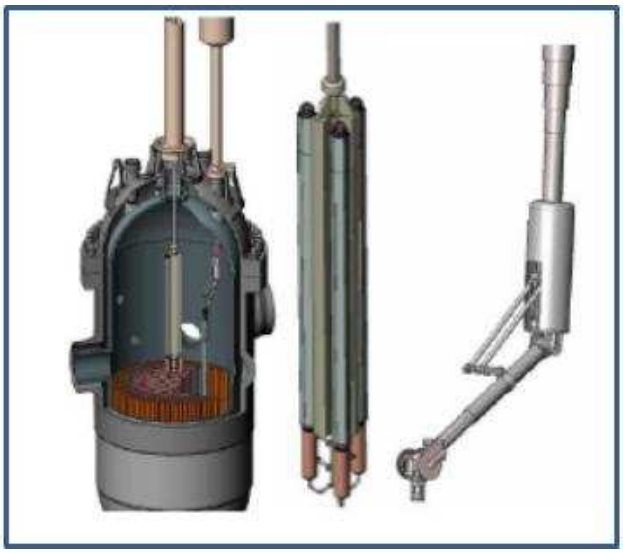

Figure 23: Handling arm and transfer cask (reference 64)

4.3 CONTROL OF SAFETY FUNCTIONS

4.3.1 *REACTIVITY CONTROL*

Reactivity is controlled by neutron feedback and by a system that introduces neutron absorbers.

The nature of the spectrum influences the neutron feedback coefficients. Compared with the SFR and LFR concepts, the neutron energy spectrum in the GFR core may favour the higher energy levels, as helium interacts less with neutrons than sodium or lead (from a neutron slowing-down and capture perspective). This could adversely impact some neutron feedback coefficients. However, the reactor cores envisaged for the GFR would contain a non-negligible quantity of metals (tantalum, niobium, tungsten or vanadium) which have a moderating effect, as does the silicon carbide in the cladding and hexagonal tubes in the fuel assemblies. Consequently, regarding the effect of neutron feedback, there do not currently appear to be any significant differences between the GFR spectrum and those of the other fast neutron concepts. On the other hand, the weak interaction between helium and neutrons would limit the increase in reactivity in the event of primary circuit depressurisation (void effect) without imposing major core design constraints, unlike in the case of the SFR. The GFR designers have assumed a void effect of the same order of magnitude as the proportion of delayed neutrons (approx. +320 pcm at equilibrium at end of life – GoFastR studies).

The shutdown system (control rods) is an essential safety feature in the GFR. Initial studies for the GFR concept show, for example, that it would be necessary to rapidly shut down the nuclear reaction in the event of loss of normal cooling or depressurisation. It should also be noted that an earthquake signal-triggered automatic shutdown function appears to be essential, as the reactor core is vulnerable to strain in the fuel pin lattice, all the more so as the coolant does not play a damping role as it does in the SFR.

Two redundant and diversified systems would be used to manage and control reactivity. In the reference design, only one of the two systems (the control and shutdown system) is used for control operations, the other (the diverse shutdown system) being used only for automatic shutdowns.

The rod mechanisms are located underneath the reactor vessel, and the control rods above the fissile core (see Figure 19). This configuration ensures that most of the mechanical systems are positioned in the area through which cold helium flows. It also enables rod drop blocking systems to be used in the event of the mechanism being ejected, the aim being to ensure that the control rods remain in the fissile part of the core. It is also possible to design rods so that they are heavy enough to limit the risk of ejection.

Lastly, it should be noted that, as with the other fast neutron concepts, the GFR core reactivity control function is influenced by the related fuel management objective (depending on the options adopted with regard to minor actinide transmutation, plutonium breeding, plutonium consumption, etc.).

4.3.2 *DECAY HEAT REMOVAL*

Despite the limited power density[29] of 100 MW/m^3, devising an appropriate decay heat removal strategy for normal and accident operation is a major challenge for the GFR concept, due to the low thermal inertia of the coolant. Decay heat removal systems must be suitable for several sets of operating conditions characterised by three pressure ranges. This resulted in a relatively complex system architecture from the outset of the GFR project. The cooling systems shown in Figure 24 feature three high pressure (3x100%) decay heat removal (DHR) loops and one low-pressure loop. The low-pressure DHR loop requires a powerful blower (possible option: turbo-blower).

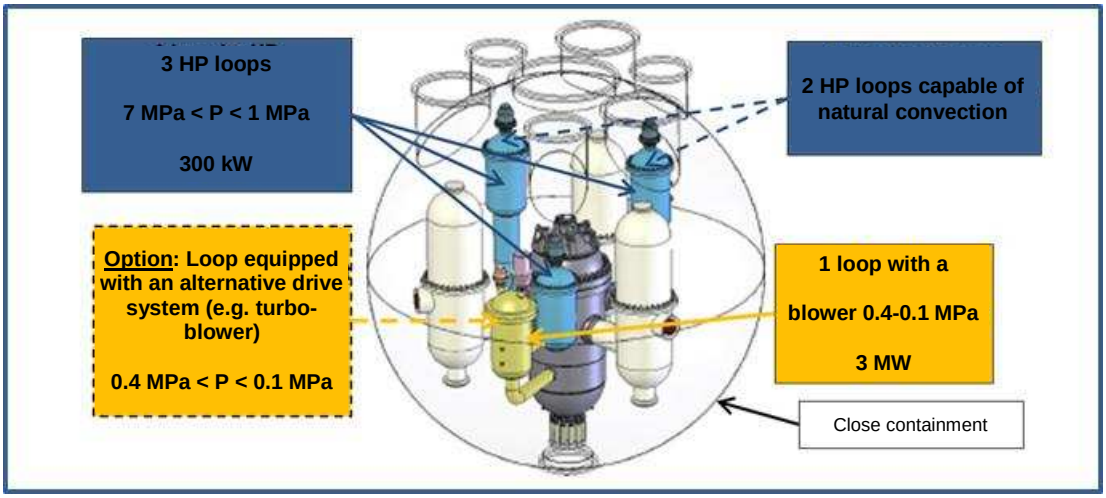

Figure 24: Decay heat removal – 2007 configuration (reference 67)

Figure 25 shows the operating principle of a high-pressure DHR loop. The DHR system would feature a reactor coolant loop carrying helium, which would be cooled in a heat exchanger by water from the secondary cooling circuit (at a pressure of 1 MPa). The water in the secondary loop would be cooled by a heat-exchanger immersed in a water pool. The pool, situated inside the reactor containment, would be cooled by a tertiary system (not shown in Figure 25) equipped with an air/water heat exchanger installed outside the containment. The mass of

[29] A lower power density (of around 50 MW/m^3) would in theory provide more satisfactory safety margins in conditions involving depressurisation of the primary circuit, but this would increase the cost of the fuel cycle (as a greater plutonium inventory would be required).

water in the pool would be sufficient to enable the DHR systems to operate autonomously for 24 hours without cooling by outdoor air.

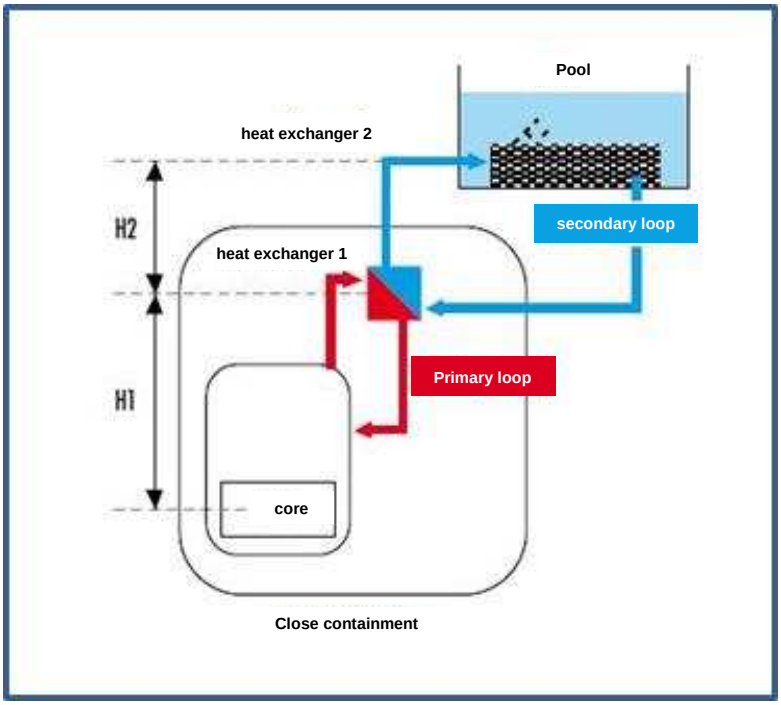

Figure 25: Overview of a high-pressure DHR loop (reference 63)

IRSN has not been provided with a description of the low-pressure DHR loop. CEA has noted that it should be equipped with a powerful blower.

The following three categories of conditions were studied with a view to validating the DHR design basis:

- loss of cooling with no reduction in the mass of helium in the primary circuit;
- depressurisation of the primary circuit with no close containment break;
- depressurisation of the primary circuit with close containment break.

In order to manage the first two scenarios, the three high pressure DHR systems would include speed-controlled blowers designed to operate efficiently at between 7 MPa and 1 MPa and fitted with backup electric power supplies. Two of these three systems are designed to allow operation in natural convection mode:

- immediately after an emergency shutdown if the helium inventory is maintained;
- after 24 hours in the event of a break in the primary circuit.

In the third scenario, decay heat removal would be provided by the “low-pressure” DHR system.

In all cases, when the DHR systems are started, the circuits used in normal operation must be isolated to prevent core bypass.

As well as a deterministic study of the various reactor cooling degradation transients, the designers conducted probabilistic studies to assess the suitability of the chosen system architecture with DHR heat exchanger reliability targets. The DHR heat exchanger system was modified in the light of the results obtained, in particular to reduce the risks associated with frequent shutdown conditions (risk of failure of the DHR loops to start) and scenarios involving close containment failures. These changes are described in Figure 26.

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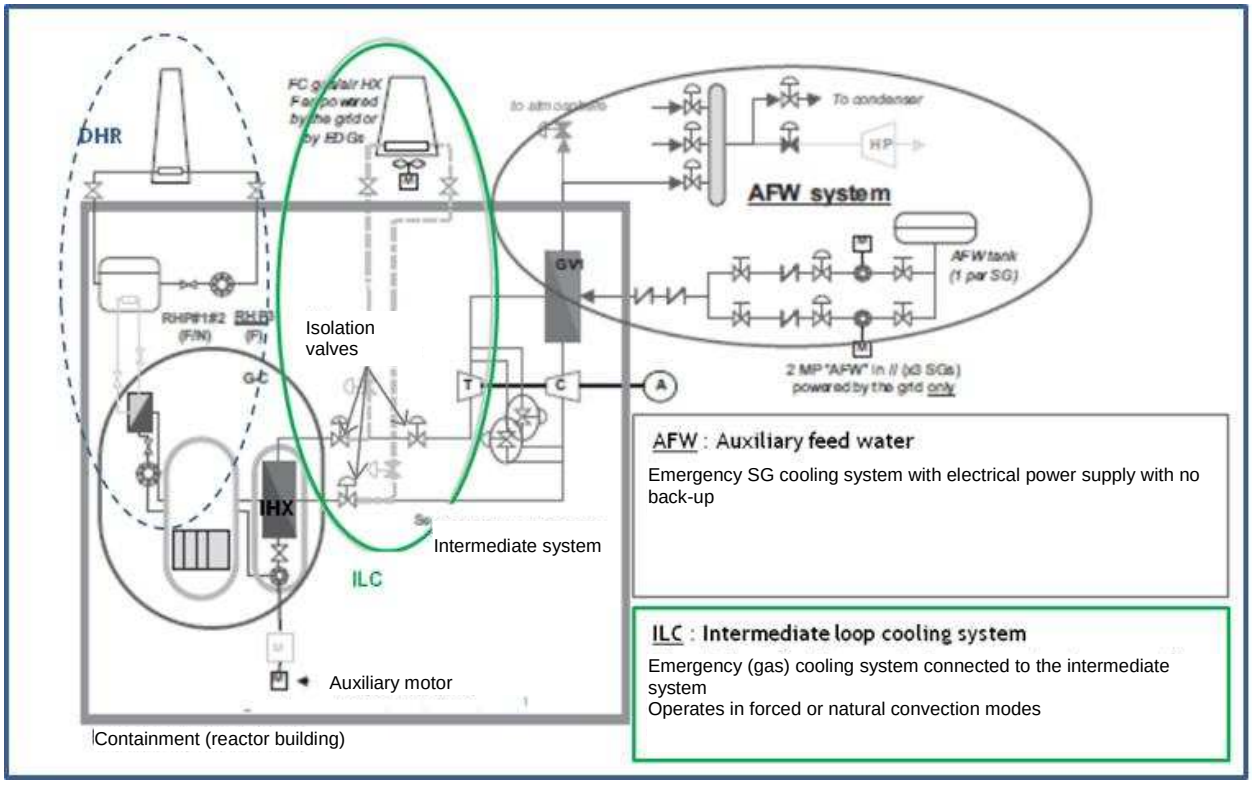

Figure 26: Changes to the GFR DHR heat exchanger system (reference 67)

During a pressurised reactor shutdown, coolant helium would be circulated by the primary circuit blowers, equipped with emergency-supplied auxiliary motors. The core would be cooled by at least one of the three normal circuits, each equipped with a specific cooling system (AFW system). In case of unavailability of the AFW systems, decay heat would be removed by a system connected to the intermediate circuit featuring a gas/air heat exchanger (ILC system). Only the active cooling system of the ILC heat exchanger would be emergency-supplied.

In case of unavailability of the normal systems, the aforementioned emergency DHR cooling systems would perform the decay heat removal function. In this configuration, the normal cooling systems are isolated. The DHR systems are designed such that gas does not need to be injected into the primary circuit. Compared with Figure 24, the number of high-pressure DHR systems would be reduced to two, and there would be two low-pressure DHRs rather than one.

Other changes intended to enhance the efficiency of the decay heat removal function were considered, such as injecting nitrogen into the primary circuit from tanks located inside the close containment. This measure may enable minor leaks from the primary circuit to be managed with no need for active systems. However, the consequences of nitrogen injection on core reactivity and corrosion of internal structures have not yet been examined.

Furthermore, an alternative solution consisting in mechanically coupling the primary circuit blower to the intermediate system turbine and compressor (Figure 27) is under consideration. With this solution, coolant would be circulated with no need for an independent electric power supply (reference 69).

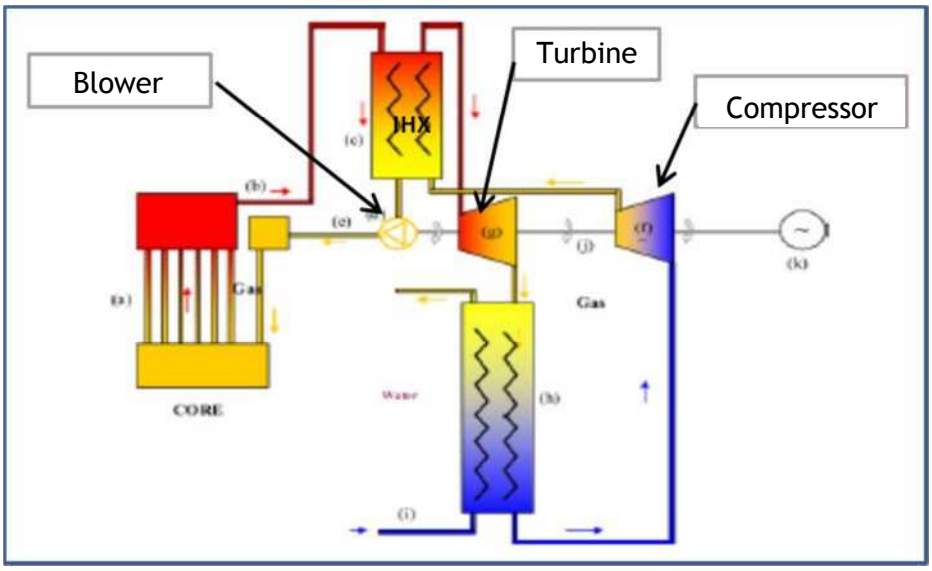

Figure 27: Conversion cycle with coupled primary circuit blower (reference 69)

IRSN comments on the heat removal function

The GFR decay heat removal strategy has not yet been finalised. However, IRSN notes that it uses hard-to-diversify equipment requiring multiple support, control and monitoring systems (valves, pumps, blower control system, gas flow measurements, etc.). As a result, the planned changes to the DHR heat exchanger system architecture, aimed at enhancing its theoretical reliability, make the original arrangement more complex – with the addition of the ILC, AFW, auxiliary motors for the primary circuit blowers, etc., thereby increasing the uncertainties relating to the actual gain in reliability. In this respect, it should be noted that unlike with the SFR or LFR, the DHR systems must be isolated in normal operation to prevent core bypass (valves must be opened when the DHR systems are started). Furthermore, demonstrating the ability of the high-pressure DHR systems to operate early enough in natural convection mode would probably be a problematic facet of the safety demonstration.

A risk of core bypass also exists in the event of a ruptured cross-duct, where the cold leg of the helium system communicates with the close containment. In such a case, part of the cold helium injected by the DHR systems flows into the containment rather than passing through the reactor core (due to higher pressure losses in the core). This accident is a core meltdown initiating event.

Concerning the assurance of maintaining a back-up pressure in the primary circuit, it should be noted that multiple systems pass through the close containment. In addition, this containment will have to include penetrations affording access to the primary circuit (hatch for core refuelling, component maintenance, etc.). Such penetrations also represent potential leakage routes. Lastly, in the event of aggression, the leaktightness of this

containment may be compromised and its leakage rate increased with no possibility of detecting the source of leaks (raising repairability issues).

In conclusion, IRSN considers that the low thermal inertia of the GFR primary circuit requires a rapid, appropriate response by the shutdown system and emergency cooling systems in the event of a cooling fault. As a result of this weakness of the concept, the present design of the primary circuit and reactor core as well as the architecture of the safeguard systems should be reworked at some stage in the future. From this perspective, and considering the objective of enhancing safety compared with third-generation reactors, IRSN considers that the feasibility of the GFR concept has yet to be demonstrated.

4.3.3 *CONFINEMENT OF RADIOACTIVE MATERIALS*

The confinement function of the first barrier is performed by the metal liner sandwiched between the two SiC/SiC_f composite tubes.

The GFR designers have set the target of reducing the leakage rate from the reactor primary circuit to 10% of its helium mass per year. IRSN compared this value against the leakage rate observed on HTRs, and notes that this target is very ambitious, given the larger volume of the GFR primary containment[30]. In any case, a continuous leak of coolant helium during normal operation would disperse contamination outside the primary circuit. The presence of a close containment may help to limit such contamination, but its role for this function remains to be defined (leaktightness requirement, ventilation, etc.). In any case, the impact of primary circuit leaks during normal operation remains to be studied.

Regarding the study of accidents liable to involve radiological releases, the short-term consequences of primary circuit depressurisation accidents should be limited first by the helium purification system and then by the presence of the close containment, which would delay and mitigate such releases, even if it is not explicitly assigned a barrier role. In its capacity as a barrier, the reactor building should be designed to withstand the pressures and temperatures entailed by the various depressurisation accidents (no data available at the time of publication).

Lastly, the reliability with which the various systems connected to the coolant helium envelope can be isolated will play an important role in the confinement function. Such systems (helium purification, DHR and intermediate gas systems) should, however, be located in the reactor building.

4.4 RISK ANALYSIS

4.4.1 *RISKS SPECIFIC TO THE CONCEPT*

Due to the use of pressurised coolant, a large break in the primary circuit might damage the reactor core and internals. Such an accident would result in sudden depressurisation, within no more than a few tens of seconds. If the break were to occur sufficiently suddenly, a shock wave would be generated and travel through the primary circuit, exerting forces on the core support structures and fuel assemblies. Simply maintaining a back-up pressure does not offer adequate provision to ensure the DHR function. The core must also remain in a geometric state that enables it to be cooled by circulating helium. A shock wave in the primary circuit may damage the fuel assemblies

[30] The HTTR has a leakage rate of approximately 75% of the helium mass per year at 4 MPa (JAEA data – LOFC project).

and create blockages. The thermal protection would also be subjected to high loads (tear-off effect). Furthermore, the shock to the core would almost immediately cause power oscillations.

4.4.2 *ACCIDENT SEQUENCES*

4.4.2.1 Category 3 accidents

No safety demonstration has yet been undertaken for the GFR concept currently under development, but numerous preliminary studies of accident transients have been conducted as part of the GoFastR project. The principal accidents envisaged on the basis of the 2007 version of the GFR concept (reference 67) are discussed below.

The third category of accidents includes mainly a small break (diameter less than three inches) in the primary circuit, a leak from an intermediate heat exchanger and loss of off-site power lasting longer than two hours. For such accidents, the safety criteria envisaged by the designer relate to the maximum temperatures of the cladding (1450°C) and the upper plenum (1250°C). For the pessimistic case of a three-inch break with a failed DHR system, the maximum cladding temperature was calculated to be 1180°C and the maximum hot plenum temperature 886°C (reference 67).

4.4.2.2 Category 4 accidents

Regarding category 4 accidents, the considered criteria are of the same type as for category 3 accidents, but with a higher maximum acceptable fuel temperature of 1600°C. The reference accident is a 10-inch break aggravated by the loss of a DHR system, resulting in rapid depressurisation of the primary circuit (reactor shutdown, blower shutdown within 60 seconds, normal cooling systems and close containment isolated). Approximately 46 seconds after reactor shutdown, the blowers in the high-pressure DHR systems (two of the three DHR systems) start up, creating a helium flow rate of approximately 30 kg/s through the core (compared with 1000 kg/s in nominal operation). In such conditions, the maximum cladding and hot plenum temperatures are evaluated at 1470°C and 1160°C, respectively. The accident case involving a large break at an intermediate heat exchanger is less pessimistic.

In the event of loss of emergency-supplied power to the DHR blowers, the cooling capacity with 2 DHR systems (2007 design) operating in natural convection mode was assessed, but the category 4 criteria cannot be satisfied unless the pressure in the primary circuit exceeds 1.5 MPa (i.e. higher than the back-up pressure). According to the designers' initial studies, injecting nitrogen may enable compliance with the aforementioned criteria, probably at the cost of loss of the investment (due to nitriding of the reactor core and internals).

The above scenario would also cover the following category 4 transients:

- break in the secondary cooling system;
- pipe rupture by an intermediate heat exchanger;
- break between the hot leg and cold leg (bypass).

The designers also defined design extension conditions covering accidents considered improbable as well as a number of complex sequences involving multiple failures, which were studied without considering aggravating factors. These sequences are listed below:

- guillotine break of a cross-duct;
- water ingress into the primary circuit
- total loss of off-site power and emergency power (diesel generators);
- rupture in an intermediate heat exchanger combined with a loss of electric power;
- concomitant loss of leaktightness in the primary circuit and close containment;
- loss of blowers not protected by the shutdown system;
- unprotected power excursion;
- unprotected loss of coolant.

For these sequences, the maximum cladding temperature is the same as for category 4, but a temperature below 1850°C would be tolerated for approximately 10 minutes. No criterion relating to the structures surrounding the core has been defined. For some complex sequences, such as an intermediate heat exchanger break combined with total loss of electric power, the designers envisage maximum transient core temperatures of up to 2000°C.

Studies relating to such transients appear to show that a number of them could be managed in a manner enabling compliance with the aforementioned criteria (reference 67). In particular, total loss of electric power (with cooling by natural convection only) would not lead to core degradation if the helium inventory was maintained. On the other hand, unprotected transients (particularly those involving loss of helium) would rapidly lead to core meltdown, although mitigating measures (nitrogen injection) could probably increase the grace period.

Note that the list of accidents studied is not comprehensive and the proposed classification is based on design objectives rather than the result of a safety assessment.

Notes by IRSN

IRSN does not have sufficient information to evaluate the accident transient studies. Furthermore, the Institute notes that numerous uncertainties affect the results at this preliminary design stage. For example, calculation of the power fluctuation caused by water ingress is subject to significant uncertainties relating to the effective ^{238}U absorption cross-section. Similarly, the mechanical consequences of rapid depressurisation on the reactor core have not been assessed in detail, which would have required a more comprehensive definition of the reactor internals and fuel assemblies. In addition, the computational tools used, such as the "gas" version of the CATHARE2 software application have to date been only partially qualified for this purpose, and the reliability of the calculated fuel temperatures is therefore limited.

The criteria relating to the temperatures of the fuel and the hot plenum have not yet been validated, and are not sufficient to ensure reactor safety. The ability of the second barrier, reactor core support structures and control rods to withstand the aforementioned accident conditions must also be verified.

Lastly, several accident sequences remain to be studied, such as those caused by:

- ejection of a control rod mechanism, causing a break near the core support structures;
- sudden insertion of a control rod into the core (causing a rapid rise in rod temperature);
- helium flow inversion (causing the core support structures to overheat);
- internal hazards and in particular those induced by a blower failure (missiles).

It will also be necessary to identify and examine initiating events liable to cause core movements, including earthquakes (the GFR core, like all fast-spectrum cores, is sensitive to compaction effects).

4.4.2.3 Severe accident

Drawing on the initial safety studies based on the 2007 concept, CEA defined around 15 scenarios liable to result in a severe accident (reference 68). These scenarios were studied with the aim of estimating the effectiveness of the prevention measures, but the consequences of core degradation were not examined.

CEA reported having attempted to characterise the degradation modes for a core built with refractory materials (reference 67). CEA used thermodynamic calculations to separately characterise the behaviour of the core materials at high temperatures and their potential interactions in the fuel element (SiC cladding, "liner", fuel). Exploratory calculations to ascertain the neutron effects of relocating the selected materials were also performed, using the ERANOS software application. For example, examining the case of fuel melt involving seven fuel assemblies revealed that inserted reactivity may reach approximately 1$ if the various materials and the fuel become segregated.

In this respect, we note that the main advantage of the GFR in the event of core meltdown is the absence of thermodynamic or chemical reactions between the reactor coolant and the molten fuel. The intermediate gas system adopted in the GFR concept decreases the risk of high-energy reactions with water (used as a tertiary coolant). However, in case of failure of the primary circuit envelope and the close containment, air would enter the primary circuit. Silicon carbide would then oxidise together with the very hot structures around the core.

IRSN notes that the risk of core meltdown with the primary circuit pressurised is not mentioned in the documents to which the Institute had access. By analogy with PWRs, it is reasonable to assume that dedicated studies and provisions will be necessary with regard to this risk. Furthermore, a release of energy causing of the fuel to be vaporised in the event of a power excursion cannot be ruled out at this stage. Such a scenario may occur in the event of a core meltdown, for example if the core reached prompt criticality following compaction of molten materials. Compared with an SFR, the GFR reactor vessel is designed to withstand a static internal pressure in excess of 7 MPa (nominal pressure plus regulatory margin), which is an improvement. However, a high-energy criticality scenario might be more severe for the GFR than for the SFR, as corium thrown against the vessel would not be adequately cooled by the helium. Accordingly, IRSN considers that it will be necessary to demonstrate, at the least, that the risk of energetic recriticality is sufficiently limited.

Regarding degraded core management, IRSN considers that the GFR concept is clearly unfavourable from a corium cooling perspective. One solution currently envisaged would be to contain the degraded core inside the vessel, but no design options have yet been developed to implement such a provision. Special-purpose systems to cool the degraded core inside the main reactor vessel have yet to be developed. Furthermore, IRSN is unaware of any studies relating to degraded core cooling outside the reactor vessel.

4.5 ENVIRONMENTAL IMPACT AND RADIATION PROTECTION

Primary coolant composition

The helium in the primary circuit naturally contains impurities such as H_2, H_2O, CO, CO_2 and CH_4. Such impurities may also arise as a consequence of leaks from the primary circuit and various auxiliary systems. In a helium reactor, the concentration of oxidising molecules is too low to passivate alloys such as stainless steels or other chromium alloys at the temperatures encountered in the GFR. In particular, helium is corrosive to the tubes in the intermediate heat exchangers, which operate at temperatures above 800°C. As a result, in order to preserve the unit systems in general, the helium must be continuously purified (as for the VHTR – see Chapter 3).

Radiation protection

It is not easy to assess the radiation protection aspects at this stage of development of the process. A benefit with helium is that it does not become activated. Conversely, disadvantages include the fact that helium is hard to confine and does not help to shield workers against radiation exposure. In this loop-based concept, the second barrier has a large surface area (due to the numerous branches). The resulting difficulty in locating leaks from this system might increase the corresponding doses. As a result, the minor actinide transmutation option would result in an even more pronounced increase in worker exposure than with the SFR and LFR concepts.

Note that the process would form ^{14}C (due to the presence of boron carbide (B_4C) in control rods and carbon in the silicon carbide cladding), which may lead to a significant increase in worker exposure.

However, exposure during fuel assembly loading and unloading operations will be limited by the fact that handling operations would be carried out without opening the reactor vessel.

Liquid and gaseous waste management

IRSN is unaware of any studies relating to atmospheric releases and effluent potentially emitted by a GFR in normal operation. However, some similarities may be drawn with the VHTR concept regarding atmospheric releases and effluent associated with use of helium-containing systems.

Accordingly, the helium purification system can be assumed to be among the principal sources of radioactive releases, particularly when regenerating saturated filters. Similarly, the coolant helium is likely to contain gaseous fission products and activated particles released by corrosion and erosion in the primary circuit (in which helium would circulate at between 50 and 70 m/s in some locations). Leaks from the primary circuit into the close containment represent another release route. The release routes of these radioactive elements should be studied. In this respect, it should be noted that uncontrolled diffuse releases are not permitted for reactors in normal operation.

Lastly, a part of the tritium produced in the core and retained by the filter system may be released into the liquid effluents, but a certain proportion would also be released into the reactor containment in gaseous form (the liquid route is preferable).

Note that, unlike the SFR and LFR concepts, the GFR does not require fuel assemblies to be washed, which should significantly decrease the quantities of liquid discharged, compared with those designs. Any primary circuit equipment removed for maintenance would still need to be decontaminated.

Helium has no radionuclide retaining capability. In the event of an accident, fission and activation products as well as corrosion products and rare earths could be dispersed into the environment. No benefits are to be expected in terms of trapping radioactive products.

Ultimately, GFR characteristics should enable liquid effluent discharges to be significantly lower than with the other fast neutron concepts. Such a claim cannot be made for gaseous releases in normal operation and especially in accident conditions, as the level of leaktightness of the primary circuit and the efficiency of the purification system have not been fully determined.

4.6 MATURITY OF CONCEPT AND R&D REQUIREMENTS

In IRSN's view, only the major technological challenges posed by the GFR concept have been identified at this stage, due to the lack of operating feedback relating to this concept. However, experience feedback and current research relating to the HTR concept may yield numerous solutions of benefit to the GFR. This applies principally to the following areas:

- development of structural materials suitable for high-temperature operation;
- thermal insulation technology;
- helium valve technology (in particular fast acting isolation valves);
- helium blowers;
- intermediate heat exchanger and steam generator technology (and in particular, experience feedback from the HTTR);
- helium purification technologies.

Significant R&D efforts will need to be devoted to the following themes:

- fuel elements (cladding and assembly structures);
- emergency cooling systems (reliability, capacity in natural convection mode, etc.);
- high-power blowers.

Regarding variable-speed blower motorisation, note that the target power levels for the GFR primary circuit (in excess of 15 MWe) are not currently available.

4.7 CONCLUSION

The GFR concept has not yet reached the level of maturity at which the construction of an industrial prototype may be considered. Due to its very ambitious specifications, in particular in terms of the desired operating temperatures, a number of technological problems are far from being resolved. On the other hand, it may be possible to build a low-power experimental reactor (ALLEGRO project), which would be an essential step in the development of this concept. The GFR may also reap the benefits of R&D efforts focused on thermonuclear fusion and the VHTR concept.

The preliminary data currently available for the safety demonstration relates almost exclusively to safety system performance and reliability. The main drawback with the GFR concept is the relatively high power density of the reactor core, given the low thermal inertia of the coolant. There is no way to significantly decrease this power density for reasons, in particular, relating to the viability of the fuel cycle. This difficulty is currently being approached by seeking to develop a refractory fuel that would remain intact at temperatures exceeding 1600°C. However, in the light of research published to date, the feasibility of such a fuel has yet to be demonstrated.

Short-term core cooling requires active means, which may generate significant power in the event of primary circuit depressurisation. Although it would probably be possible to design theoretically very reliable emergency cooling systems, there would be considerable uncertainty regarding the reliability values, thereby weakening the safety demonstration (due to the complexity of the systems involved, the multiple support systems, sensitivity to external and internal hazards, etc.).

Regarding the radioactive product confinement function, note that it would be difficult to prevent coolant helium from passing the second barrier. Furthermore, the feasibility of a close containment featuring multiple access hatches that is capable of maintaining back-up pressure during the initial hours of a primary circuit depressurisation transient remains to be demonstrated, particularly for a large reactor.

Helium is not activated, which is an advantage from a radiation protection and environmental impact perspective; however, this advantage is offset by the fact that helium does not provide protective radiation screening for workers (radiation protection) or structures (reactor vessel and primary circuit internals). Nevertheless, the concept appears to be intrinsically more promising in terms of the quantity of liquid effluents produced during normal operation.

Lastly, the severe accident study has yet to be conducted for the GFR. In particular, this study should aim to better ascertain the physical phenomena associated with transformations involving reactor core materials. A number of issues remain to be resolved in terms of managing corium, keeping it subcritical and assessing the feasibility of cooling it.

At the current stage of development, the GFR does not feature intrinsic safety qualities liable to yield significant advances over Generation III reactors. Due to its low thermal inertia, it would even have shorter grace periods. It also appears less promising than the other systems selected by GIF from a severe accident management perspective.

This judgement should be placed in perspective: as the current GFR design is not final, innovations may be developed that would improve the safety characteristics of the concept.

5. LEAD-COOLED FAST REACTORS (LFR)

This chapter covers fast reactors cooled using molten lead (LFR) or a lead-bismuth eutectic alloy (LBE).

5.1 HISTORY AND PROSPECTS FOR DEVELOPMENT

5.1.1 *BACKGROUND*

LBE-cooled reactors were built in the Soviet Union for use as propulsion units in military submarines. Seven Alfa-class military submarines powered by 155 MW OK-550 reactors, and later BM-40A reactors, were operated using highly enriched ^{235}U fuel between 1967 and 1983. However, no lead- or LBE-cooled reactors have been built to date for civilian purposes.

5.1.2 *ONGOING PROJECTS AND PROSPECTS FOR DEVELOPMENT*

Russia continues to be interested in this type of reactor and is developing two prototypes: a lead-cooled reactor (the 300 MWe, nitride fuel BREST-300) and an LBE-cooled reactor (the 100 MWe SVBR 75-100, for which several types of fuel are being studied). These two reactors are currently scheduled to begin operating in 2018.

In Europe, ESNII has adopted the LFR, alongside the GFR, as an "alternative" solution to SFR technology. Consequently, the ELSY (European Lead-cooled SYstem) project was launched in 2006, to define the principal options for the 1500 MWth, 600 MWe, industrial-scale ELFR (European Lead Fast Reactor), followed in 2010 by the LEADER (Lead-cooled European Advanced DEmonstration Reactor) project, both funded by the European Commission under EC's FP6 and FP7, respectively. The LEADER project is also researching a 100 MWe 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) intends to build a demonstrator 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 subcritical nuclear fission reactor. The reactor would use LBE as a coolant and as a proton beam-activated spallation source. The reactor is designed to also operate in critical mode. The Central Design Team for a Fast Spectrum Transmutation Experimental Facility (CDT) project included in EC's FP7 should yield a preliminary design. Submission of the MYRRHA safety options file to the Belgian Nuclear Safety Authority is planned for 2014. Reactor commissioning is planned for 2024. Despite the significant differences between this system and the reactors studied for the LEADER project (LBE coolant, etc.), there are strong synergies between the projects (relating to the definition of the reactor shutdown system, in-service inspection solutions, etc.). MYRRHA should yield valuable information for the development of LFRs.

Lastly, there are projects for small-scale reactors in the USA (SSTAR and STAR-LM, designed to generate between 10 MWe and 100 MWe), in South Korea (PASCAR), in Japan (100 MWe PBWFR) and for an experimental reactor in Sweden (0.5 MW European LEad Cooled TRAining reactor - ELECTRA).

5.2 MAIN CHARACTERISTICS

The descriptive information in this chapter refers to the configuration currently planned for the industrial-scale reactor called ELFR (600 MWe, 1500 MWth) being studied by the EU's LEADER project. It is informed by the reference documents 70, 71 and 72. The only exception, due to a lack of information, concerns the description of the containment systems, which relates to the ALFRED demonstrator, although the systems adopted for the ELFR would probably have a fairly similar design[31].

5.2.1 *GENERAL DESIGN*

As with the SFR, LFR concepts exist in both "integrated" and "loop-based" versions. As lead does not react violently with water, an intermediate cooling circuit is not essential. Accordingly, some "integrated" reactors such as the ELFR do not have intermediate cooling circuits: the steam generators are immersed directly in the main vessel (see Figure 28). "Loop-based" concepts avoid locating the steam generators in the main vessel.

LFR systems operate at low pressure due to the high boiling points of lead and LBE.

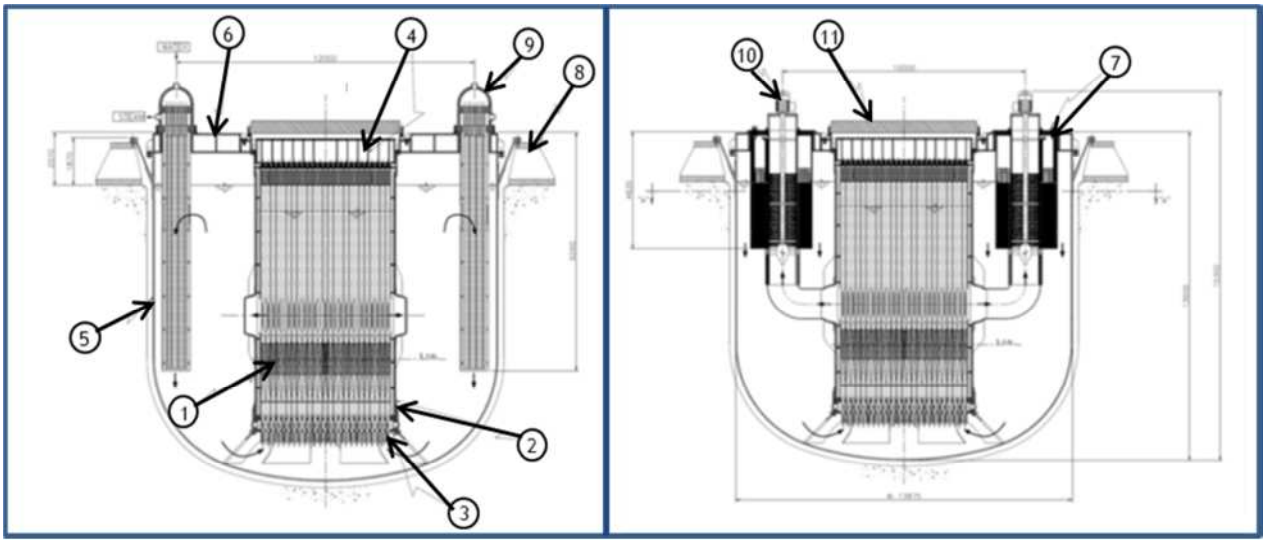

Figure 28: ELFR primary circuit: (1) fuel assembly, (2) internal vessel (separation between the cold zone and hot zone of the lead in the primary circuit), (3) diagrid, (4) core upper grid , (5) main vessel, (6) reactor cover, (7) steam generator, (8) vessel support structures, (9) DHR dip coolers , (10) primary pump, (11) fuel assembly cover

The main vessel is cylindrical in shape with a hemispherical bottom head. It is suspended in the reactor pit. The reactor cover supports the core and the various components of the primary circuit. The volume between the free surface of the lead and the reactor cover is filled with inert gas (cover gas).

[31] In general, the ELFR configuration is largely based on the configuration defined for the ALFRED demonstrator, also for the LEADER project. However, some ALFRED components are simpler in design, to accelerate the licensing and construction phases. The differences are identified later in this chapter.

The internal vessel (Figure 29), which is the first structure around the core, is secured to the reactor cover; it supports the fuel assemblies and forms a separation between the hot plenum and cold plenum. The lower part of this vessel supports the diagrid and the upper part supports the core upper grid. It is closed by the fuel assembly cover. The cooling system comprises eight steam generators.

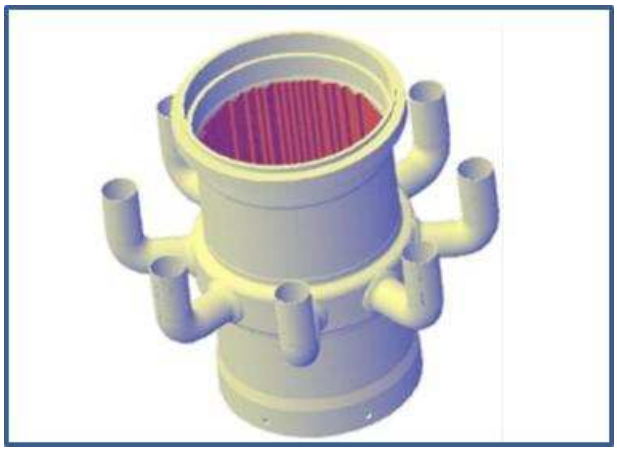

Figure 29: Internal vessel

The steam generators, primary pumps and one of the two decay heat removal systems (DHR N2, see Section 5.3.2) are located in the gap between the internal vessel and the main vessel.

The steam generators and primary pumps are integrated in individual vertical units (Figure 30).

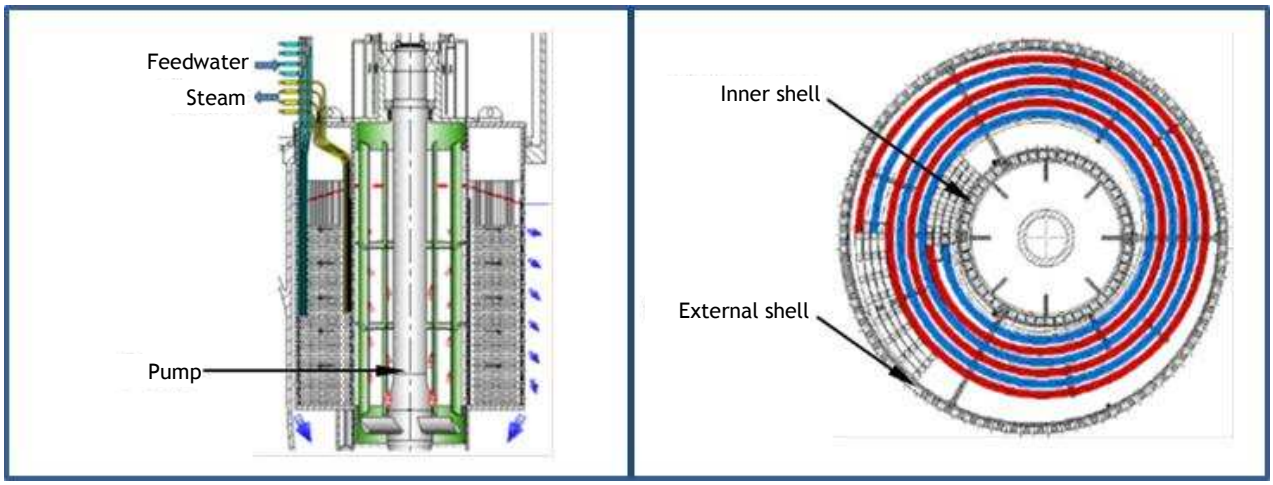

Figure 30: ELFR – Coiled steam generator

Each primary pump is installed in the centre of a coiled steam generator[32]. Hot lead from the core outlet enters the pump, rises vertically towards the pump shaft and then flows radially through the SG tubes. The steam generators are compact vertical units. The internal and external shells consist of two walls with openings allowing lead to flow around the tube bundle (through which water circulates) to the cold plenum. The system must be able to cool the core in natural convection mode in the event of loss of mechanical lead circulation.

[32] For the ALFRED demonstrator, the steam generator consists of straight, double-wall bayonet-type tubes to simplify the design.

It is important to note that the close relationship between the steam generator and the primary pump implies that damage to either component would automatically impact the other, particularly in the event of an earthquake. In particular, the risk of the primary pumps seizing must be studied. More generally, the issue of the feasibility of building this type of steam generator must be examined, taking into account the required level of quality and the total absence of industrial experience, but also in-service inspection and repairability considerations, or the ability to replace such components during reactor operation.

5.2.2 *GENERAL SAFETY OPTIONS*

Lead- and LBE-cooled fast reactors must fulfil the same objectives as the other fast-spectrum concepts in terms of energy resource use and closed fuel cycles. They can be breeders.

Some designs harness the low chemical reactivity (lead does not react violently with water, air or fuel) and good thermodynamic and neutron properties inherent to this concept in order to simplify plant design. However, the design must also reflect the drawbacks associated with using a highly corrosive and erosive metal that is chemically toxic and has a relatively high solidification point.

Accordingly, the general design options adopted to ensure the safety functions of an LFR are as follows:

- Core featuring a “spaced out” fuel assembly lattice to decrease flow velocities, which must be limited in order to mitigate erosion phenomena.
- Protective measures are necessary for structures in contact with the highly-corrosive lead (or LBE).
- Inspection devices must be developed that can operate in the presence of protected structures (see previous point) and dense, opaque fluids at high temperature (in the case of lead).
- The primary circuit must be kept at a temperature above the melting point of lead in all foreseeable conditions, to prevent freezing risks.
- The size of the primary circuit must be limited to minimise structural issues relating to the high mass of the coolant, in particular in the event of an earthquake.
- A double envelope must be implemented around the primary circuit, equipped with a leak detection system to render any loss of reactor coolant highly improbable in all foreseeable conditions.

5.2.3 *CORE CONFIGURATION*

The configuration adopted for the core features fuel assemblies with a wrapper tube. This solution offers several benefits over a configuration with open assemblies (more stable core geometry, easier handling, etc.). However, it also has some disadvantages, and in particular the risk of assembly blockage. A configuration with multiple inlet openings was adopted for the ELFR to prevent the risk of instantaneous total blockage.

Design constraints relating to the erosion risks result in a “spaced out” fuel assembly design, with each assembly containing a coolant amount enabling to decrease the velocity of the coolant flow through the core to an acceptable level. Such a design, made possible by the favourable nuclear properties of lead, results in cores with relatively low power density, making the LFR less sensitive to local cooling defects than the SFR.

The fuel assemblies (Figure 31) measure approximately 10 m in height and extend above the free surface of the lead into the cover gas, which facilitates inspection and handling operations. Tungsten ballast will be used to counter the buoyancy forces exerted by the lead. Lastly, springs will prevent lift-off caused by the hydrodynamic loads and enable the fuel assemblies to accommodate any axial thermal expansion.
The target lead temperatures at core inlet and outlet are 400°C and 480°C, respectively.

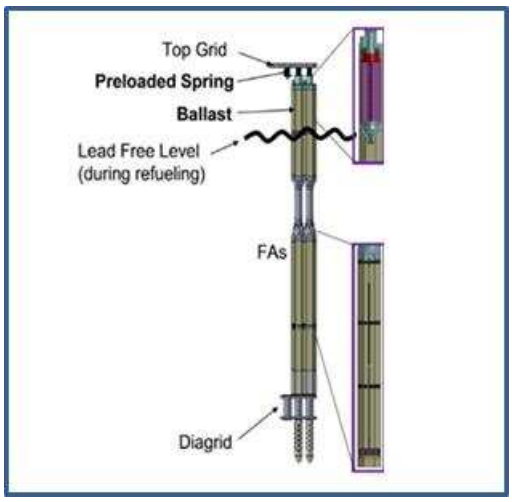

Figure 31: Fuel assembly

5.2.4 *FUEL*

The fuel currently envisaged for the ELFR is MOX for the first fuel load, and it will include minor actinides for subsequent loads. The possibility of using a nitride fuel is also being studied. Carbide-based fuels, with higher densities than MOX, can also be considered for LFRs. They would enable the reactor to operate at lower temperatures, generate no risk of forming oxides with lead or LBE in the event of cladding failure, and reduce the loss of reactivity during a fuel cycle, in turn enabling control rod worth to be decreased, reducing risks in the event of inadvertent removal of these rods.

5.2.5 *COOLANT*

Lead and LBE have useful neutron properties. Specifically:

- low moderating effect due to their high density, and lower than in case of sodium ;
- lower neutron absorbing effect than sodium;
- excellent reflecting properties, greatly limiting neutron leakage.

These materials have high boiling points (1745°C for lead and 1670°C for LBE at a pressure of 0.1 MPa).

The specific heat capacity of lead (C_p) is much lower than that of sodium (by a factor of around 9); however, due to its high density (approximately 10,500 kg.m^{-3}), the thermal inertia of lead is 50% higher than that of sodium for a given volume of coolant.

As already stated, lead and LBE offer the advantage of not reacting violently with air or water. There is also no risk of chemical reaction between lead and MOX fuel.

Nevertheless, using lead and LBE entails numerous disadvantages, which are analysed in Section 5.4.1.

5.2.6 *BARRIERS*

5.2.6.1 First barrier

The first barrier consists of the fuel element cladding. One of the main problems with the LFR relates to the integrity of cladding when subjected to the corrosive and erosive action of lead (see Section 5.4.1.1).

First barrier monitoring

Regarding monitoring of the first barrier, the solution envisaged for the ELFR at this stage is based on detecting noble gases in the cover gas space.

5.2.6.2 Second barrier

As with the SFR, the second barrier consists of:

- main reactor vessel and a safety vessel designed to collect lead coolant in the event of a leak from the main vessel (the two vessels are arranged such that the SG lead intake openings are not exposed in the event of a leak); the safety vessel consists of a steel film lining the reactor pit;
- top head: the solution under consideration for the ELFR consists of a thick steel plate with penetrations for the steam generator/primary pump units and one of the decay heat removal systems (DHR N2, see Section 5.3.2);
- steam generator tubes;
- heat exchanger tubes in the DHR heat exchanger system N2;
- auxiliary systems carrying lead coolant or cover gas away from the primary circuit.

As with the SFR, the second barrier is relatively complex inasmuch as it involves several components, some of which are themselves complex (for example, the steam generators in the ELFR). Furthermore, like the SFR it is likely to be subject to leaktightness problems relating to inert gas leaks around the primary circuit top head or the opening of pressure relief valves designed to control the pressure in the cover gas space (IRSN does not have any information on this topic, however).

Note that the high density of lead makes LFRs very sensitive to earthquake hazards and poses a problem with regard to the second barrier, especially in an integrated concept. For the ELFR, design provisions were adopted concerning, for example, the vibrations created in the reactor internal vessel by an earthquake. The designers have also devised some highly innovative designs with a view to limiting the volume of lead. R&D initiatives are in progress in this area.

Due to this problem, the prospect of developing very large LFRs does not appear very realistic.

Monitoring of the reactor second barrier

The gap between the safety vessel and the main vessel is large enough to allow the necessary in-service inspection equipment to be set up. However, the use of lead poses the following difficulties relating to in-service reactor vessel inspections and the reactor vessel internals:

- The cold shutdown temperature is very high (380°C) due to the risk of the lead freezing, requiring inspections to be performed at high temperature.
- The high density of lead makes inspection instruments hard to insert.
- Optical techniques cannot be used, making the use of alternative techniques such as ultrasonic methods

necessary.

- The low difference in density between lead and steel decreases the sensitivity of ultrasonic inspection methods.
- The oxide layer (necessary in order to limit the corrosion risk) disrupts the contact between the coolant and the system walls, impairing ultrasound transmission quality.

5.2.6.3 Third barrier

As already stated, IRSN does not have any information about the third barrier envisaged for the ELFR. This section therefore contains information relating to the ALFRED prototype.

The third barrier consists of a primary containment and a secondary containment (Figure 32). The envelope of the primary containment consists of a cylindrical reinforced concrete building with a domed roof. The secondary containment corresponds to the space between the primary containment envelope and the reactor building.

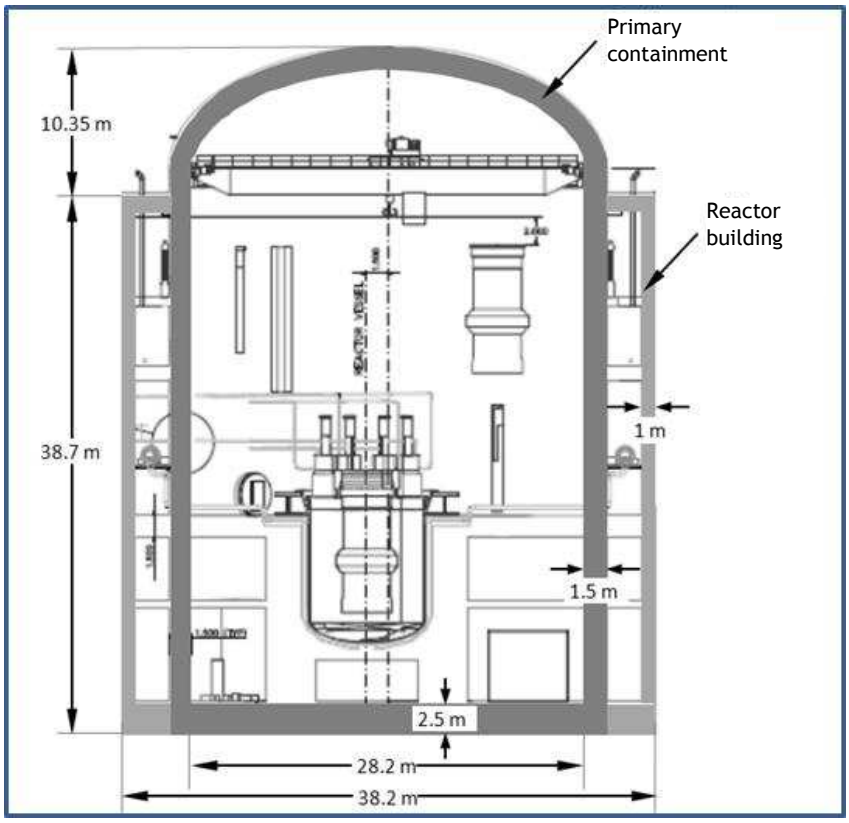

Figure 32: Primary and secondary containments

5.2.7 *FUEL ASSEMBLY HANDLING AND STORAGE*

The top of the internal vessel, which is the first structure surrounding the core, is sealed by the fuel assembly head. The fuel assemblies, the tops of which extend above the free surface of the lead, may be extracted from the core by removing this head. The FPs trapping capability of lead ensures that radioactivity in the gas cover space is low.

The necessary fuel assembly cleaning processes prior to their storage in water appear problematic. Nitric acid-based cleaning processes have been tested in the laboratory, but their application on an industrial scale is as yet untested.

Detailed studies of the fuel handling and storage system have not yet been conducted.

IRSN notes the potential difficulties relating to the significant height (~10 m) and weight of the fuel assemblies (active part + tungsten ballast), as well as the temperatures (~400°C) and washing processes.

5.3 CONTROL OF SAFETY FUNCTIONS

5.3.1 *REACTIVITY CONTROL*

Neutron feedback

As with the SFR, LFR cores are subject to a risk of "void effect" reactivity insertion.

Although the moderating and neutron absorption capability of lead is low, a multi-dollar effect may be possible due to the large volume occupied by coolant within the fuel assemblies.

It is not necessary to consider a scenario resulting in coolant boiling for the LFR, as the main reactor and steel internals would melt before the boiling point of lead was reached (at around 1700°C).

However, there may be a risk of lead overheating and core "voiding", for example if a fuel assembly were to become blocked, if cladding failure created a fission product gas bubble or if steam were to be injected as a result of a break in one of more steam generator tubes. Such a scenario, which would affect the active part of the reactor core, where the void effect would be positive is considered unlikely by the concept designers.

IRSN considers that the feedback coefficient associated with expansion/"voiding" of the lead in the active part of the core is a major issue for LFR safety; consequently, a detailed analysis of the various scenarios potentially resulting in significant reactivity insertion is necessary, even if their probability of occurrence is considered to be low by the concept designers.

Core compaction risk

Reactivity may also increase in the event of core compaction as the fuel assemblies in the reactor core are moved closer together (particularly as they are arranged in a "spaced out" configuration). However, the positioning of the fuel assemblies, similar to that in an SFR (with the assemblies set into the diagrid and with plates between them), and the choice of a hexagonal wrapper tube help to decrease the risk of core compaction.

In addition, the high density of lead compared to that of the materials used in the reactor internals and fuel further reduces this risk.

Means of reactivity control

In view of the aforementioned risks, effective means of shutting down the chain reaction are essential. Diversified reactor trip systems have been designed. For example, the ELFR features two separate, redundant diversified reactor trip systems:

- *a control rod (CR) system* used to control the reactor in normal operation (startup, reactivity control over the fuel cycle, normal reactor shutdown) and for reactor trips. The 12 control rods are inserted from the bottom of the active part of the core, aided by the strong buoyancy forces exerted by the lead coolant. In normal operation, the control rod pushes the rods down with a pressure screw located on the top vessel head, shielded against radiation by a protective block. The actuator is coupled by means of a long rod via an electromagnet. When the electromagnets are de-energised (in the event of a reactor trip), the rods are free to rise; the insertion time is approximately 60% longer than with a gravity-dropped rod. The control rods consist of a bundle of 19 pins (B_4C, enriched to 90% with ^{10}B) cooled by the lead. The upper part of the pins includes a plenum in which to collect the helium and tritium released by reactions involving the ^{10}B;

- *a diversified safety rod (SR) system*, used exclusively in reactor trip conditions, operating redundantly with the control rod system. The 12 rods are inserted from above by a pneumatic system. In case of loss of this system, a tungsten ballast slowly inserts the rod under the effect of gravity with a reduced insertion velocity. Safety rods have a similar structure to control rods, but with a bundle of 12 pins. No plenum is planned, and the small amounts of gas that form are released into the coolant

Core monitoring

As stated previously, the core must have a low power density, due to corrosion constraints. Depending on reactor power, this may result in a large core, which is a disadvantage from a monitoring perspective and in terms of ensuring uniform power distribution.

IRSN considers that controlling reactivity in an LFR is likely to be more challenging than in other reactors, due to the strong buoyancy forces of lead, making rod insertion by the pneumatic system difficult and requiring the presence of tungsten ballast in case of failure of the pneumatic system. The rod insertion time is a critical aspect for the design of the reactor trip systems.

5.3.2 *DECAY HEAT REMOVAL*

The LFR has high thermal inertia as a result of the large volume of lead and its high density, resulting in long grace periods during which to start the DHR heat exchanger systems in response to a loss of cooling. However, as with the SFR, an extended lack of cooling would cause the temperature of the internals to rise, with a risk of thermal creep-induced failure. Furthermore, the low pressure drop (concomitant with the low lead flow velocity required in order to minimise erosion-related problems) combined with the high density and high thermal expansion coefficient of lead facilitate core cooling by natural convection. However, as with sodium, natural convection-based cooling of the reactor core and DHR heat exchanger systems has not at this stage been demonstrated to be satisfactory.

Decay heat removal systems

Various types of system may be used to remove decay heat. For the ELFR, the first system, used both in normal operation and in accident conditions, is the secondary cooling system (SG flow control and condenser bypass).

In the event of unavailability of the secondary system, the ELFR decay heat removal function is performed by two independent, redundant and diversified systems (see Figure 33):

- DHR N1: Isolation Condenser System (ICS);
- DHR N2[33]: Water Decay Heat Removal System.

The two systems are considered as passive components but starting them requires valves to be actuated. These valves are powered by redundant, diversified energy sources (in particular, batteries). The principal aim is to ensure that the effectiveness of the two systems is not compromised in the event of total loss of electric power.

[33] Note that, for ALFRED, there are no diversification requirements, and DHR N2 and DHR N1 are identical.

The activation procedure must preclude simultaneous operation of the two systems, to prevent excessive cooling that might cause the lead to freeze. DHR N1 would be activated first, switching to DHR N2 in case of failure of DHR N1.

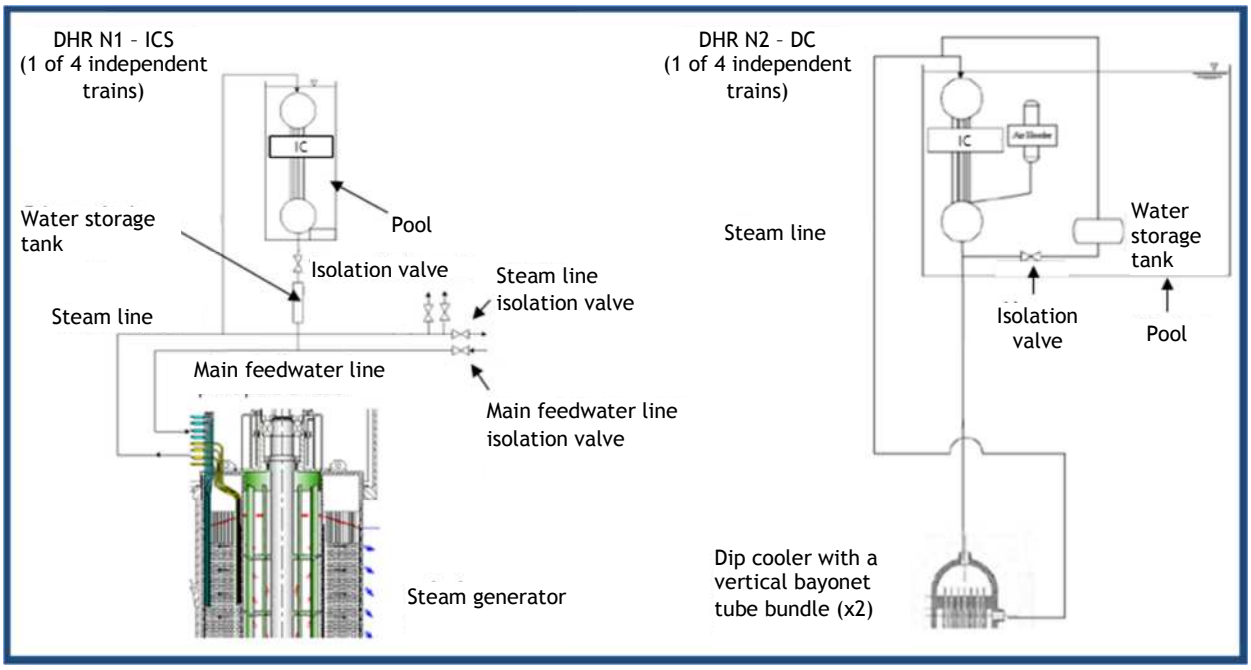

Figure 33: ELFR decay heat removal systems

Isolation Condenser (IC-DHR N1)

The Isolation Condenser System consists of four independent sub-systems connected to four of the eight steam generators. Each of these systems comprises:

- a water/water heat exchanger (isolation condenser – IC) consisting of a bundle of vertically-mounted tubes with horizontal manifolds at the top and bottom. The top manifold is connected to the main steam line; the bottom manifold is connected to the main feedwater line;
- a pool in which the isolation condenser is immersed; the pool contains a sufficient quantity of water to ensure that the decay heat removal function can be performed for three days;
- an isolation valve, which is doubled up in order to comply with the single failure criterion;
- a water storage tank.

In normal operation, the isolation valve at the base of the IC heat exchanger (which is filled with water) is closed and there is no heat transfer. To activate the system, the main feedwater line and the steam line must be isolated and the heat exchanger isolation valve must be opened in order to inject water into the steam generator. The steam thus generated enters the IC, where it condenses.

The isolation condenser system is designed to remove 7.5 MWth per sub-system.

Water Decay Heat Removal Systems (DHR-N2)

DHR N2 consists of four independent systems with eight dip coolers immersed in lead that use water as a secondary coolant. Each of the four systems comprises:

- two dip coolers with a vertical bayonet tube bundle;

- an isolation condenser component (currently at the design stage) to condense steam and return water to the system (in order to operate in a closed cycle without releasing steam to the atmosphere);
- a pool (in which the IC is immersed) containing enough water to ensure that cooling can continue for three days;
- a water storage tank.

DHR N2 is started upon concomitant reception of the reactor trip, main feedwater line isolation, main steam line and DHR N1 failure signals. If all conditions are met, the water storage tank isolation valve is opened and water is gravity-discharged into the heat exchanger tube bundle, where it evaporates. The steam thus produced is condensed in the isolation condenser before returning to the heat exchanger.

In the event that the aforementioned systems are unavailable, other decay heat removal options could be considered (including reactor pit concrete cooling systems and flooding the gap between the main reactor vessel and the safety vessel). IRSN does not have any detailed information relating to such options.

IRSN wishes to make the following observations regarding decay heat removal:

- The high thermal inertia of an LFR strongly limits coolant temperature-rise kinetics in the event of loss of cooling, ensuring significant grace periods (several hours) during which to implement decay heat removal solutions before the reactor reaches temperatures liable to damage the reactor vessel. In addition, natural convection-based core cooling may be possible, although the effectiveness of such a solution has not been determined at this stage.
- High and low coolant temperature limits must be observed:
    - the maximum temperature (~500°C) relates to the corrosion risk (see Section 5.4.1.1);
    - the low temperature (~400 °C) relates to the risk of the lead freezing. The risk of embrittlement of certain materials may also be a decisive factor.
- Particular attention must be paid to common-mode risks for the two DHR heat exchanger systems (DHR N1 and DHR N2), due to their presence inside the reactor vessel.
- The effectiveness of natural convection cooling in the event of stratification in DHR N1 must be examined.

5.3.3 *CONFINEMENT OF RADIOACTIVE MATERIALS*

As for the SFR, the “integrated” LFR concept offers an advantage from a confinement perspective, as the lead remains confined in the reactor vessel.

Little information is available regarding the auxiliary systems and the inert gas system in the gas cover space. However, as in the SFR, they may be assumed to form a bypass route for the second barrier.

Lastly, as with the SFR, the confinement strategy must consider both the risk of radiological releases and the risk of toxic releases inherent to lead, particularly in severe accident conditions.

5.4 RISK ANALYSIS

5.4.1 *RISKS INHERENT TO THE USE OF LEAD OR LBE*

5.4.1.1 Risks of erosion, corrosion and degradation of the mechanical properties of structures

This seems to be a key issue for LFRs. The information in this section is drawn from reference documents 73, 74 and 75.

Corrosion risk

Lead and LBE are highly corrosive to steel structures, and in particular the reactor vessel and fuel cladding. The iron, nickel and chromium contained in the steels used for these structures are liable to dissolve in lead.

Currently, the reference solution adopted to prevent this phenomenon is based on controlled injection of oxygen in order to form a protective layer of iron oxide on the surface of structures that come into contact with lead. This process, developed further to operating accidents involving Soviet submarines, is applied in LBE-using experimental facilities, which are primarily used for ADS research. Some adaptations have been made for lead, but only for small systems.

This solution requires the concentration of dissolved oxygen to be managed within a very narrow range. The oxygen content must be limited, to prevent precipitation of lead oxide, which might otherwise form clumps liable to obstruct fuel assemblies. Conversely, if the oxygen content is locally too low, the structures throughout the system may not be adequately protected. Lastly, it is important to note that the quantity of lead oxide produced and the thickness of the iron oxide layer are temperature-dependent, and the temperature in the primary circuit varies. Accordingly, reliable electrochemical oxygen meters must be used to monitor local oxygen contents, in order to properly protect structures that operate at different temperatures, while preventing lead oxide precipitation. The oxide layer must also be monitored.

Although this process is feasible for a small system, implementing it for a large installation would be extremely complex, as the integrity of the protective layer must be ensured and its characteristics (in particular its thickness) optimised for all operating conditions (including draining/filling conditions), throughout the primary circuit and in areas at different temperatures. In the event that the protective coating is damaged, the risk of oxide particles being released into the reactor must be considered, as well as poorer resistance to corrosion.

In any case, based on currently available materials, temperatures must remain below 500°C.

Creating a surface coating on the steel is being studied, to facilitate management of lead chemistry and enable higher cladding temperatures to be achieved. One of the possible methods consists in forming a preliminary coating on the steel by means of an aluminium-containing alloy (Fe-Cr-Al) and then melting this coating and the surface of the steel with a pulsed electron beam (using the GESA process developed by the Karlsruhe Institute of Technology - KIT). The feasibility of applying this process industrially in an irradiated environment remains to be demonstrated.

Other protection methods, also used in conventional facilities, are also being studied.

A final solution being researched involves directly modifying lead chemistry to limit its corrosive action (corrosion inhibitors).

Erosion risk

Lead is highly eroding and for this reason its flow velocity in the primary circuit must be limited to approximately 2 $m.s^{-1}$. This implies having a large lead flow cross-section in the core, and hence a spaced-out lattice to ensure that it is correctly cooled. For this reason, to limit core size, only low-power projects (between 100 and 600 MWe) are currently being studied.

Erosion/corrosion is a major problem, in particular for the primary pump, due to the fact that it is located in the hot leg, that the relative speed between the blades and the lead is approximately 10 $m.s^{-1}$ and that the blades cannot be protected by an oxide layer. R&D programmes have been set up to research special-purpose materials (Maxthal ceramics, Ti_3SiC_2, Noriloy, etc.) or tantalum-based coatings.

Degradation of mechanical properties

Independently of the corrosion risk, some steels can become embrittled in contact with lead or LBE, and this may be exacerbated by irradiation. Studies (reference 75) conducted for T91 steel, as used in the ELFR cladding, revealed that this embrittlement is only liable to occur in certain specific circumstances: with test specimens that are hardened and "notched" (either mechanically or by a corrosion process), in a temperature range of approximately 350-400°C, with a low oxygen content and intimate contact between the T91 steel and the molten metal. It therefore appears necessary for the designers and related R&D organisations to examine this study in greater depth.

The resistance of structures to creep and wear is also affected by contact with lead. As before, an oxide layer or aluminium-based protective coating (if determined to be feasible) may be applied with the aim of ensuring a high standard of protection.

IRSN considers that controlling corrosion through precise management of the quantity of dissolved oxygen at all points in the primary circuit would appear to be an extremely challenging task, particularly as no genuinely usable experience feedback is available. At the time of publication, the validity of the various tested methods has not been demonstrated.

A combination of the corrosion risk and erosion risk appears to be particularly complicated to manage.

IRSN notes that this issue entails entrusting the safety of the installation largely to operational solutions that are particularly hard to manage, which is not a desirable outcome for a Generation IV reactor.

5.4.1.2 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 compounds would decrease the heat-exchange properties between the coolant and internals and possibly obstruct fuel assemblies causing overheating or even fuel melt, as occurred during the 1968 accident involving a Soviet submarine reactor (see Section 5.6.1**).**

Solutions to monitor the oxygen concentration in the primary circuit, purify the reactor coolant and eliminate lead oxide are planned.

5.4.1.3 Risk of coolant freezing

Lead freezes at 327°C, which is only 73°C lower than the core inlet temperature in normal operation. This implies the need for heating systems for all lead-containing systems, to prevent lead freezing, in particular during reactor shutdowns. Like sodium, lead expands upon thawing.
IRSN has no information regarding the consequences of lead freezing or thawing in an LFR (the LEADER project includes R&D initiatives on this topic, see Section 5.6.2).
It should also be noted that restrictions on the minimum acceptable temperatures (350°C/400°C) may be required, due to the risks of steel embrittlement.

The principal benefit of using LBE is that it freezes at 125°C, which is significantly lower than the freezing point of lead and similar to that of sodium (98°C). This means that the freezing risks for an LFR using LBE are in principle easier to manage. Note that LBE nevertheless froze in the reactors of three Soviet submarines, one of which was under the icecap at the time. This led to their permanent shutdown.

5.4.2 *ACCIDENT SEQUENCES*

This section describes the studies of the various protected[34] and unprotected accident sequences analysed for the ELFR. The information in this section is taken from reference document 76.

It should be noted that these studies were conducted using tools that are still in the process of being validated.

The most important transients analysed in the aforementioned studies are as follows:

- protected loss-of-flow transient;
- protected core overcooling transient;
- protected SG outlet steam line break transient;
- unprotected loss-of flow transient;
- unprotected loss-of-cooling transient;
- unprotected reactivity insertion transient;
- unprotected partial fuel assembly blockage transient;
- steam generator tube rupture.

Regarding protected transients, the facility can be durably maintained in a safe state by tripping the reactor, isolating the secondary cooling system and starting the decay heat removal systems. For all the protected transients, the freezing temperature of lead in the coolest areas is reached after a number of hours, allowing adequate time for human intervention. Nevertheless, R&D efforts are being conducted with the aim of eliminating the need for action by the operator for this type of transient.

[34] A protected accident sequence is one in which the control rods can be inserted.

The analysis of the unprotected transients and the steam generator tube rupture accident is described below. The behaviour of LFRs in a severe accident is also addressed, although the research conducted to date has not revealed any sequences liable to result in core meltdown.

5.4.2.1 Unprotected loss-of-flow transient

The unprotected loss-of-flow transient is initiated by primary pumps shutdown with the transition to half-flow occurring in 0.5 seconds. The protection system is assumed to be inoperative throughout the transient, but the steam generators operate normally.

During the first part of the transient, the power level increases due to the positive feedback coefficient resulting from the expansion of the lead in the central part of the core. Thereafter, the temperature rise causes negative neutron feedback due to expansion in the diagrid, the top grid and the plates. These effects, combined with negative feedback caused by the Doppler effect and axial expansion of the fuel cause the power level to decrease and subsequently stabilise at around 1200 MWth. The maximum temperatures reached by the fuel and cladding during the transient are respectively around 1580°C and 830°C, which are below their melting points.

In conclusion, the calculations described above show that an unprotected loss-of-flow transient does not result in fuel melt or cladding melt in the ELFR. However, no information has been provided regarding the ability of structures such as the main reactor vessel and the steam generators to withstand the transient conditions.

5.4.2.2 Unprotected loss-of-cooling transient

The unprotected loss-of-cooling transient is initiated by a loss of feedwater to all steam generators without rod drop. The secondary systems are automatically isolated and DHR N1 is activated (three of the four isolation condensers are started).

The core continues to be cooled in forced convection and the core power level decreases as a result of negative neutron feedback due to the increased core inlet temperature. The temperature in the primary circuit continues to gradually rise, owing to the mismatch between the heat generated by the core and that removed by DHR N1. In parallel, the temperature gradient in the core decreases due to the drop in power output, resulting in a near-uniform temperature distribution throughout the primary circuit. The reactor vessel walls reach thermal equilibrium with the lead in the medium term. One hour after the start of the transient, the primary circuit reaches a temperature of around 1000°C, jeopardising the integrity of the fuel cladding.

The main risk relates to the reactor vessel, the integrity of which may be compromised. The reactor vessel reaches a temperature of 700°C approximately 12 minutes after the start of the transient, and 1000°C after an hour (see Figure 34).

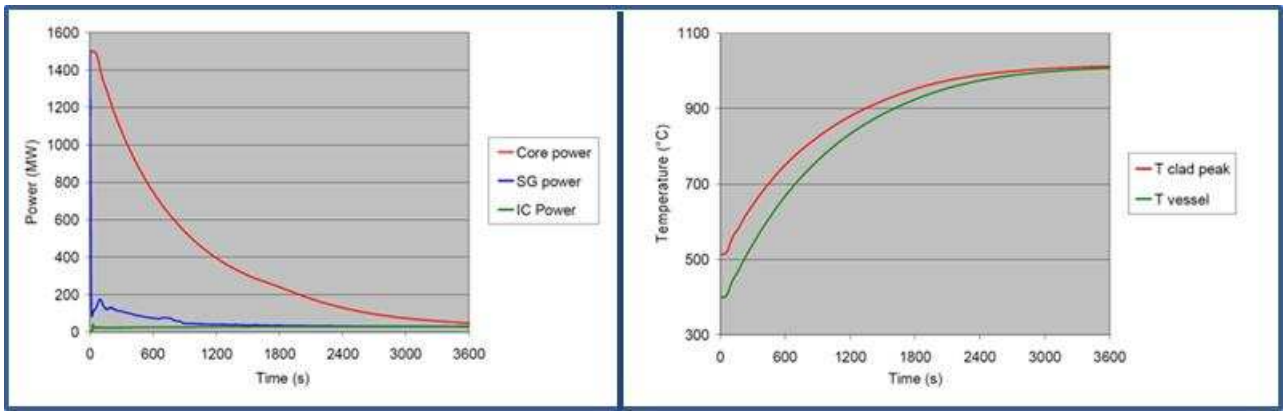

Figure 34: Core/SG/IC power trends and maximum cladding and reactor vessel temperatures calculated at the start of the cycle (using ENEA's RELAP code).

The solution currently envisaged as a means of preventing excessively high temperatures from being reached in the event of a loss of core flow or loss of cooling is based on optimising neutron feedback in the core.

5.4.2.3 Unprotected reactivity insertion transient

Reactivity insertion of 260 pcm over 10 seconds at the end of the fuel cycle with the reactor operating at nominal power has been studied. This transient is considered to envelope other transients such as fuel handling errors, inadvertent control rod withdrawal or core compaction following an earthquake. Following the aforementioned reactivity insertion, the reactor power increases by a factor of 2.4. The maximum fuel and cladding temperatures respectively increase from ~1500°C and ~500°C to ~2680°C (resulting in only partial melting of the centre of the hottest pellet) and ~720°C (at which temperature there is no cladding failure).

The risk of propagation to neighbouring assemblies in the event of partial fuel melt must be examined, as for the SFR, although the risk may be lower.

5.4.2.4 Unprotected partial fuel assembly blockage transient

For the unprotected partial fuel assembly blockage transient, various cases were studied, considering blocked fractions of the total cross-section of fuel assembly inlet orifices ranging from 20% to 97.5%.

These simulations were performed using the SIM-LFR code[35], assuming the reactor to be operating at nominal power, at the end of the fuel cycle with no control rod insertion. Coolant flow through the blocked assembly was considered to be linearly proportional to the blocked cross-section; radial heat transfer was not considered. The results obtained reveal that:

- Core melt never occurs, even when 97.5% of the inlet cross-section is obstructed.
- The fuel cladding reaches a temperature of 700°C when the obstructed cross-section reaches around 85%.

No mention is made of studies of local blockages internal to the pin bundle.

IRSN notes that the risk of obstruction with an LFR is likely to be greater than with an SFR, owing to the oxide formation risks.

[35] No reference documentation for this code was provided.

5.4.2.5 Steam generator tube rupture

This accident is initiated by a sudden break of one or more steam generator tubes. This allows a flow of water to be injected into the lead, causing a pressure surge as the water vaporises. The consequences of such an accident, examples of which include the production of steam bubbles at high pressure or the propagation of pressure waves through the SG or throughout the primary circuit, depend on the flow rate of the injected water, which reacts thermodynamically with the hot lead.

A perforated double wall casing was introduced to prevent the effects of this accident from spreading to the core (the holes in the two walls are arranged to prevent lead from flowing through them when in contact, see Section 5.2.1). Furthermore, the feedwater and steam manifolds are located outside the reactor vessel, avoiding the risk of excessive overpressure.

Two types of calculation were performed using the SIMMER-III code, respectively simulating a single-ended guillotine break and a double-ended guillotine break. In the second case, a check valve failure is also considered. Furthermore, the use of venturi[36] water flow limiters in each tube was also tested.

Preliminary analyses of the two types of calculation reveal that:

- Even without installing venturi flow limiters or other mitigating solutions, the pressure generated by the interaction between water and lead, if water vaporises instantly, may be sufficient to present a risk to the structural integrity of the steam generator but would not pose a risk to the reactor vessel internals.
- Steam bubbles liable to create a positive void effect are unlikely to reach the core inlet cross-section, due to the hydraulic thrust exerted by the lead. Mechanical devices may be installed at the steam generator outlets to facilitate the separation of water vapour and lead.

In general, conducting an in-depth analysis of the phenomena liable to occur during this accident requires a dedicated, full-scale installation.

5.4.2.6 Severe accident

Because molten MOX fuel is lighter than the reactor coolant (lead), the designers consider that it is certain to float and that there is therefore no need for a core catcher. In view of the difference in density between molten fuel and lead, they also consider that the fuel would have a tendency to disperse.

The ability of molten fuel to float remains to be demonstrated, however, in particular due to the possible formation of eutectics. More generally, a number of issues remain open, regarding the phenomenology of a severe accident in an LFR. In the event of a severe accident, cooling of the molten fuel must rapidly be restored, by either natural convection or forced circulation using the available emergency systems, in order to limit structural damage. The location and configuration of the corium must be determined before the effectiveness of this cooling can be assessed. Several problems must be considered:

- Part of the corium may be entrained into the cooling loops, preventing the system from being cooled in loop-based concepts.
- As part of the corium floats to the free surface of the lead, part of the decay heat would be dissipated toward

[36] These flow limiters act like a membrane during normal operation.

the upper part of the reactor vessel. It would then be necessary to demonstrate that this power output did not significantly heat the top of the vessel, potentially leading to its destruction. Such a demonstration may be problematic, given the uncertainties associated with the various phenomena: it would first be necessary to assess the geometric configurations in which corium may be found, and then seek to understand the nature of the heat transfers in such configurations. More than 30 years after the TMI2 accident, this issue has still not been definitively resolved with respect to light water reactors. It will also be necessary to demonstrate that corium occurs in a geometric configuration that precludes any return to criticality. It does not appear possible to rule out the possibility of molten fuel accumulating at the surface of the liquid lead in a reactive geometric configuration.

- Neither is it possible to preclude the possibility of corium segregating into several phases, in a manner similar to the phenomena observed in PWRs. In such cases, corium may be only partially floatable in lead.

In the event of rupture of the main reactor vessel and the safety vessel, molten core-concrete interaction may occur, accompanied by the production of flammable gases (due to lead oxidisation and corium metallic phases), if the design does not feature an external core catcher and related cooling system. Other reactions (for example between hot lead and concrete) are also possible and should be studied.

Although physically possible, a high-energy reaction between corium and molten lead appears relatively improbable, as, in addition to the fact that lead has a very high boiling point (1800°C), the drops of molten fuel would form a crust on the surface, limiting fragmentation and preventing the reaction from escalating.

There appears to be a greater risk of a lead-water reaction in the heat exchangers (NB: the secondary system is pressurised and a break in the heat exchanger would cause water to enter the primary circuit). Priority should be given to studying this risk.

Furthermore, note that the reaction whereby lead oxidises in contact with water or air is not a high-energy reaction, which is an advantage of the LFR with regard to severe accidents.

Although the accident sequences described by the ELFR concept developers do reveal neither scenarios resulting in core meltdown nor high-energy phenomena inside the reactor vessel (caused by an SGTR, for example), it is important to note that, given the limited knowledge of certain physical phenomena involved and the qualification states of the codes used, these studies cannot at this stage be assumed to be exhaustive or of an envelope nature. Additionally, for some sequences, the structural integrity of the reactor vessel cannot be guaranteed, due to the high temperatures reached. Significant additional R&D efforts are therefore required.

5.5 ENVIRONMENTAL IMPACT AND RADIATION PROTECTION

Primary coolant composition

As for the SFR, in the event of corrosion, the metallic elements in steels would be released into the coolant, causing its viscosity to increase. These elements may form precipitates that may in turn cause fouling.

Corrosion and erosion products, together with products deriving from the activation thereof, would form impurities in the coolant, alongside the pre-existing impurities such as silver, cadmium, copper, chromium and nickel.

As with the SFR, a coolant purification system will be necessary in order to limit such impurities and the fouling or blockage risks that they might otherwise represent, depending on their solubility in lead or lead-bismuth eutectic alloy.

Furthermore, activation of lead-bismuth coolant would lead to significant production of polonium-210 (^{210}Po). Note that even with an all-lead coolant, polonium would still be created, albeit in much smaller quantities, as bismuth is a native impurity in lead.

All the aforementioned elements are included in the source term and the potential effluents.

Radiation protection and protection against toxic substances

No dosimetry data relating to the operation of this reactor type has been provided.

It should nevertheless be noted that using lead or LBE would greatly limit worker exposure to gamma radiation, compared with other coolants. The respective benefits and disadvantages of loop-based and integrated concepts are the same as for the SFR.

Worker exposure is likely to be greater with the LFR than with the SFR, as fuel handling operations are conducted with the vessel open.

In addition, as already stated for the SFR process, minor actinide transmutation and more extensive use of plutonium compared with current fuel management strategies would result in higher effective and peak doses, particularly during the cycle front-end and back-end phases.

It should also be noted that lead is highly toxic. Conventional industry is striving to eliminate lead from industrial processes wherever possible. However, due to its low vapour tension, the lead concentration inside the containment during reactor refuelling and in-service inspection operations (with the primary circuit open) appears to be reasonably low.

Polonium-210 is a high-energy (5.34 MeV α emission), short-lived (138-day half-life), volatile radio-isotope liable to partially migrate to the free surface of the primary circuit, where it may form aerosols. This would result in significant activity in the gas cover space, posing problems for in-service inspection and repair (ISIR) operations.

However, if lead is used rather than lead-bismuth eutectic alloy, polonium formation does not appear to represent a major problem from a radiation protection perspective. In this respect, total production of ^{210}Po in the lead used for the ALFRED prototype (3400 tonnes, with a target purity of 99.985%) has been estimated at 0.4 g. Furthermore, the volatile fraction of ^{210}Po is very low, due to its chemical interactions with lead. For ALFRED, the corresponding values for this fraction at temperatures of 480°C and 800°C are respectively 2.10^{-10} and 3.10^{-7}.

Liquid and gaseous waste management

With the LFR, sources of releases outside the containment include:

- aerosols and noble gases released from the gas cover space;
- reactor coolant leaks into the containment and through the containment (direct leaks).

Current understanding of fission product retention and activation in lead and LBE is limited. However, lead and LBE appear to be effective in retaining certain elements such as caesium, iodine, strontium and polonium. Retention is also enhanced by the fact that lead is unlikely to boil in accident conditions. Analyses conducted for the Eurotrans EC's FP6 and LEADER projects revealed that, in the event of core meltdown, only a small proportion of the elements present in the coolant would be vaporised in the cover gas. However, this property will be largely irrelevant if the corium is floatable.

In the event of vessel leakage, lead spreading onto concrete should solidify at its surface, potentially helping to retain the fission products. Gas bubbles generated as the concrete is degraded may escape through the crust, however, entraining certain fission products.

In the event of atmospheric release or if the basemat fails, lead, fission products, corrosion products, impurities and activation products would be disseminated into the environment, affecting the population.

Tritium production resulting from tertiary fission and activation of boron in control rods should be similar to those in an SFR with the same thermal power rating. The quantity of tritium diffused through the steel cladding may depend on the coating on the outer surface of the assemblies. The amount remaining in the coolant will depend on the ability of the lead purification system to trap part of the tritium. Lastly, the risk of tritium entering water in the steam generators will be greater than with an SFR, all other things being equal, due to the fact that there is only a single wall (the SG tube wall) between the lead and the water.

Lastly, note the toxicity risks associated with the use of lead, the formation of polonium (particularly if LBE is used) and certain impurities present in lead (native impurities, corrosion products and activation products). The use of large quantities of toxic materials is a major drawback with this type of reactor.

5.6 MATURITY OF CONCEPT AND R&D REQUIREMENTS

5.6.1 *OPERATING EXPERIENCE FEEDBACK*

The only available operating experience feedback relates to the use of propulsion units in Soviet military submarines, as stated in Section 5.1.1.

In 2007, IAEA published an experience feedback document on fast reactors (reference 77); this document includes details of notable events that occurred on three such reactors:

- 1968: several fuel assemblies were blocked by oxides of LBE (in particular), which had accumulated mainly during outages for primary circuit maintenance (while depressurised), causing partial core meltdown due to a lack of suitable procedures in response to the signals received in the control room. This accident resulted in several fatalities due to acute irradiation. The most notable provision adopted for submarine reactors after this event was a hardening of the procedures for monitoring oxygen in the primary circuit and for purification to remove any oxides formed.
- 1971: damage to primary circuit pipework was observed, consisting of corrosion of the external surface of these pipes, caused by excessive humidity in the reactor compartment, which was in turn attributable to a lack of leaktightness in the steam generator. This led to a leak of radioactive LBE.
- 1982: widespread corrosion of the steam generator tube bundle caused by poor-quality feedwater. This led to steam ingress into the primary circuit containing the LBE. At the end of a fairly complex chain of events, including human errors, 150 litres of radioactive LBE leaked into the reactor compartment.

The Soviet navy abandoned this type of reactor in the light of these accidents. It should be noted, however, that Russian agencies continued research in this field, with a view to developing a civilian power generation application, but their expertise in this area is not in the public domain.

5.6.2 *R&D REQUIREMENTS*

The principal technological difficulties identified to date with regard to the construction of a lead-cooled fast-spectrum reactor concern:

- management of corrosion and erosion risks;
- sensitivity of the concept to seismic activity, due to the high mass of lead;
- in-service inspection of the second barrier components and of the structures and equipment inside the reactor vessel, in the “integrated” concept;
- reactor coolant purification.

A number of R&D programmes addressing these areas have been defined (reference 78):

- corrosion and erosion: qualification of an austenitic steel for the reactor vessel, corrosion- and erosion-resistant equipment for the reactor internals, a protective coating for fuel cladding and steam generator tubes, and special-purpose corrosion- and erosion-resistant materials for primary pump blades;
- seismic risk: research is underway to apply seismic isolator technology to the LFR, as part of the SILER project (reference 79);
- inspection: development of the necessary techniques and instrumentation to enable in-service inspection of SG tubes, the reactor vessel, top heads and core support structures (allowing for the opacity and corrosive nature of lead and the fact that the instrumentation will have to be qualified as irradiation-resistant);

- replaceability: study to examine the possibility of replacing components inside the reactor vessel during the life of the facility;
- purification: development and validation of a lead purification technique for use prior to reactor vessel filling and during operation, to prevent the formation of aerosols (and related risk of blockage of fuel assemblies), and the development and validation of oxygen monitoring techniques in order to control the oxidation process.

Other R&D initiatives have been identified, some of which are in progress, to gain a clearer understanding of reactor behaviour in accident conditions:

- phenomenology of steam generator tube rupture accidents: preliminary experimental tests (at the LIFUS facility, ENEA Brasimone) have been carried out in order to gain adequate knowledge of the phenomena liable to occur in such conditions and to qualify the relevant design codes; other experimental and design activities are planned, using an SG mock-up at the ATHENA experimental facility (ENEA Brasimone); there are plans to build a full-scale model of a double wall SG in the near future;
- behaviour of molten core in the primary coolant;
- definition of the quantity of fission and activation products liable to be present in the cover gas following core meltdown: this issue represents a major challenge for the LFR (and SFR). An R&D effort is necessary, for example by examining the sequence whereby fission products from the fuel are released into the coolant and then from the coolant into the cover gas;
- risk associated with lead freezing: experimental analyses and calculations are underway to assess the risk of severe fuel degradation in the event of subsequent thawing; a design change to the DHRs is currently being investigated (MAXIMA project) with a view to preventing lead freezing with no need for operator intervention.

Lastly, there are plans to conduct research relating to fuel loading operations with the coolant at a temperature in the region of 400°C, and the strength of structures subjected to high temperatures in an irradiating environment.

5.7 CONCLUSION

The LFR concept has been shortlisted by GIF, which considers it to be well placed to satisfy the stated objectives (more effective use of uranium resources, enhanced safety, limitations on nuclear proliferation, etc.), largely as a result of the advantageous chemical, thermal-hydraulic and nuclear properties of lead.

Firstly, lead causes violent chemical reactions with neither air nor water, prompting some designers to install steam generators directly inside the primary reactor vessel. IRSN considers, however, that this design choice would probably not be approved in France, for a number of reasons (risk of water ingress into the reactor vessel, unsuitability for in-service inspection, non-repairability, etc.). An intermediate cooling circuit would appear to be essential for integrated concepts.

The nuclear properties of lead (low moderating and neutron absorption coefficients, and high reflecting capacity make it possible to design reactors with a low power density (~100 MW/m^3); this is essential as the reactor coolant flow velocity must be limited, due to the erosion risk.

The high boiling point at atmospheric pressure (1745°C for lead and 1670°C for LBE) provides large margins in normal operation, precludes the risk of coolant boiling and therefore limits the risk of reactivity insertion due to boiling-related void effects. However, it should be noted that structures would be destroyed at significantly lower temperatures.

Lastly, the grace periods in the event of a loss of cooling are particularly long, due to the large volume of lead and its high density, giving the LFR high thermal inertia. Furthermore, the low pressure loss in the core (also due to the need for a low reactor coolant flow velocity) and the high density and expansion coefficient of lead are conducive to the formation of natural convection conditions in the core.

The studies conducted as part of the LEADER project for the industrial-scale ELFR did not identify any scenarios leading to generalised core meltdown; however, IRSN remains highly sceptical regarding these conclusions, as, firstly, the studies performed were not demonstrated to be exhaustive and, secondly, the level of understanding of the physical phenomena involved and the qualification status of the codes used in the studies were limited. The RNR-Na Project shares this view.

In this respect, no studies relating to the phenomenology of core meltdown accident are currently available; the designers' assumptions regarding the floatability and dispersion of the molten fuel, which would prevent any return to criticality and vessel rupture are not currently supported by any research.

In any case, IRSN considers that, for defence-in-depth reasons, severe accident scenarios resulting from fuel melt or destruction of reactor core structures should be taken into consideration.

The major difficulty with a lead- or LBE-cooled fast reactor relates to the corrosive and erosive effect of the coolant on stainless steel structures. The only process currently available for limiting corrosion risks consists in forming a surface layer of iron oxide on structures. Controlling this process appears to be a significant challenge, which would impose severe operational constraints in terms of operating temperatures and reactor coolant purification. Furthermore, the effectiveness of such a process remains to be confirmed, particularly for a large reactor and for all operating conditions. The longer-term solution under consideration, consisting in creating a coating on steel surfaces, would simplify the lead chemistry management issue and enable higher temperatures to be reached. Controlling the risk of lead freezing also implies maintaining a relatively high minimum operating temperature, including during outages.

As well as the challenge of managing the composition of the reactor coolant, the difficulty – even more marked than with the SFR – in inspecting structures and equipment immersed in the lead and the second barrier must be addressed. The LFR concept cannot be considered acceptable from a safety perspective until this issue has been resolved.

Furthermore, the LFR concept also inherently suffers from two other major drawbacks in terms of safety and worker protection: great sensitivity to seismic activity and the significant toxicity of lead and derivatives, including polonium, particularly if LBE is used.

Earthquake sensitivity and the difficulty in managing the oxygen concentration suggest that this concept should only be implemented in small reactors.

Ultimately, IRSN considers that any verdict on the ability to achieve a significantly higher safety level than targeted by Generation III reactors would be premature at this stage.

6. MOLTEN SALT REACTORS (MSR)

Molten salt reactors (MSR) include:

- reactors in which salt is both coolant and fuel; in such reactors, the fuel is dissolved in a molten eutectic alloy salt; thus the heat is produced directly in the coolant. The term “fuel salt” is used in this report to describe this concept;
- reactors that use a solid fuel and a molten salt coolant.

Sections 6.1 to 6.7 relate to reactors in which fuel is dissolved in salt. Some concise information relating to solid-fuel MSRs is provided in Section 6.8.

6.1 HISTORY AND PROSPECTS FOR DEVELOPMENT

6.1.1 *BACKGROUND*

To date, two experimental molten salt reactors have been 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. This reactor was intended for use as a ground-based prototype for a military aircraft engine. It was a thermal-spectrum reactor using beryllium oxide as a moderator. It operated for around 100 hours.
- 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. This was also a thermal-spectrum reactor, consisting of a graphite block used as a moderator, through which channels were made in order to carry the fuel salt. This reactor did not use a fertile material but ^{235}U-based fuel, plutonium and then ^{233}U. This reactor, which had a power output of approximately 8 MWth, was shut down in 1969 after around 13,000 operating hours. The reactor building is unchanged since December 1969, when the reactor was decommissioned, and the fuel salt remains in situ, with no reprocessing, and in solid form. Dismantling operations are planned but had not yet begun in September 2010.

A project for an industrial-scale (2500 MWth) prototype thermal-spectrum reactor known as the Molten Salt Breeder Reactor (MSBR) was subsequently studied in the United States but was never built.

6.1.2 *ONGOING PROJECTS AND PROSPECTS FOR DEVELOPMENT*

Many countries maintain an interest in MSR technology even though no such reactors are currently under construction:

- Europe: MSR development projects have been carried out under Research and Development Framework Programmes since 2001. In particular, the MOlten Salt Reactor Technology (MOST) project identified the key issues that must be studied if the MSR concept is to be developed: salt cleaning processes, reprocessing schemas, extraction of gaseous fission products and noble metals, tritium management, component integrity and salt property monitoring.

- In 2007, the ALISIA European initiative was launched, with support from Euratom. The aim of this initiative was to strengthen the European network of expertise in the MSR field and share the results of national programmes. ALISIA sponsored the creation in 2011 (as part of the EC's FP7) of the EVOL (Evaluation and Viability Of Liquid fuel fast reactor system) project, which featured an initiative focusing on the design and safety of this type of reactor.
- France: CNRS built on the research into the Molten Salt Breeder Reactor (MSBR) initiated by ORNL and relayed in France by CEA and EDF. CNRS subsequently turned its attention to the Molten Salt Fast Reactor (MSFR) concept for a 3000 MWTh fast-spectrum reactor. This project is being conducted within an international framework featuring GIF and European projects in collaboration with Russia.
- Russia: Russia has been operating an MSR programme since the 1970s, with the aim of improving the MSBR concept. This research resulted in the definition of the MOlten Salt Actinide Recycler and Transmuter (MOSART) reactor, designed to incinerate transuranium elements (i.e. plutonium and minor actinides). MOSART was developed under the aegis of the International Science and Technology Center (ISTC), which brought together scientists from Russia and other former Soviet republics, Canada, Europe, Japan, the Korean Republic and the United States. The research effort is led by the Kurchatov Institute in Russia; MOSART is a 2240 MWth reactor designed to operate exclusively using transuranium elements (i.e. not uranium or thorium); it must be able to incinerate minor actinides. This reactor has an epithermal neutron spectrum. Alternative configurations are being studied for a thorium-cycle breeder reactor. This research is being carried out through the Minor Actinides Recycling in molten Salt (MARS) project led by a Russian company, Rosatom, in association with the EVOL project.
- China: In 2011, the Chinese academy of science launched a high-priority strategic research programme on MSRs, with particular focus on the Thorium Molten Salt Reactor (TMSR) project, with the aim of building a thermal-spectrum molten salt research reactor by 2017, and a 100 MWe demonstrator by 2035.
- Japan: Japan is interested in thermal MSRs and is conducting several projects under the generic FUJI banner;
- USA: American research efforts are focused on liquid salt-cooled high-temperature reactors (see Section 6.8). However, some R&D fields overlap with R&D relating to reactors that use fuel salt; examples include chemistry, materials and the high-temperature behaviour of liquid fluoride salts.

6.2 MAIN CHARACTERISTICS

Unless otherwise stated, the information in this chapter relates to the MSFR concept developed by CNRS. Note that GIF's revised roadmap (reference 2) states that this is henceforth the reference concept for its research.

6.2.1 *GENERAL DESIGN*

The MSFR is being developed with a thermal power of 3000 MW and an electricity generating power of 1300 MW, representing an efficiency of 43%.

In the current MSFR design, the reactor contains 18 m^3 of fuel salt, of which 9 m^3 are present, during normal operation, in a cylindrical "central cavity". The reactor produces heat in the fuel salt as it circulates upwards through this "central cavity". The reactor has a power density of approximately 330 MW/m^3.

Core reactivity depends on the temperature of the fuel salt and the quantity of fissile material in the central cavity ; unlike other projects such as the MOSART project, the MSFR does not feature control rods. Furthermore, using a liquid fuel should make for a more uniform fuel burn-up than in other reactors. However, the distribution of fissile material in the central cavity and the power in the fuel salt appear to be hard to control, potentially resulting in hot spots in the systems.

The fuel salt is surrounded by a fertile blanket of thorium fluoride and lithium. This blanket is contained in a Hastelloy annular tank, which if necessary may be equipped with heat shielding on the fuel salt side. The fertile blanket is cooled by a dedicated system. Above and below the central cavity there are upper and lower plenums as well as reflectors. These are solid components shaped to facilitate coolant flow; like all surfaces in contact with fuel salt, they may be fitted with thermal shielding. To the sides, the fertile blanket provides a screening effect, although additional B_4C protection is planned. These elements can be seen in Figure 35 which shows a conceptual diagram of the MSFR.

The fuel salt temperatures at central cavity inlet and outlet should be 700°C and 770°C, respectively, with a coolant at low pressure (< 0.5 MPa), as the boiling point is high (1800°C). The choice of a fuel salt containing fluorine limits structural damage due to the high effective inelastic diffusion cross-section of fluorine for neutrons with an energy of a few hundred keV.

Fuel salt circulates through 16 primary or recirculation loops, each equipped with a pump, salt/gas separator, heat exchanger, gas injector and draining system. Cold fuel salt from the recirculation loops mixes in the lower plenum and hot fuel salt is mixed upon exiting the central cavity and then returned to the 16 primary loops. A full fuel cycle is completed in a few seconds (3-4 seconds in nominal operating conditions) and an overflow system (in which a cover gas fills the space above the liquid) accommodates any changes in the level of fuel salt resulting from temperature fluctuations.

Heat will be removed from the recirculation loops by an as-yet undetermined number of intermediate loops, each equipped with a circulation pump and containing inert salt with a melting point between 250°C and 450°C (for example FLiNaK) or lead[37]. Lastly, a secondary cooling system, possibly using supercritical water, would convert this energy.

[37] For the purpose of this document, the intermediate loops are assumed to contain inert salt.

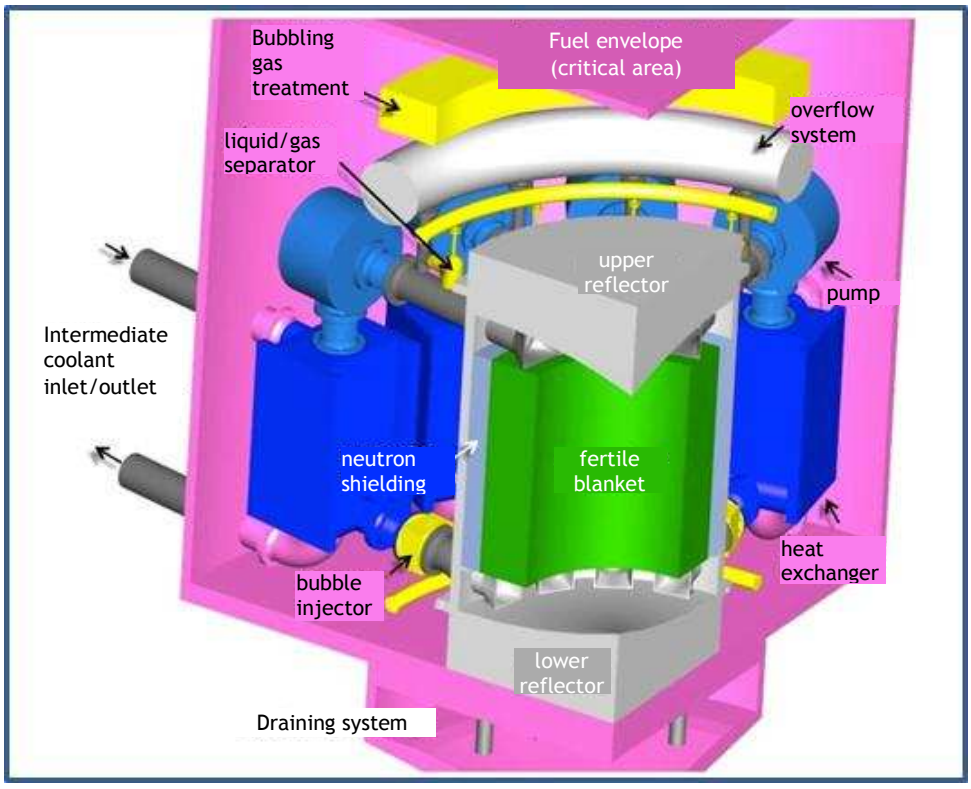

Figure 35: Conceptual view of the MSFR

A neutral gas injection and extraction system (shown in yellow in Figure 35) is connected to each recirculation loop; gas bubbles are injected as the coolant exits the heat exchangers. The salt is returned to the pump after the liquid/gas separation operation. This bubbling process continuously purifies the fuel salt by stripping suspended solid particles and dissolved gaseous fission products. More detailed explanations are provided in the "On-site treatment unit" section below.

The fuel system, consisting of the central cavity and the 16 recirculation loops, the fertile blanket and related cooling system and the bubbling system together form the "**critical area**".

The concept also features a fuel salt draining system located under the lower reflector; this system retrieves salt (via a manifold) from the low points in the fuel system. In the event of loss of normal cooling, salt can be rapidly gravity-drained into dump tanks ("**subcritical area**"), the geometry of which is designed to prevent any risk of criticality and facilitate decay heat removal. Draining is initiated by operating valves or, passively, when solidified salt plugs melt. This "subcritical area" is submerged in a water-filled pool, which is in turn cooled by a system of water/air heat exchangers (Figure 37). A stockpile of poisoned solid inert salt is located below the subcritical area. This is a passive solution for long-term conservation of the fuel in the event of a leak from the subcritical area.

Lastly, there is a **"storage area"** where salt can be held during extended outages (longer than one to two weeks). This area is also where salt is conditioned prior to startup. It enables draining system maintenance to be performed if necessary.

On-site salt treatment unit

MSRs are connected to a dedicated fuel salt treatment unit. The size of the treatment unit depends very heavily on the neutron spectrum. With a thermal-spectrum reactor, continuously removing the fission products that form the neutron poisons is essential, failing which the chain reaction may be interrupted. With a fast-spectrum reactor, treatment unit and reactor operation are not so closely correlated, as there is less risk of the core being poisoned by neutron-absorbing fission products. The principal purpose of treatment is to maintain the chemical properties and characteristics of the fuel salt, and in particular its redox potential. Such units are able to operate in discontinuous mode (collecting a few tens of litres per day), and this is the case with the MSFR.

Figure 36 shows a diagram of the treatment unit incorporated into the MSFR.

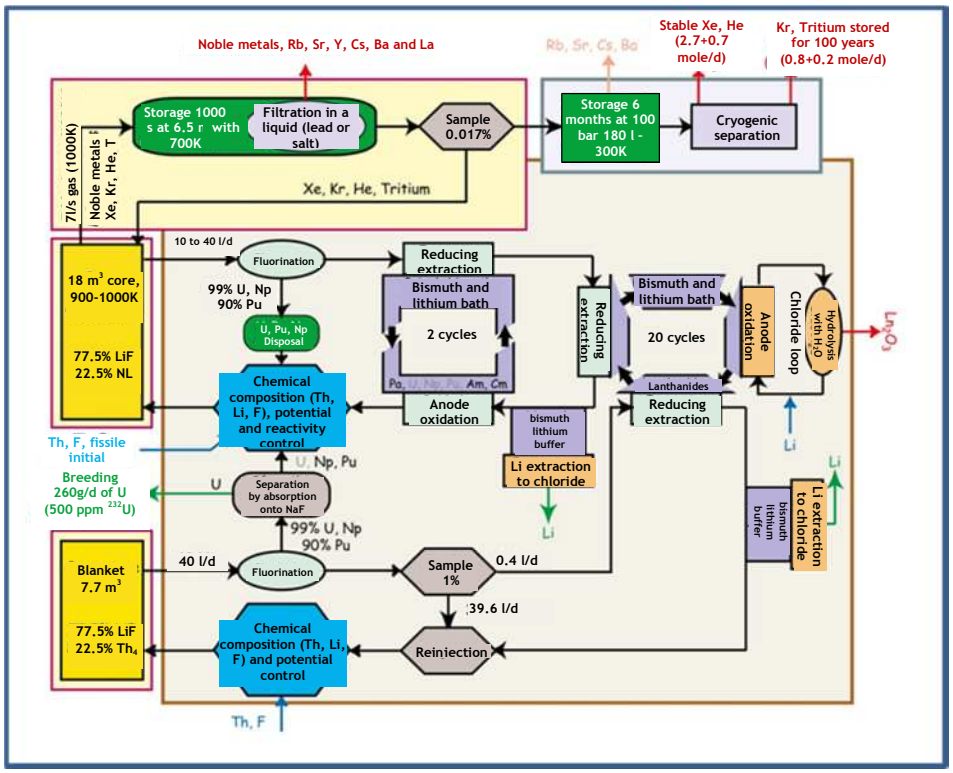

Figure 36: Overview of the MSFR treatment unit

The treatment unit has two distinct parts:

- the aforementioned neutral gas injection and insoluble and gaseous fission product extraction system (using a bubbling process), which aims to decrease the formation inside the reactor of daughter products from decaying gaseous isotopes (alkaline or alkaline earth elements such rubidium (Rb), strontium (Sr), yttrium (Y), caesium (Cs) and barium (Ba)) and prevent particles of noble metals from being deposited in cold spots in the reactor. The core has closed-circuit gas sweeping at a flow rate of around 7 litres/s. The gas thus collected is separated in the liquid/gas separator. It is then stored in a tank (for a duration for 1000 s) while being continuously filtered for an hour using a liquid metal. The gas is then stripped of any solid particles of metallic elements and alkaline or alkaline earth elements by the fission gas decay process; the elements thus recovered accumulate in the filter (the life and conditioning of which are yet to be determined). The filtered gas is then returned to the reactor core. A very small proportion of this purified gas is nevertheless removed, to ensure that the total quantity of gas in the bubbling system remains constant, allowing for fission gas production. The gas thus removed is stored for approximately six months, to allow the gaseous isotope decay

process to be completed. This step may be followed by cryogenic separation, to separate stable xenon and helium and recover krypton isotope 85 and tritium - two radioelements that require even longer decay periods (approximately 100 years) before they can be released;

- a pyrochemical reprocessing system that collects 10 to 40 litres of fuel salt daily, transfers it to a "processing area" and then reinjects it when the fission products (lanthanides, etc.) have been removed. The quantity of salt reprocessed in the MSFR represents approximately 0.2% of the total fuel salt volume, compared with 10% in the case of the MSBR thermal-spectrum reactor. Chemical treatment separates fissile elements and minor actinides by successive fluorination and reduction extraction in contact with a liquid metal such as bismuth (Bi). Thorium remains in the salt. All the aforementioned radionuclides are then reinjected into the primary circuit after adjusting the salt composition to maintain the core isotope inventory. Lanthanides (the most highly neutron-absorbing fission products) recovered after processing are packed and stored in oxide form (Ln_2O_3) prior to disposal. A similar system with a single fluorination step exists for the blanket salt.

Aspects specific to the MSFR fuel cycle are addressed in Section 9.6.

6.2.2 *GENERAL SAFETY OPTIONS*

The MSFR combines the characteristic benefits of fast reactors in terms of resource use and waste reduction (such reactors can operate in breeder, isogeneration and burner modes, and are potentially able to effectively transmute and recycle long-lived radionuclides with an "integrated" fuel cycle) and the advantages associated with using a liquid fuel (uniform burn rates, near-immediate neutron feedback, etc.).

The design options must also reflect the drawbacks associated with the use of salts (high solidification temperature, low thermal inertia, etc.).

The main safety function-oriented design options reflect these characteristics:

- develop materials resistant to high temperatures, irradiation, erosion and corrosion caused by salts;
- define the geometric characteristics of the fuel system with a view to controlling salt reactivity, taking into consideration the intended operating temperatures and the changes in fuel salt composition over the course of the operating cycle (which requires the strong relationship between the nuclear, thermal-hydraulic and thermochemical aspects to be taken into account);
- design a fuel salt emergency draining system to enable decay heat removal from the fuel during outages and in accident conditions, and to keep the salt subcritical;
- adopt measures to maintain the salt temperature above its solidification point in all foreseeable conditions;
- define a confinement strategy with appropriate consideration for the chemical toxicity of the salts used and the products generated during reactor operation and in the treatment unit.

As a matter of principle, the design should reflect the need to perform the safety functions, regardless whether the fuel salt is located in the critical area, the subcritical area, the storage area or the treatment area.

In the case of thermal-spectrum MSR concepts, the design options must also reflect the greater requirements in terms of fuel salt purification and processing (poisoning risk, need to reprocess large volumes of salt each day, etc.).

6.2.3 *FUEL AND COOLANT*

6.2.3.1 Fuel salt selection criteria

The fuel salt, which serves a dual power generation and heat removal function, must satisfy multiple criteria (see Volume 4 of reference 12 and reference 80); in particular, it must:

- have a sufficiently high boiling point;
- have a moderate solidification temperature;
- not contain elements with a high neutron-absorbing factor, particularly in the case of thermal-spectrum reactors;
- not generate hard-to-manage radioisotopes;
- have a high expansion coefficient in order to harness strongly negative feedback coefficients and the potential for natural convection;
- enable fissile and fertile elements to readily dissolve;
- remain stable when irradiated;
- have good heat exchange characteristics;
- not adversely affect the strength of structural materials.

The intention is to use inert chloride or fluoride salts, but chlorides produce large quantities of chlorine isotope 36, which is a hard-to-manage waste product. The fluorides under consideration include FLiBe (a mixture of LiF and BeF_2), which was used in the MSRE, and FLiNaK (a mixture of LiF, NaF and KF). Other fuel salts may also be used.

6.2.3.2 Fuel salt adopted for MSFR

Lithium fluoride (LiF) is the salt selected for the MSFR, to avoid the presence of elements such as beryllium, sodium and potassium, which would complexify the necessary processing.

The solubility limit of plutonium in LiF (approximately 6%) restricts the choice of fissile material.

The chosen fuel salt consists of a mixture of ^{233}U fluoride (fissile material), thorium fluoride (fertile material) and lithium fluoride. The lithium is very highly enriched with isotope 7, which, unlike isotope 6, is a very poor neutron absorber. The fuel salt has a crystallisation temperature of 585°C, requiring high operating temperatures (see Section 6.4.1.3).

Around 5000 kg of ^{233}U is required at reactor startup. As ^{233}U is present only in trace amounts in the current fuel cycle, other fissile isotopes would have to be used for the initial fuel loads, with the corresponding constraints:

- using exclusively ^{235}U necessitates enrichment in excess of 20%, which may pose a problem with regard to the risk of proliferation; such use may be considered in combination with the following solutions, however, which would limit the Pu content;
- using transuranium elements, including minor actinides from existing PWRs, requires the quantity of plutonium to remain below 6% (as stated above), to prevent the formation of plutonium crystals, due to its limited solubility in LiF;
- using thorium mixed oxide (MOXTh) implies irradiating (in PWRs) a fuel consisting of plutonium and thorium isotope 232 (the latter in place of depleted uranium). This fuel would then be suitable for use in liquid form in

the MSFR, as the quantity of plutonium would be within its solubility limit.

The MSFR must therefore breed ^{233}U (see Appendix III). Breeding is achieved by incorporating thorium-containing fertile blankets into the design.

6.2.4 *BARRIERS*

The MSFR concept features three barriers.

6.2.4.1 First barrier

In normal operation, the structures potentially in contact with fuel salt are very complex. Consequently, the first barrier is formed not by these structures, but by a much simpler "envelope" (known as the **"fuel envelope")** that surrounds all the structures in contact with fuel salt, including the dump tanks and the bubbling system. The fuel envelope only comes into contact with the salt in the heat exchangers, which do form part of the first barrier.

The fuel envelope (shown in pink and purple on Figure 37) encompasses several different, not necessarily fully contiguous areas: the critical and subcritical areas, the draining system manifold and transfer channel (shown in pink) and the fuel salt processing and storage areas that handle discontinuous batches of salt ("treatment area" and "storage area" – shown in purple). It is penetrated by the hot and cold legs of the intermediate cooling circuit. In the MSFR, much of the first barrier is exposed to a weak neutron flux and operates at low pressure. It must be able to withstand high temperatures (700°C at the hottest points) and the effects of direct interaction with salt. The salt temperature in the upper part of the central cavity may not be totally uniform, resulting in hot spots or temperature fluctuations. Accordingly, heat shielding must be installed, cooled by deviating the intermediate coolant, to limit the temperature of the metallic materials that keep the fuel system leaktight.

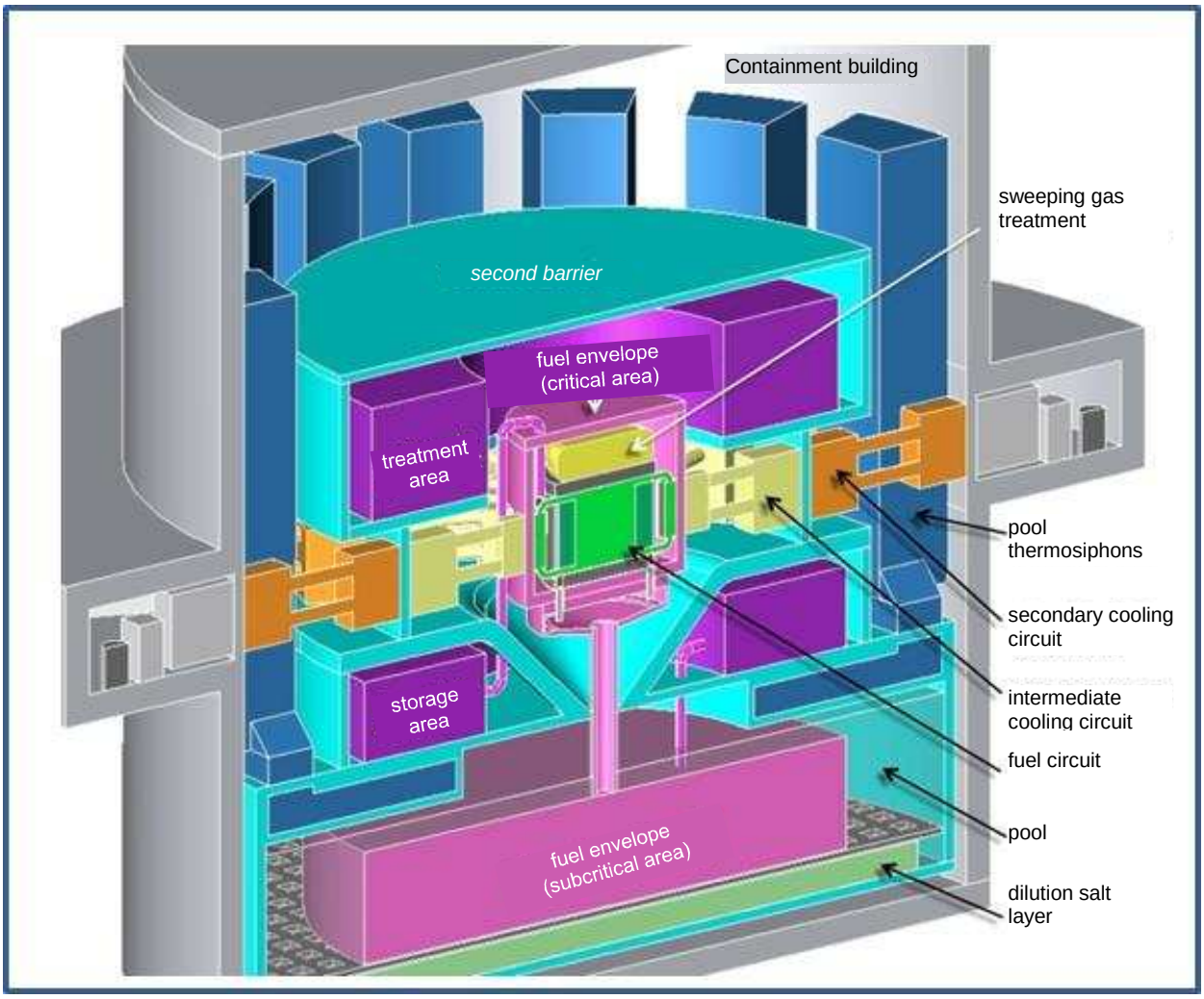

Figure 37: Overview of the MSFR and barriers

First barrier monitoring

The means of detecting fuel salt leaks have not been fully defined at the time of publication, but activity measurements able to detect any introduction of fuel salt into the intermediate salt are already planned.

In-service inspection of the various areas is possible, subject to transferring the fuel salt to a suitable location. When the reactor is in operation, it should be possible to inspect the subcritical area and auxiliary areas by remotely operated means. The critical area and the equipment installed in it can be inspected with the reactor shut down. Lastly, when the fuel salt is held in solid form in the tanks in the storage area, the entire reactor can be inspected.

6.2.4.2 Second barrier

The second barrier (shown in pale blue in Figure 37), the boundaries of which have yet to be finalised, is likely to be simpler than the first barrier.

It surrounds the envelopes that form the first barrier. The lower part of this barrier consists of the fuel envelope cooling pool in the subcritical area. The heat exchangers between the intermediate and secondary cooling systems are part of the second barrier, as are the water/air heat exchangers that cool the pool.

6.2.4.3 Third barrier

The third barrier consists of the reactor containment, which will probably be the reactor building. It contains part of the secondary cooling system, which could carry non-radioactive water at high pressure (>22 MPa).

6.3 CONTROL OF SAFETY FUNCTIONS

6.3.1 *REACTIVITY CONTROL*

Note that the description of reactor neutron behaviour in this section is based on calculations performed using simplified models (reference 81). These initial calculations will need to be confirmed by more detailed studies.

The geometry of the central cavity is defined such that criticality is reached at the nominal temperature for a specified salt composition.

The highly negative feedback coefficients, the release of fission energy directly into the coolant and the speed at which the fuel salt circulates (approximately 4 seconds) together enable the reactor to automatically and rapidly stabilise itself, depending on the power demand: a decrease in flow rate automatically causes reactor fission power to decrease. Consequently, unlike with other concepts, there are no control rods in the MSFR.

If less power is removed, the temperature of the fuel salt increases. As the salt expands considerably more than the constituent materials of the central cavity, some of the fuel salt is forced out of the central cavity into dedicated expansion (or overflow) tanks. The decrease in density of the fuel salt enhances its neutron transparency, which allows more neutrons to leak from the central cavity. This density effect is amplified by a Doppler effect, helping to decrease power output and therefore the salt temperature. As a result of these characteristics, the reactor returns to a stable critical state within a few tens of seconds when the demand of power changes (see Figure 38).
The mean core temperature stabilises itself at the nominal value, which is the mean operating temperature of the reactor irrespective of the power output[38] (as long as the core geometry and salt composition remain unchanged).

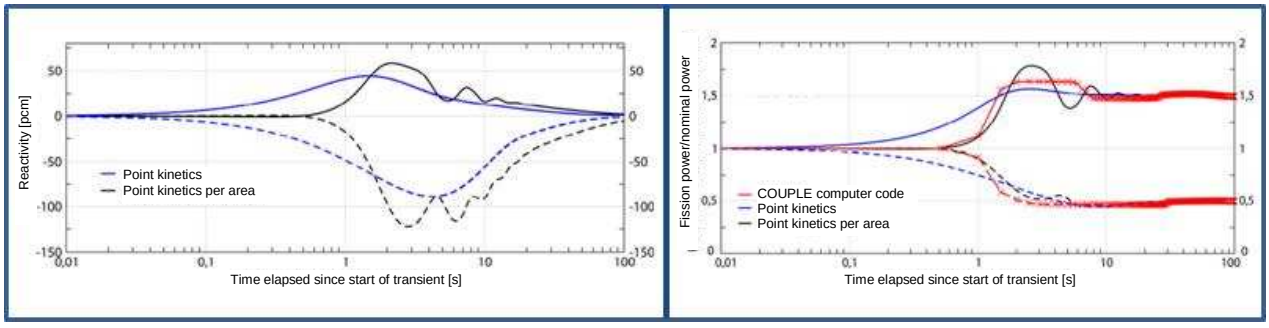

Figure 38: Variations in reactivity and fission heat calculated for instantaneous variations in heat removal from 100% to 150% (solid lines) and 100% to 50% (dotted lines) using various models

[38] The mean core temperature might, however, stabilise at a slightly different value due to reactivity insertion resulting from a change in the proportion of delayed neutrons emitted in the central cavity, which depends on the salt flow velocity profile.

These characteristics theoretically allow load tracking at rates of up to a 50% change in power in 10 minutes. The relative change in the parameters that enable reactor power to be modified must be controlled, to limit fatigue in materials. A special procedure will be defined accordingly. The following parameters are concerned:

- fuel salt circulation velocity, controlled by adjusting the fuel system pumping power;
- intermediate salt circulation velocity, controlled by adjusting the intermediate cooling circuit pumping power;
- inlet temperature of the intermediate salt in the heat exchangers.

If the power removed is less than the power generated as the fission products decay, the core automatically becomes subcritical[39]. Decay heat removal is still necessary, to prevent the materials in the subcritical area from overheating. If heat produced in the core cannot be removed, the only solution is to drain the salt into the subcritical area.

In the event of a change in reactivity inside the core (due to a change in core geometry or salt composition, for example), the feedback coefficients stabilise reactor power at a higher temperature, if reactivity has increased, or a lower temperature if the reactivity has decreased (see Section 6.4.2.1).

The subcritical approach, as implemented in current reactors during the startup phase, is not possible, due to the absence of control rods. Criticality is initiated when the fuel salt has been transferred to the fuel system and the circulation pumps started. If, after loading, the reactor is extremely subcritical, it can be difficult to achieve criticality. In such conditions, it would be necessary to either inject very large quantities of fissile material (1 kg of ^{233}U increases reactivity by only 10 pcm) or greatly reduce fuel salt temperature. This would appear difficult to achieve, due, firstly, to the limitation by design of the rate at which fissile material can be injected into the fuel system, and secondly, to the minimum temperature that must be maintained in order to prevent any risk of the salt solidifying. As a result, the concentration of fissile materials to be injected in the fuel system must be calculated and adjusted such that the anti-reactivity margin is not excessive. For the first reactors built, this concentration will be difficult to determine due to the significant uncertainties surrounding the effective cross-sections of the elements contained in the salt.

Lastly, in the MSFR, there is no need to have burnable poisons to offset reactivity variations due to changes in the fuel salt composition over time, as it is possible to inject fissile material into the salt.

Fraction of delayed neutrons

The effective fraction of delayed neutrons is approximately 330 pcm, although it should be noted that the “useful” fraction of delayed neutrons (i.e. those emitted inside the central cavity) depends on the fuel salt circulation velocity; this fraction represents around 50% of the total effective fraction of delayed neutrons.

[39] If this occurs, there is no need for fission power to satisfy the power demand.

6.3.2 *DECAY HEAT REMOVAL*

The decay heat removal function must be performed regardless of fuel salt location and composition, which changes over time (see Figure 39). Furthermore, not all decay heat is released in the fuel salt, as some of the fission products (actinides and thorium) are present in the bubbling and treatment units. Decay heat removal systems must therefore be designed to handle the various possible configurations.

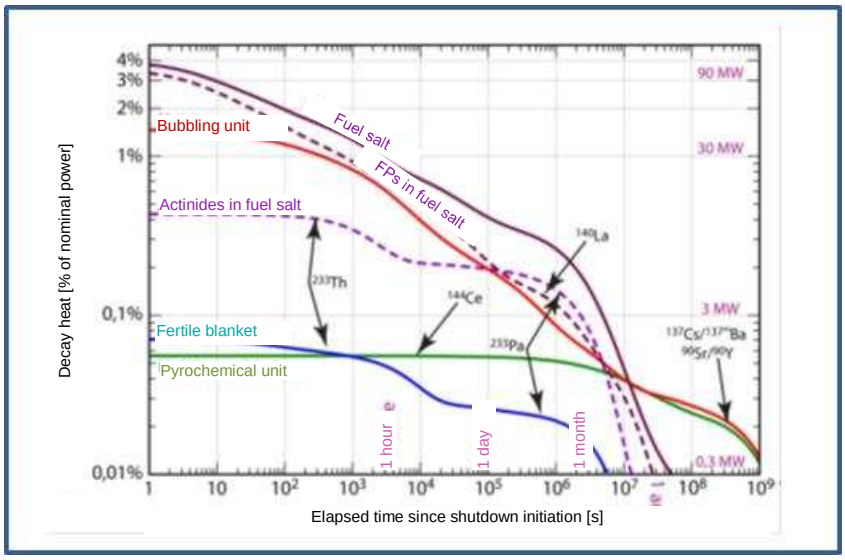

Figure 39: Decay heat in the various radioelement-containing coolants, from full fission shutdown in an MSFR in steady-state conditions

In the event of a planned outage (for inspections, maintenance, etc.), the reactor will be held at low power (i.e. a few per cent of nominal fission power) for a few hours, to reduce decay heat before draining to the subcritical area. The aim is to reduce fuel salt heating in the critical area prior to draining, to limit structural stresses. There is also the issue of removing decay heat during the start-up phase, while the fuel system is being filled. The operator has only a limited period of time, which depends on the decay heat generated by the injected fuel salt, in which to start the circulation pumps and avoid structural damage.

In case of loss of the normal heat removal means (i.e. the salt coolant/intermediate coolant and intermediate coolant/secondary coolant heat exchangers), the core becomes subcritical and the only solution for removing decay heat is to drain the fuel salt into the subcritical area. Based on the available assessments, the fuel salt would have to be drained within around 10 to 20 minutes in order to prevent damage to the parts of the fuel system not fitted with heat shielding. This time may vary, in particular according to the rate of loss of cooling versus time, fuel salt thermal inertia, heat removal by natural convection (where applicable) and the presence of heat shielding on structural components.

After draining, decay heat is removed by a pool containing several thousand cubic metres of water at ambient temperature, in which the subcritical area is immersed. Water in contact with the walls of the subcritical area is heated and evaporates. The steam thus formed is condensed in steam-water heat exchangers that operate in natural convection mode. The heated water is in turn cooled in water-air heat exchangers by a thermosiphon (chimney) effect. Ultimately, passive cooling for the subcritical area is provided by the outside air, with three barriers between the fuel salt and the environment (i.e. the walls of the subcritical area, the steam-water heat exchanger and the water-air heat exchanger). The volume of water in the pool is calculated such that the heat to be removed is less than the heat removal capacity of the water-air heat exchangers when its temperature reaches 90°C.

6.3.3 *CONFINEMENT OF RADIOACTIVE MATERIALS*

The radioactive material confinement strategy must reflect reactor specificities:

- extended and some cases complex barriers (see Section 6.2.4), the definitions of which are still subject to change, culminating at the treatment unit;
- a radioactive inventory subject to changes in terms of quantity, location and composition (radionuclide spectrum);
- multiple auxiliary systems, some of which may carry gases:
    - fertile blanket coolant system;
    - fluid treatment system used when processing gas and fission products collected by the bubbling process;
    - storage, compression and treatment system for the cover gases at the surface of liquids;
    - system corresponding to branches of the intermediate coolant system needed to limit heating in certain parts of structures.

Particular attention must be given to the fluid system required for the bubbling system, inasmuch as the fluid used to process the gas gradually concentrates the fission products stripped from the fuel. It may therefore be a significant source of decay heat.

The radioactive material confinement strategy must be defined globally for the facility, with due consideration for the treatment unit and the products (in particular hydrogen fluoride) resulting from the processes used.

6.4 RISK ANALYSIS

6.4.1 *RISKS INHERENT TO THE USE OF SALTS*

6.4.1.1 Structural corrosion

Defining and controlling the risk of corrosion of structures in contact with fuel salt are key challenges for the MSR safety demonstration.

In the MSFR, the fluorides in the fuel salt cause metal alloys to oxidise and prevent the formation of a protective passivating layer. Intragranular corrosion may also occur, caused by some fission products such as tellurium isotopes, which appear in the salt during reactor operation.

R&D into salt-related structural corrosion has been carried out since the 1950s, particularly in Japan and USA. Corrosion is a complex phenomenon influenced by multiple parameters, including salt composition, purity and oxygen content, as well as electrochemical effects between the salt and metals.

A process for managing the corrosion risk by controlling the salt redox potential was developed for the MSRE (see Section 6.6.1). The MSFR designers consider that such a process might be appropriate. The redox potential would be measured in the removed salt and adjusted as necessary by modifying the composition of the salt injected into the reactor. The feasibility of this type of control is yet to be validated for the MSFR, however.

In any case, it can be assumed that fully understanding corrosion mechanisms requires a great deal of additional R&D, in order to better assess the corrosion resistance of the various metallic materials liable to be used in an MSR.

6.4.1.2 Chemical and thermodynamic interactions between salts and water

Chemical interactions

Fluorides are hydrolysed in contact with water vapour, producing hydrogen fluoride (HF), a highly toxic, corrosive gas. No dedicated research has been conducted into the possible chemical reactions between water and fluorine-containing fuel salt, but the behaviour of fluorinated steel-making slag reveals that hydrolysis only occurs within a restricted temperature range.

Upon coming into contact, water and fuel salt are expected to form solid oxyfluorides, which could form precipitate and concentrate fissile elements. No data relating to this topic is available, but precipitation of oxyfluorides of heavy metals is known to be possible.

Thermodynamic interactions

The high operating temperature of the fuel salt (in excess of 700°C) raises the fear of a risk of thermodynamic interaction between fuel salts and water (such as a steam explosion). No relevant experimental data is available at the time of publication. However, the conditions appear relatively conducive to steam explosions, for at least two reasons:

- The low density of the salts allows them to fragment into relatively large drops (as a result of which, crystallisation should be limited, and little steam would be generated during the pre-mixing phase); in addition, the similar densities of salt and water facilitate mixing; taken together, these two arguments account for the fact that much more powerful explosions were reported during experiments using alumina than during experiments with UO_2/ZrO_2 corium.
- Temperatures of around 700°C may result in unstable transitional boiling conditions, which are a major trigger for steam explosions, as has been demonstrated by extensive experimentation with tin.

6.4.1.3 Salt crystallisation risk

The salts used in the reactor solidify at high temperature (585°C). In view of the upper temperature limit imposed for structural integrity reasons (700°C), the operating temperature range is relatively narrow.

The risk of salt crystallising and pipes jamming must therefore be studied, in particular with regard to the decay heat removal function (see Section 6.3.2). The transient calculations currently available reveal the appearance of cold spots in the systems; these cold spots may prompt salt to crystallise causing obstructions, particularly in small-diameter pipes and heat exchanger tubes. The most recent studies conducted for the EVOL project show that, in case of loss of all fuel salt circulation pumps, salt would solidify in the heat exchanger tubes in less than one minute. A fuel salt circulation outage would cause the temperature of the salt in this system to increase significantly, potentially damaging some components.

To counter this risk, the current designers intend to implement heat exchanger bypass systems to maintain a minimum circulation flow.

6.4.1.4 Risk of reprocessing system-related effects

The presence of the reprocessing system in the vicinity of the reactor is a hazard source. The system safety analysis will identify the design load cases for the various parts of the facility, in particular in terms of pressure and toxic releases. The overall aim is to prevent any risk of an accident in one part of the facility triggering a domino effect.

6.4.2 *ACCIDENT SEQUENCES*

A variety of malfunctions are liable to affect MSFR systems (in the critical area, subcritical area, intermediate cooling circuit, conversion system, storage area, reprocessing units, etc.). Some of these may at some stage cause toxic materials (radioactive or chemical) to be released into the environment. A few examples of accidents in the critical and subcritical areas are described below. Calculations performed using simple tools are also described (reference 81) with the aim of presenting the physical phenomena involved and providing orders of magnitude for the changes in some system parameters.

6.4.2.1 Reactivity anomalies

Reactivity anomalies include:

- rapid reactivity insertions due to an increase in the volume of fuel salt in the central cavity, as may occur if heat shields become detached, in case of structures thermal expansion in the central cavity or in the event of a break in the fertile wall;
- slow reactivity insertions, for example following uncontrolled addition of fuel salt or a change in local fissile material concentrations, a variation in salt temperature or density, a higher proportion of “useful” delayed neutrons due to a fuel salt circulation outage, etc.;
- reactivity decreases resulting, for example, from the presence of a foreign fluid in the fuel salt, a change in salt temperature or density, excessive bubble injection, etc.

A time-dependent linear reactivity insertion of 500 pcm for insertion time from 1s to 500 s was studied. This approach appears to adequately cover the various reactivity insertion conditions described previously.
Analysing the transient using a zone-based point kinetics model in the fuel system revealed that no power surges liable to damage reactor internals occur when the insertion time is longer than approximately 1 second.
Furthermore, the designers indicate that despite the low proportion of delayed neutrons, the reactor would be able to tolerate reactivity insertions of up to 1000 pcm in 1 second, at which level the core would reach prompt criticality (taking feedback into consideration).
In the event of significant reactivity insertion, irrespective of the insertion time, the salt temperature increases considerably (ΔT=100K for reactivity of 500 pcm) and stabilises, if the core reactivity does not subsequently decrease (for example, following an increase in the volume of salt in the central cavity or the addition of fissile materials in the central cavity). The reactor is stable, but design measures must be adopted to ensure that structural materials in contact with fuel salt are not damaged over time.

Anti-reactivity insertions may cause a decrease in power and hence temperature, which may in turn lead to a risk of solidification of the fuel salt. Inserting anti-reactivity of around 1000 pcm may cause the temperature to fall below the solidification point.

6.4.2.2 Loss of fuel system cooling

Malfunctions affecting the intermediate cooling circuit (pumps shutdowns, jamming, inadequate system cooling or breaks) may result in partial loss of fuel salt cooling. Based on a qualitative analysis of the various transients, it would appear that the occurrence of a break in the intermediate cooling circuit results in the fastest loss-of-cooling sequence. A break in the inlet part of the heat exchanger can lead to a loss of fuel system cooling within a few seconds. Pipe break transients calculated using the heat removal decrease law $P_{rem} = P_0 exp(-\frac{t}{\tau})$ were analysed for various characteristic times (τ).

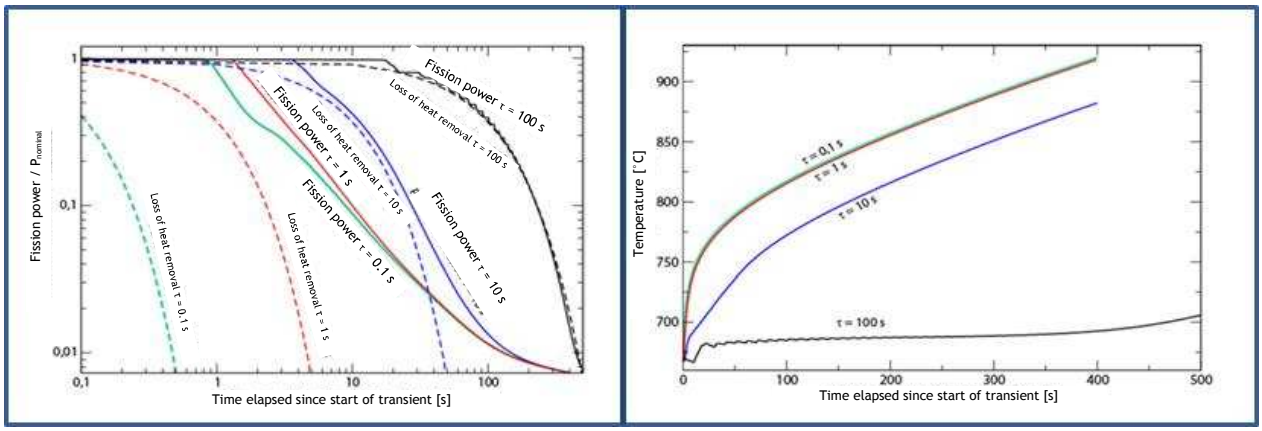

Figure 40: Heat removal (dotted line), corresponding fission power (left) and average fuel salt temperature (right) for a loss-of-cooling transient (τ = 0.1 s in green, τ = 1 s in red, τ = 10 s in blue and τ = 100 s in black)

When loss of cooling occurs rapidly (τ = 0.1 s and τ = 1 s), core power significantly exceeds the heat removed at the start of the transient, causing the fuel salt temperature to rapidly increase (see Figure 40). A few tens of seconds after the start of the transient, the temperature continues to rise but less rapidly as the chain reaction stops. Note that the two transients with characteristic times (τ) of 0.1 s and 1 s are very similar. The rate at which fissile power decreases is limited by the decrease of delayed neutron precursors and by the time taken by the fuel salt to travel through the section of the system between the intermediate heat exchanger and the central cavity. It should also be noted that, for the transients with characteristic times (τ) of 0.1 s, 1 s and 10 s, fission power tends towards the same asymptote after 100 s. The remaining fission power is attributable to neutrons emitted by the long-lived delayed neutron precursors circulating with the fuel salt. The value of this parameter is therefore independent of the characteristic time of the transient.

At τ= 100 s, the loss of cooling occurs in a manner similar to a load transient. The change in heat removal occurs slowly enough to allow the fission power to adjust appropriately.

The study provides an order of magnitude for the grace period in which to drain the fuel salt. As already stated, this period is approximately 10 minutes. Thereafter, the average fuel salt temperature approaches 1200°C in the event of a rapid loss of cooling. Although the structural materials will not have reached the same temperature (as the salt is a poor thermal conductor), it must be drained very quickly to protect them.

6.4.2.3 Loss of flow in the fuel system

The initiating events for a loss of flow in the fuel system are a pump shutdown and a jamming of the system (although total jamming appears improbable, a severe reduction in flow rate cannot be ruled out). The intermediate coolant is assumed to continue circulating, and natural convection is simulated in the core.

The transients associated with a loss of flow are shown in Figure 41.

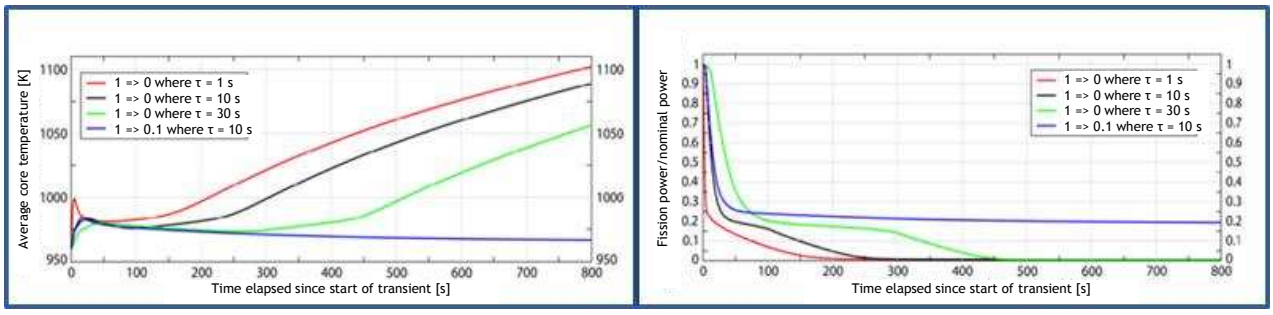

Figure 41: Transients involving flow rate reductions from 100% to 0% (exp(-t/τ)) with τ = 1 s, 10 s and 30 s, and from 100% to 10% with τ = 10 s: mean core temperatures and standard fission power as a function of the time since the start of the transient

The average core temperature rises rapidly at the start of the transient: the temperature increase for a pump shutdown with a characteristic time (τ) of 1s is greater (ΔT ~ 40K in 5 seconds) than for slower transients (ΔT ~ 10 to 20K). The fission power decreases to zero within a few hundred seconds of the start of the transient and the temperature increase is due to decay heat only. The temperature profiles shown in Figure 41 should be considered with caution, however, as natural convection is not accurately modelled in these transients.

The consequences of the temperature surge generated by this type of accident are negligible, compared to a loss-of-cooling accident. However, continuous salt heating or rapid expansion may damage structures (when adiabatic expansion is assumed, the salt must be drained within 10 minutes, as for the loss-of-cooling transient).
The risk of salt solidifying in heat exchangers must also be taken into consideration, as the fuel salt temperature may in theory fall to match that of the intermediate coolant.

6.4.2.4 Inability to drain the fuel salt with the intermediate cooling circuit inoperative

If the fuel salt cannot be drained within around 10 minutes and the intermediate cooling circuit is inoperative, the resulting temperature increase rapidly leads to structural collapse. The designers intend to create a weak spot that will give way first. This weak spot is located at the base of the central cavity in the critical area. The feasibility of such an option is subject to confirmation, as the hottest areas are not located near the bottom of the reactor vessel. The temperature distribution will need to be assessed and taken into consideration. As the weak spot is located near one or more draining valves, the fuel salt would flow to the tanks in the subcritical area, which must be designed to withstand the presence of salt at very high temperatures, potentially exceeding 1200°C (thermal inertia, refractory materials, etc.).

This malfunction may lead to partial collapse of the base of the central cavity. However, if the fuel salt can ultimately be drained to the subcritical area, this accident should not result in environmental releases.

6.4.2.5 Loss of fuel salt

Consideration must be given to the possibility of fuel salt leaks from the various fuel system components (heat exchanger, pump, air-salt separator, fertile blanket, wall etc.) in both the hot and cold legs. In the event of a leak from pipes or pumps, fuel salt will leak through the break into the fuel envelope in the critical area (see Figure 35). The manifolds at the bottom of this envelope will drain the salt to into the dump tank (in the subcritical area).
Nevertheless, salt may also flow into the intermediate cooling circuit and travel towards the gas processing unit or enter the fertile blanket cooling system.
In the event of a loss of leaktightness on an intermediate heat exchanger, the fuel salt and intermediate salt will mix. If radioelements are detected in the intermediate cooling circuit, the shutdown and draining procedures will be triggered. The volume of the subcritical area is designed to accommodate all of the fuel salt and part of the intermediate salt. If too much intermediate salt were to pour into the subcritical area, it might overflow, potentially compromising the first barrier. The MSFR includes design provisions to prevent additional intermediate salt from being supplied to the affected loop.

6.4.2.6 Severe accident

The definition of a severe accident for the MSR is by necessity very different to that adopted for other reactors, as the fuel is in liquid form during normal operation. Ultimately, the loss of integrity of the fuel envelope might be adopted as the defining feature of a severe accident. In such an event, there is a risk of high-energy phenomena resulting from thermodynamic interaction between salt and water (see Section 6.4.1.2), potentially jeopardising the integrity of the second or even the third barrier. The confinement function would be “significantly degraded”.

Based on current knowledge, the MSFR designers do not appear to have adopted this assumption in the design basis for the second and third barriers.

There will also be stockpiles of poisoned inert salt to prevent any return to criticality by fuel salt that has flowed into the cooling pool in the subcritical area. The available research shows that it may be possible to discount the risk of a return to criticality, even without neutron poisons.

6.5 ENVIRONMENTAL IMPACT AND RADIATION PROTECTION

Primary coolant composition

The source term consists of all the radio and chemical elements contained inside the fuel envelope, including fuel salt circulating in the reprocessing unit as well as the reactor, if the two installations are linked.

Metal oxides formed during corrosion processes are not stable in fluoride salts and are likely to be dissolved. Fluorides have the peculiarity of bonding with a very wide variety of metallic elements (M^{n+}). The molecules thus formed with fission, activation and corrosion products, with the formula $MF_x^{(n-x)+}$, may be soluble in salt, present in gaseous form or insoluble.

The extraction and separation processes used in the processing unit are based on acid-base, redox, complexing and decomplexing reactions. This is achieved by introducing chemicals into the processes. Examples include fluorine (F_2) and hydrogen (H_2), which are used to extract and recover 99% of the uranium (reference 82). These reactions produce a number of substances, including hydrofluoric acid (HF), which is produced when lanthanides are oxidised.

Radiation protection

The fact that the coolant and fuel are mixed together has a negative impact on radiation protection, as there is no separate coolant to provide a screening effect. Worker exposure will depend on the nature and composition of the fuel, which have yet to be determined for this reactor (choice between thorium, plutonium and uranium).

Exposure during maintenance activities is likely to be quite significant, as the primary circuit will still contain traces of coolant after being drained. Due to the highly corrosive nature of molten salts, maintenance operations will probably be required frequently.

Although having an integrated reprocessing unit limits the need to handle irradiated fuel, the complexity of the processes implies the need for multiple systems, which in turn represent additional exposure sources. Ultimately, the order of magnitude of total worker exposure is difficult to predict, as it will depend to a great extent on the degree to which maintenance operations are automated.

Liquid and gaseous waste management

As for the SFR and LFR processes, this reactor operates in near-atmospheric pressure conditions, limiting the risk of dispersing hazardous products in the event of an accident.

The solubility of actinides and fission products in the event of an accident depends heavily on the type of salt used and its temperature; solubility properties are an important criterion when selecting salts. Actinides and soluble fission products will remain trapped in the salts in severe accident conditions (resulting in the formation of fluorides). Insoluble fission products include gases, aerosols and noble metals, the in-line reprocessing system will limit the inventory, although it is not currently possible to accurately state to what extent. In the event of a leak, the surface of the salts should freeze (as they have a solidification point in the region of 550°C), which should limit any dispersion of radioactive materials.

As stated previously, contact between water and fuel salt is theoretically possible, for example in the pool in the subcritical area. Studies should be conducted to determine the solubility of the various salts in water and the vapour pressures at different water temperatures, in order to estimate the types of product and the quantities of radioactivity liable to be released into the environment in the event of a severe accident.

Lastly, the radioactive materials removed from the primary circuit during in-line reprocessing must be stored; they therefore represent a significant source of contamination. The safety studies should be coupled, to take into consideration both the reactor and the storage facilities, which are potentially a major factor in the radiological risk.

When assessing releases, due consideration should be given to the fact that the reprocessing unit situated near the reactor contains highly reactive substances such as hydrogen as well as highly toxic and corrosive substances, including hydrofluoric acid and fluorine. Until the composition of the reactor coolant in terms of fluorinated molecules is known, it is hard to determine coolant chemical toxicity. Furthermore, it should be noted that molten salts create unusual physicochemical conditions, in particular in terms of redox potential. Thus, reactor coolant toxicity is due not only to its composition but also to its reactivity, which is liable to form other highly toxic substances in the event of an environmental release.

Case of tritium in nominal conditions

In normal operation, tritium would be produced by ternary fission but also by lithium and boron, if present in the neutron protection. Tritium is removed by the bubbling system, with most of it being reinjected into the fuel salt. A fraction of the tritium may subsequently enter the intermediate salt and water as a result of diffusion through the heat exchanger walls.

6.6 MATURITY OF CONCEPT AND R&D REQUIREMENTS

6.6.1 *OPERATING EXPERIENCE FEEDBACK*

The information in this section is taken from the MSRE experience feedback. Although the MSRE reactor is not representative of MSFR specificities in terms of neutron spectrum, salt composition, reactivity control mechanisms, operating temperatures, etc., a number of lessons may be learned from its operation.

For structures in contact with fuel salt, the MSRE's designers chose Hastelloy N, a high-nickel alloy that also contains molybdenum, chromium and iron; Hastelloy provides effective corrosion resistance (due to the nickel), allows a passivating layer to form in contact with air (due to the chromium) and enhances the mechanical properties of such systems. The MSRE design provided for operation with Hastelloy N surfaces at a temperature of 700°C and a fuel salt pressure of 0.35 MPa.

MSRE operation yields useful experience feedback regarding structural corrosion, thanks to a series of test specimens that were submerged and subsequently removed at regular intervals during the operational life of the reactor. The solution for controlling the redox potential (see Section 6.4.1.1) was based on the concentration ratio between beryllium (Be) and the Be^{2+} ions in the fuel salt. It was possible to limit corrosion at fuel salt temperatures up to 650°C by adjusting this ratio over the course of the MSRE's operational life.

Experience acquired with MSRE also revealed problems relating to the management of fission products that are insoluble in salt, such as fission gases and noble metals. The MSRE design featured a degassing process (via the pump chamber) to remove fission gases, but this solution did not effectively remove metals suspended in the molten salt. Subsequent studies were commissioned, to develop an in-line bubbling system, as proposed for the MSFR.

The MSRE also trialled the operation of draining devices that open when a plug of solid salt melts, as well as the use of flanges with seals protected by an area of solid salt.

ORNL published numerous safety study-related reports between 1964 and 1968, none of which mention significant events concerning the safety of the MSRE reactor. It should be noted, however, that the reactor operated for less than two years.

6.6.2 *R&D REQUIREMENTS*

Numerous technological challenges remain to be overcome before the construction of an MSR can be considered. The safety demonstration for such a reactor is also likely to be difficult. The resulting R&D requirements are described below.

Nuclear/thermal-hydraulic/physicochemical coupling

The nuclear and thermal-hydraulic aspects are strongly coupled. Certain effects relating to the physicochemical behaviour of fissile materials dissolved in LiF must also be taken into consideration; for example, the solubility of fissile materials may decrease in the cooler parts of the fuel system.

It will be necessary to accurately characterise the various phenomena and develop suitable tools to reflect the strong coupling between these aspects. Given our current level of understanding, considerable R&D efforts will be necessary.

Materials

The purpose of the planned materials-related R&D initiatives is to develop materials able to withstand temperatures above 700°C in contact with fluorinated salt and fission products contained in it. Such is the maximum temperature limit adopted for the structures. Concerning the central cavity, irradiation effects must also be taken into account, as well as helium production in materials. However, it should be noted that the use of a fluorinated salt limits damage to structures in contact with fuel salt to 7 dpa/year; changing them over the course of the reactor life appears unavoidable.

For the MSFR project, CNRS is considering the Hastelloy N alloy already used for the MSRE, although it will be necessary to characterise its high-temperature behaviour when subjected to a fast-neutron flux, to assess the acceptable fluences. For temperatures higher than those encountered in the MSRE, the use of nickel alloys such as EM721 (70% nickel, 20% tungsten) or EM 722 (60% nickel, 25% tungsten) are being considered.

The designers intend to install heat shielding, to protect the fuel system walls. The materials for such shielding would also have to be qualified for reactor operating conditions. The risk of salt infiltration between the shielding and structural materials should also be taken into consideration. In any case, the feasibility of such a solution remains to be demonstrated.

Heat exchanger design

Many R&D studies are required in order to design the heat exchangers, which are a sensitive point of the facility. The heat exchangers must deliver a high heat-exchanging capacity using a limited volume of salt, and be able to withstand high operating temperatures. The possibility of deposits being present in heat exchangers must also be taken into consideration.

Solubility of salt components

The conditions in which clumps of solid matter form on system walls upon approaching the solubility limits of certain fuel salt components – and the means of limiting this phenomenon – must be studied.

Salt processing

Fluorination of uranium and transuranium elements does not raise any theoretical difficulty, but the ability to implement such a practice industrially remains to be demonstrated. Furthermore, the mechanism for extracting fission products that are not soluble in the fuel salt (other than noble gases) is yet to be researched.

Salt-water interactions

The solubility of fluorides in water is poorly understood, but is currently considered to be limited. As already stated, this is an important issue with regard to the studies of the radiological consequences of the environmental release of water that has been in contact with fuel salt. Hydrofluoric acid production by fluorine hydrolysis is also a significant aspect that must be quantified.

Instrumentation

The instrumentation (thermocouples, flow meters, leak detectors, etc.) used to monitor the barriers remains to be defined and tested.

Startup procedure

As stated in Section 6.3.1, considerable research will be necessary before the MSFR startup procedure can be defined, as the reactor concept does not include control rods.

It should also be noted that a Forced Fluoride Flow Experimental Research (FFFER) loop is currently under construction at the CNRS facility in Grenoble. This installation will be used to test the bubbling technique, instrumentation and draining systems.

6.7 CONCLUSION REGARDING REACTORS OPERATING WITH FUEL SALT

The fuel salt-based MSR concept is very different to the other concepts selected by GIF, owing in particular to the fact that the fuel is in liquid form and combined with the coolant. These characteristics give it interesting intrinsic nuclear properties, in theory enabling very stable reactor operation: the neutron feedback coefficients are strongly negative, even for a large power fast-spectrum reactor. This behaviour poses a problem, however, in terms of the approach to criticality during reactor startup, particularly with the MSFR concept, which does not feature control rods.

Conversely, the MSR is particularly sensitive to loss-of-cooling transients, as the thermal inertia of the fuel salt is very low, requiring it to be drained without delay into purpose-designed tanks. If the fuel salt is not drained rapidly, the high temperature of the salt would rapidly cause the structures in contact with it to collapse. Reactor safety is therefore dependent to a large extent on the reliability and performance of the draining system.

The risk of salt crystallising (around 550°C) requires the process to operate at high temperature. The highly corrosive nature of the salts must also be taken into consideration. As a result, the choice of materials is a major safety issue for this reactor. Significant loads will be placed on the various structures, and recent studies have revealed the existence of hot spots in reactor systems.

It should also be noted that the strong coupling between nuclear, thermal-hydraulic and thermochemical phenomena makes the safety demonstration particularly challenging. In this respect, determining the radioactive and toxic inventories (identifying locations, quantities, radionuclide spectra, etc.) throughout the system is an essential but highly complex step, given that these inventories change constantly. This evaluation is also essential for the purpose of assessing the consequences of accidents. Note that some salts and substances used or produced during pyrochemical processing are particularly toxic. The toxic effects of products formed by reactions between fluorinated salts and water will also need to be studied.

Worker exposure will depend on the nature and composition of the fuel salt and the scope for automating maintenance operations. The large surface area of the radioactive material-carrying systems is a disadvantage from a worker exposure perspective, however.

Ultimately, IRSN considers that the MSR development roadmap should include a detailed research and development plan featuring studies and tests to validate various technological options and assess their feasibility. In this context, selecting and validating a material suitable for high-temperature operation in a saline environment containing fission products, and designing appropriate intermediate heat exchangers will be key factors in demonstrating the viability of the concept. IRSN considers that the development of this process will require the construction of a low-power demonstrator, which is not a realistic short-term prospect.

Moreover, it is necessary to define a safety approach tailored to the peculiarities of MSRs operating with fuel salt (liquid fuel, coupling between the reactor and the reprocessing unit, severe accident definition, etc.).

6.8 SOLID-FUEL MSRs

This type of reactor is not expressly mentioned in the original edition of the GIF document in reference 1, but is included in the update to it (reference 2). Most related research is being conducted by American laboratories, and in particular by ORNL (since 2004) via the Advanced High Temperature Reactor (AHTR) project.

This section provides an overview of the information available in the ORNL reports references 83 and 84. The main aim of these reports is to demonstrate the feasibility and estimate the cost of such a reactor; they do not include any safety studies.

Other reactors of this type have been developed, including the Pebble Bed-Advanced High Temperature Reactor (PB-AHTR) devised by the University of California, Berkeley and described in 2008 in the paper in reference 85. The Chinese Academy of Science is also interested in this technology, and hopes to commission a 2 MWth MSR in 2016 (reference 2).

6.8.1 *GENERAL DESIGN*

The AHTR is being developed with a thermal power of 3400 MW and an electricity generating power of 1530 MW.

The main features of the reactor are as follows (see Figure 42):

- main vessel essentially containing the core and decay heat removal system, operating with a molten salt coolant at a temperature that varies between 650°C and 700°C;
- primary loops (number to be defined) through which the molten salt circulates after passing through the core;
- intermediate loops (number to be defined), through which another molten salt circulates, each connected with a primary loop by an intermediate heat exchanger;
- other loops (number to be defined) carrying supercritical water, connected to the intermediate loops.

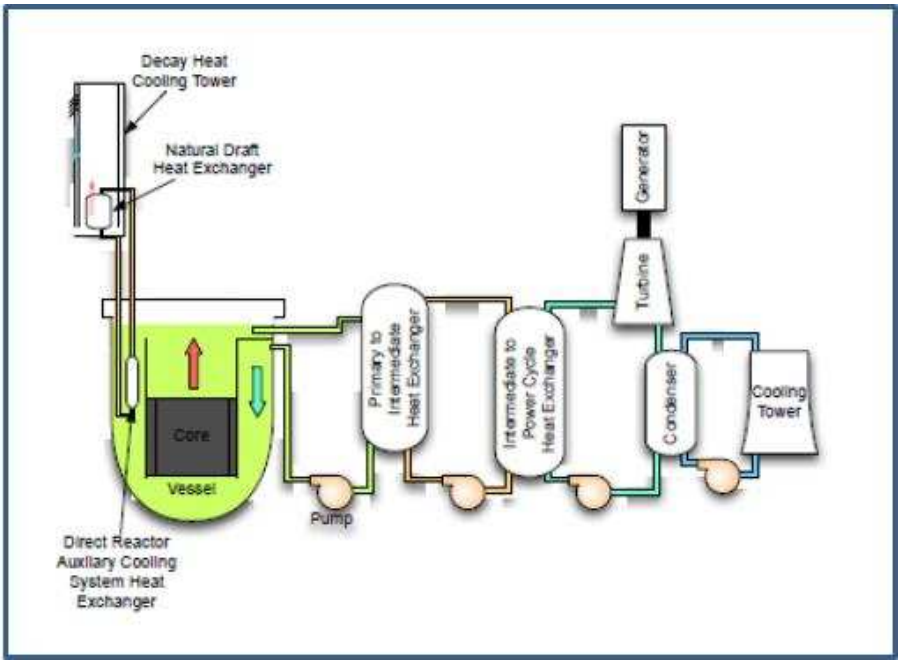

Figure 42: Schematic Diagram of the AHTR

6.8.2 *FUEL AND COOLANT*

The AHTR is a thermal-spectrum reactor. It uses TRISO particles containing uranium oxycarbide enriched to 9% ^{235}U. The fuel particles are grouped in plates, which are in turn combined in a hexagonal fuel assembly (see Figure 43) the structural components of which are fabricated using a carbon-carbon (C-C) composite.

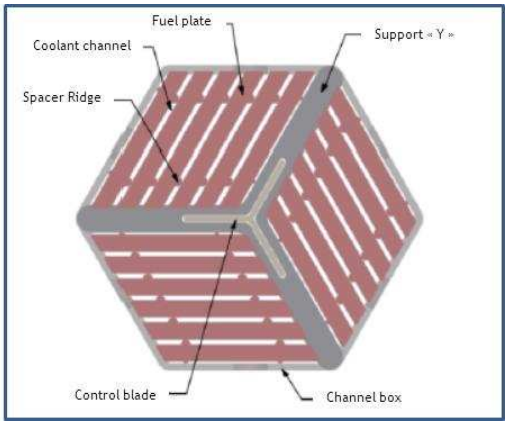

Figure 43: Cross-section of a fuel assembly for the AHTR reactor

The reactor coolant consists of a FLiBe salt (see Section 6.2.3), which has good nuclear characteristics (low neutron capture and high moderating capacity), and the coolant in the intermediate system is another salt, consisting of potassium and zirconium fluorides ($KF\text{-}ZrF_4$). Both of these salts have boiling points in the region of 1500°C, enabling them to be used at low pressure. These two coolants also have a high solidification temperature (460°C), which will influence the design.

6.8.3 *BARRIERS*

The concept features three barriers.

The first barrier consists of the fuel particle cladding. The robustness of these particles has been highlighted as justification for the strength of this barrier, even in accident conditions.

The second barrier consists of the reactor vessel and top head, the decay heat removal system heat exchanger tubes (which are immersed in the reactor vessel), the primary vessel and the intermediate heat exchanger walls. The material selected for the reactor vessel and connected systems is Hastelloy N, which has already been used for the MSRE.

The reactor building - of a similar design to those in the SFR and LFR concepts – forms the third barrier.

6.8.4 *CONTROL OF SAFETY FUNCTIONS*

6.8.4.1 Reactivity control

Using FliBe as the coolant makes for negative neutron feedback coefficients. Excessive heating of the reactor coolant salt would cause the nuclear reaction to stop.

Reactivity is controlled by means of cruciform rod clusters that use hafnium and molybdenum carbide as absorber materials. Each fuel assembly is equipped with such a rod cluster. The fuel assembly is handled with its cluster, preventing criticality accidents during core loading and unloading operations.

To offset the reactivity excess at the start of the fuel cycle, europium particles are included in the plates.

6.8.4.2 Decay heat removal

The decay heat removal system resembles the system defined for the SFR, comprising a salt-salt heat exchanger submerged in the reactor vessel and a salt-air heat exchanger located outside the vessel.

The proposed system is totally passive; no information relating to redundancy or diversification of the system has been provided.

6.8.4.3 Confinement of radioactive materials

IRSN does not have any information on this subject.

6.8.5 *CONCLUSION REGARDING SOLID-FUEL MSRs*

This concept benefits from design and operating feedback from other concepts. For example, it features:

- a thermal spectrum core with negative neutron feedback coefficients;
- a very strong first barrier, due to the use of TRISO particles;
- suitability for high-temperature (600°C-660°C), low-pressure operation;
- use of materials already tested in the MSRE reactor;
- high performance achieved by using supercritical water;
- totally passive decay heat removal system.

However, this concept is still subject to corrosion problems and the other disadvantages inherent with the use of molten salts and supercritical water.

Furthermore, the fuel assembly design is highly innovative, due to the use of a carbon-carbon composite material.

7. SUPERCRITICAL WATER-COOLED REACTORS (SCWR)

The supercritical water-cooled reactor (SCWR) is the only reactor concept selected by GIF that uses water as coolant. It is described as an advanced version of existing BWRs and PWRs, delivering a thermodynamic efficiency of around 45%, compared with values in the region of 35% for PWRs and BWRs.

7.1 HISTORY AND PROSPECTS FOR DEVELOPMENT

7.1.1 *BACKGROUND*

Various SCWR concepts were researched between 1950 and 1960, particularly by General Electric and Westinghouse in the United States, but development activities for this reactor type were subsequently abandoned. No reactors based on this concept have ever been built.

However, "supercritical technology" has been developed for coal-fired power stations, and there are now some 400 plants of this type around the world. This technology achieves superior efficiency through higher coolant temperatures at the turbine inlet, cutting the cost per megawatt and decreasing pollutant releases. Coal-fired power plants operating with supercritical water exist in the United States, Japan and Germany, in particular, some of which date back to the early 1950s. This is the case, for example, of the 1000 MWe Niederaussem plant in Germany, which burns powdered lignite and heats water to 600°C at a pressure of 26.5 MPa (delivering a thermodynamic efficiency of 45 %).

7.1.2 *ONGOING PROJECTS AND PROSPECTS FOR DEVELOPMENT*

To date, interest in SCWRs has been primarily research-oriented, and no industrial projects are planned for the short or medium term.

Canada, Japan, Russia and Europe (via Euratom) have entered into a joint R&D agreement within the GIF framework. China is also interested in this concept, and is working with Canada and Euratom.

The most active teams in this field are based in Canada and Europe:

- Canada is developing a concept that is an upgraded version of its CANDU reactors, in which the water used as a coolant is in a supercritical state and the moderating function is performed by heavy water. The goal is to operate at a pressure of 25 MPa and a coolant temperature of 625°C in the turbine;
- in Europe, the High Performance Light Water Reactor (HPLWR) – a thermal-spectrum concept with a rated power of 1000 MWe (2300 MWth) – has been developed under the EC's FP5; the aim is to achieve a core outlet water temperature of 500°C. Like the BWR, this is a direct-cycle concept. It is described in the report in reference document 86.

Other concepts have been studied, in Japan, for example, where Tokyo University researched a "Super Light Water Reactor" thermal spectrum concept with a power of 700 MWe (and a target water temperature of 500°C) as well as a fast-spectrum concept ("Super Fast Reactor"). Japanese R&D relating to SCWR concepts has been on standby since the Fukushima accident.

Russia is also researching a thermal spectrum SCWR concept.

7.2 MAIN CHARACTERISTICS

Unless otherwise stated, the information in this section relates to the aforementioned HPLWR project.

7.2.1 *GENERAL DESIGN*

The SCWR has a similar operating principle to boiling water reactors (see Figure 44): the water that cools the core is sent directly to the turbine via four steam lines. The condensed steam is then pressurized and reinjected into the vessel. Given the characteristics of supercritical water, parts of the SCWR operational design are simpler than the BWR, as the SCWR does not require water-steam separators, steam dryers or recirculating pumps. Consequently, the control rods can be inserted from above the core, as in the PWR. The pressure and temperature differentials are particularly high, rising from (5 bars, 33°C) at the low-pressure turbine to (250 bars, 500°C) at the reactor vessel outlet, which leads to significant structural loads.

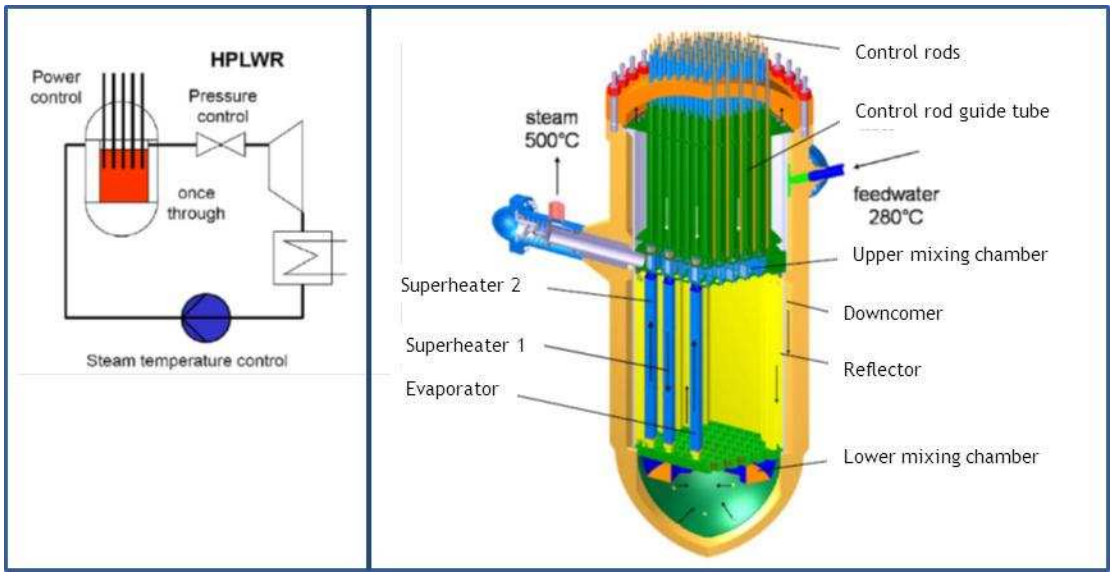

Figure 44: Schematic diagram of the HPLWR

The SCWR safety systems are fairly similar to those in the BWR. This applies to the HPLWR:

- the reactor can be shut down using control rods and a borated water injection system that maintains the reactor in a subcritical state during the post-accident phase;
- the containment can be isolated by closing various valves;
- pressure relief valves limit the water pressure in the reactor vessel;
- the supercritical water is depressurised in a pool located inside the containment;
- an emergency injection system is able to inject water into the reactor vessel;
- a decay heat removal system (suppression pool) limits pressure in the containment;

Figure 45 provides a diagrammatic representation of these systems.

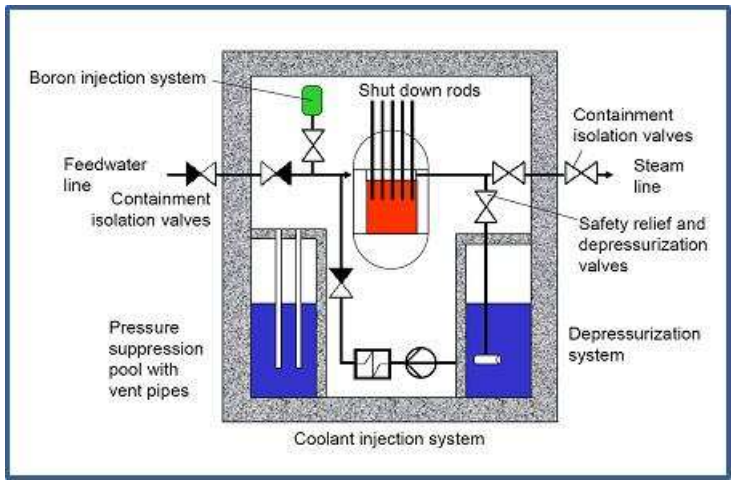

Figure 45: Schematic diagram of HPLWR safety systems

7.2.2 *COOLANT AND MODERATOR*

The SCWR uses water as both coolant and moderator; the neutron spectrum may be either thermal or fast, depending on the quantity of water in the core. Additional moderators may also be used.

The water is maintained in supercritical thermodynamic conditions: the water saturation curve features a point known as the critical point (at a pressure of 22.1 MPa and a temperature of 374°C), above which water no longer changes phase as its temperature increases. In Figure 46, the representative points characterising the operation of PWR, BWR and SWCR reactors are shown in a temperature-pressure chart (left) and a temperature-entropy chart (right).

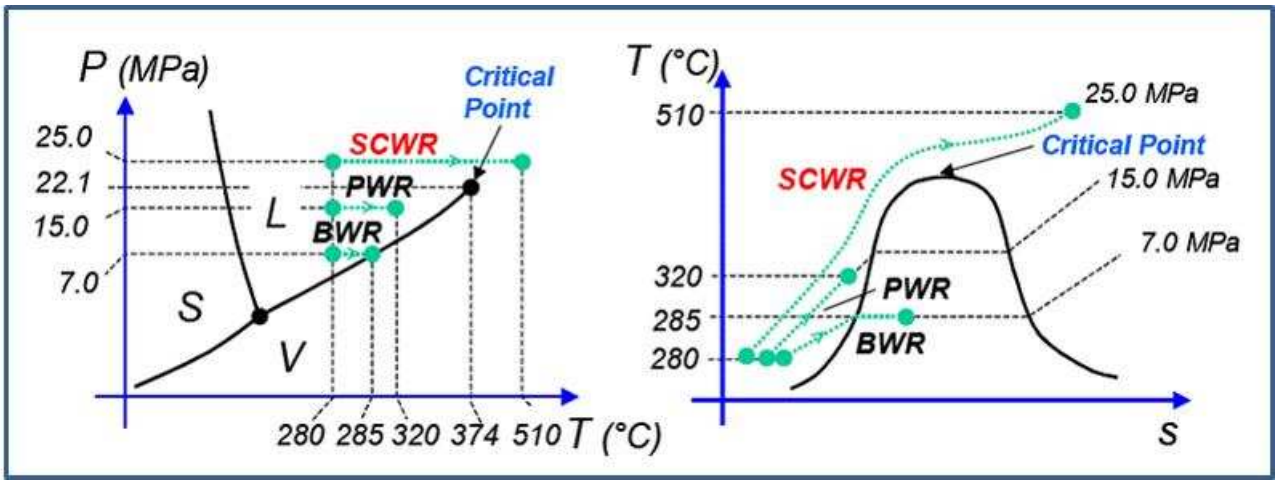

Figure 46: Left: Water pressure-temperature (P,T) chart (L for liquid, S for solid and V for vapour) Right: entropy-temperature chart showing the operating points of the BWR, PWR and SCWR

The phenomena associated with phase changes, such as boiling crisis (dry-out in the BWR and departure from nucleate boiling in the PWR), which have a restrictive impact on current light water reactors, do not occur in the SCWR when the primary circuit is operating at its nominal pressure. To illustrate the operational specificity of the SCWR, a brief overview of the main features of the HPLWR project (reference 86) are given below.

The HPLWR operates at a pressure of 25 MPa. Water enters the core at a temperature of 280°C and the variation in specific enthalpy of the coolant as it passes through the core is approximately 2,000 kJ/kg at a water flow rate of 1180 kg/s; the core outlet water temperature reaches 500°C. Figure 47 shows the changes in water temperature and specific heat capacity (Cp) as a function of the specific enthalpy of the coolant, considering an operating pressure of 25 MPa. Note the presence of a zone in which temperature variations are small due to the very large changes in specific heat capacity as a function of coolant specific enthalpy.

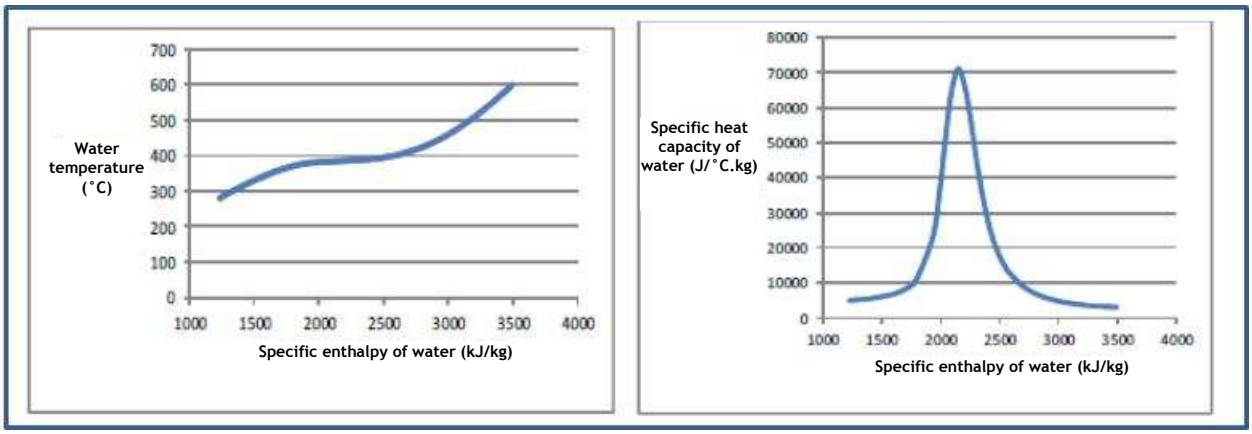

Figure 47: Variation in the temperature and specific heat capacity of water as a function of specific enthalpy

Figure 48 also shows the variation in water density as a function of coolant specific enthalpy between the core inlet and core outlet. Due to the low density of the water as it exits the core, the moderating and cooling functions will have to be uncoupled, as water that has flowed through the core will not produce an adequate moderating effect in the upper part of the core. A quantity of less-heated water is reserved in order to provide this moderating effect.

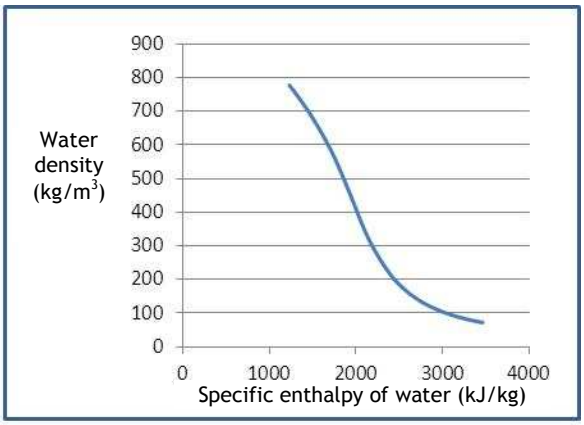

Figure 48: Variation in water density as a function of specific enthalpy

For any given pressure above the critical pressure, there exists a maximum specific heat capacity at a particular temperature known as the pseudo-critical temperature. At 25 MPa, the pseudo-critical temperature is 384°C. Around pseudo-critical conditions, significant variations in heat transfer between the fuel and water may occur under certain specific thermal-flux and mass-flow rate conditions.

Figure 49 shows a comparison of the temperature variations in an average channel (shown in green for the PWR and blue for the SCWR) and a hot channel with an enthalpy hot spot factor of 2 (shown in red for both reactor types) for an PWR and an SCWR. In the PWR, the hot channel temperature is limited by the saturation curve and the risk associated with the hot channel essentially relates to a departure from nucleate boiling. This limitation does not exist for the SCWR. For example, in the case of the HPLWR, for which the average water heating effect is 220°C and the inlet temperature is 280°C, a hot spot factor of 2 would result in a coolant temperature of 1200°C in the hot spot, which would be unacceptable for the internals and clads.

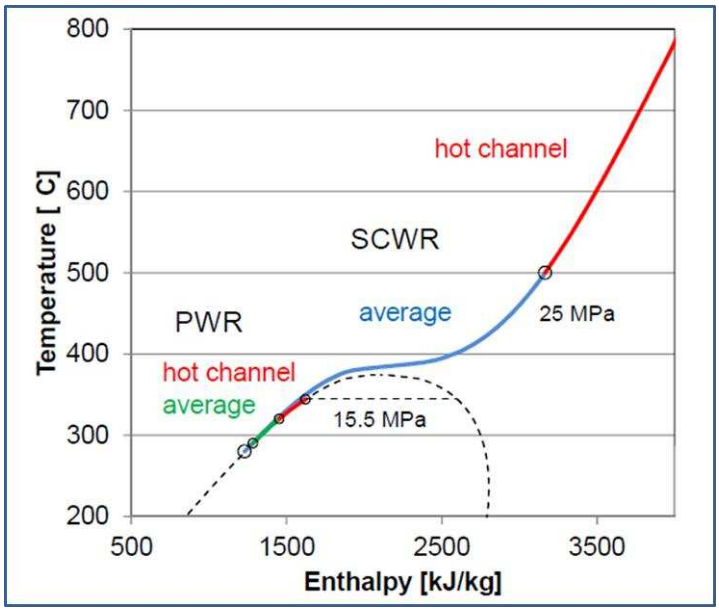

Figure 49: Comparison of temperatures reached in a PWR and an SCWR

The large variation in enthalpy as water passes through the core also results in a much lower core water flow rate than with a PWR, for a given electric power output.

Table 2 shows a few characteristics of the three plant series in the French PWR fleet in operation and the corresponding characteristics for the HPLWR. Note that the SCWR has a smaller water inventory than the PWR.

Table 2: Comparison of certain characteristics of the PWRs in the EDF fleet and the HPLWR

Reactor Type	Thermal power (MW)	Gross electrical power (MW)	Therm efficiency (%)	Water flow rate in the core (kg/s)
CP1/CP2	2785	960	34.5	12,670
P4/P'4	3817	1370	35.9	17,880
N4	4270	1520	35.6	19,137
HPLWR	**2300**	**1046**	**45.4**	**1179**

7.2.3 *FUEL*

The intended fuel is UO_2, in the form of pellets stacked inside fuel rod cladding. The level of ^{235}U-enrichment (approx. 7%) is higher than for PWR fuel assemblies, due to the intended cladding materials.

Use of MOX fuel is also possible, particularly for fast-spectrum concepts.

Lastly, a few studies have been conducted in Canada with a view to using uranium- and thorium-based fuel.

7.2.4 *REACTOR CORE DESIGN*

Cooling the core is among the key challenges for SCWR design. The water inlet and outlet temperatures are respectively 280°C and 500°C, and the cladding temperature must be limited to 630°C, to preserve its integrity despite creep and corrosion effects.

As stated in Section 7.2.2, the outlet temperature for the hottest channel would reach 1200°C if the water flowed through the core in a single passage. To limit the variation in enthalpy in the hot channel and limit the cladding temperature to 630°C, a core concept was developed to mix the coolant at various levels, thereby resetting the water heating radial form factor and limiting the corresponding core inlet and outlet values.

This core concept is illustrated in Figure 50 and Figure 51.

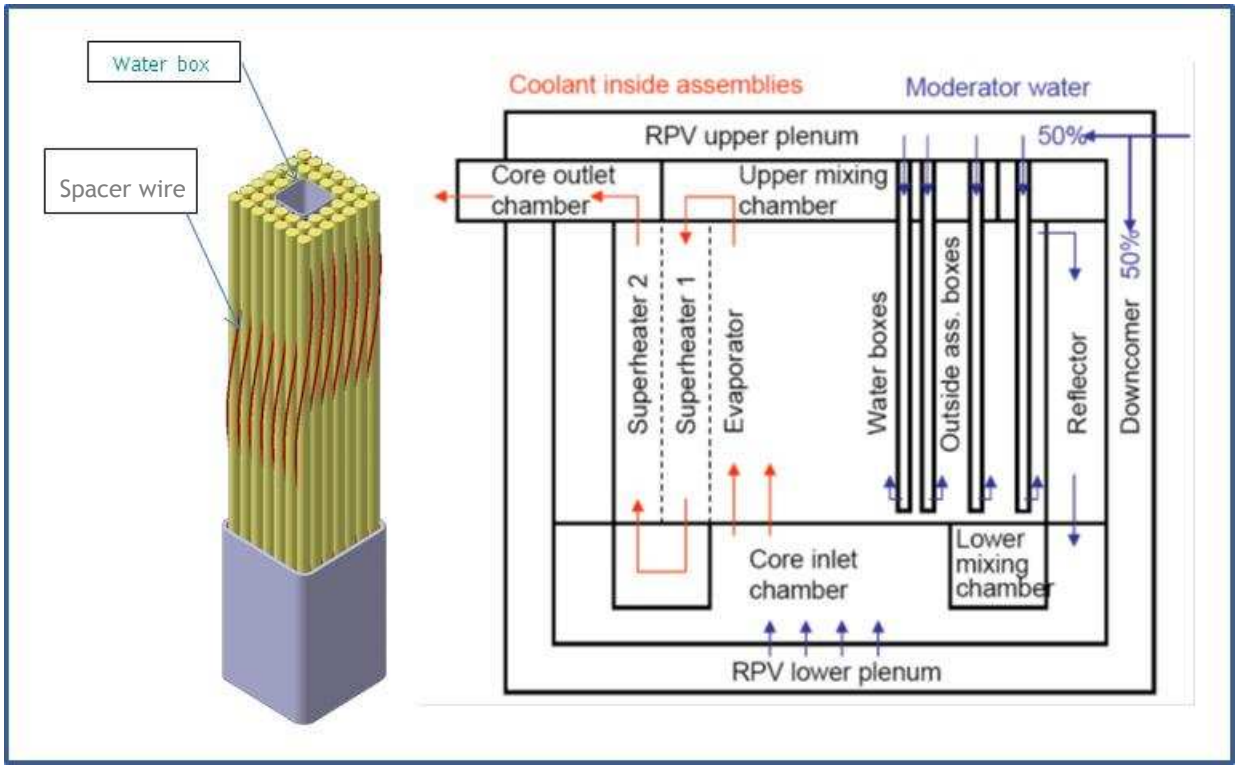

Figure 50: Diagram of a fuel assembly and the route followed by water flowing through the HPLWR core

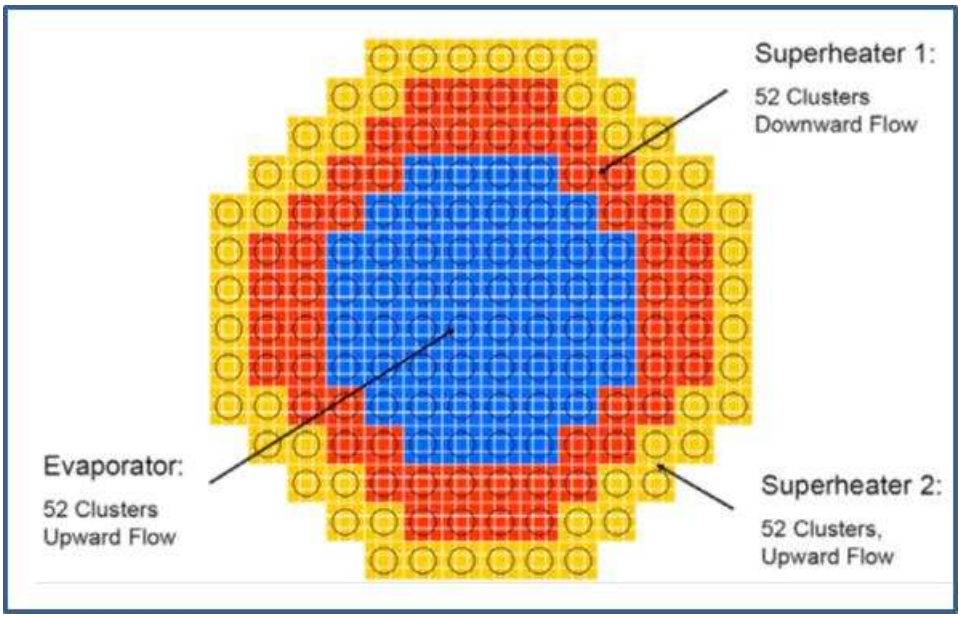

Figure 51: Diagram of the HPLWR core

The fuel assemblies, arranged in a square pattern, feature an external box and an internal "water box", with the pin bundle between the two. A mixing wire stirs the water in the bundle.

Fuel assemblies are grouped into clusters of nine assemblies; each cluster has a base that supplies water to the nine assemblies and a head used for handling operations. There are five different types of "cluster", in which the fuel rods differ in their level of U235 enrichment (3%, 4%, 5%, 6% or 7%); furthermore, the rods in the corner slots of each cluster are enriched 1% less than the other rods (see Figure 52).

Half of the water entering the core circulates from top to bottom through the internal boxes of all 1404 fuel assemblies and then flows around the external "water" boxes. Upon exiting the external boxes of the fuel assemblies, the water is collected and subsequently used to cool the lateral reflectors, again circulating from top to bottom: this fulfils the core moderation function as the water density is high (the water is not heated to a great extent as it is not in contact with the fuel rods).

This water is then mixed, in the "core inlet chamber", with the remaining half of the water from the "downcomer". This water then circulates upwards through the bundles in an initial group of 52 clusters (i.e. 468 fuel assemblies) located in the centre of the core, and upon exiting is mixed in the upper mixing chamber. When mixed, the water flows downwards through the bundles in a second group of 52 clusters ("Superheater 1"), and is then mixed again. Lastly, the water circulates upwards through the bundles forming a third group of 52 clusters located around the core perimeter ("Superheater 2"). Upon exiting the various fuel assemblies, it is mixed one last time in the core outlet chamber, exiting at a uniform temperature of 500°C.

Reactivity is controlled by 156 control rod assemblies (one assembly per cluster). Each assembly contains five control rods, each of which fits into the water boxes of the five central fuel assemblies in the cluster: in Figure 52, the control rod assemblies inserted into the core at the start of its life are shown in red.

Cluster type	U235-enrichment of most rods	U235-enrichment of the four corner rods
1	4%	3%
2	5%	4%
3	6%	5%
4	7%	6%
5	3%	2%

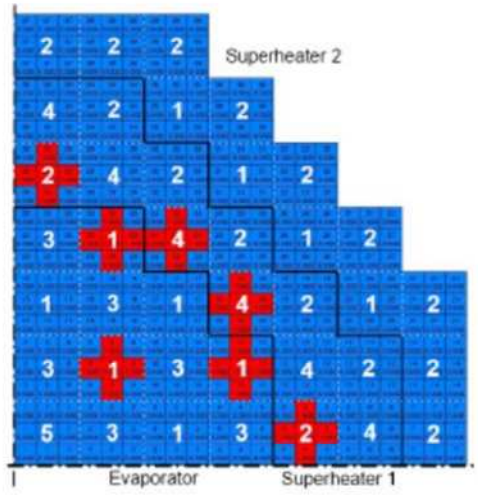

Figure 52: Cluster positions in the core and U235-enrichment of control rods

IRSN notes the complexity of the core design with regard to both nuclear and thermal-hydraulic studies, and the very strong coupling between the two aspects. In particular, effective mixing in the mixing chambers and stable flow rates through the various assemblies are essential. Note that the maximum temperature adopted for the cladding has a significant influence on core design: for example, adopting a limit of 700°C might simplify the design by removing one of the mixing levels described above.

7.2.5 *BARRIERS*

This concept features four barriers.

7.2.5.1 First barrier

The first barrier consists of the fuel element cladding. The choice of cladding material is a key consideration for SCWR safety, considering the highly corrosive nature of supercritical water, irradiation effects, the operating temperature of the cladding and the loads exerted by the high pressure of the coolant. Zircaloy must not be used as severe oxidation occurs after a few hundred hours of operation. Nickel alloys are currently being studied.

The designers intend to initially pressurise the gap between the pellet and cladding to between 7.5 and 8.5 MPa, to exert partial back-pressure that will counter the coolant pressure, thereby limiting the loads on the cladding.

A wealth of operating experience feedback relating to coal-fired power plants operating at high temperatures of 650°C exists. IRSN has no information regarding the scope for applying this experience feedback to SCWRs: firstly, the materials are different, and secondly, the water pipes have much thicker walls (7 to 8 mm) than the fuel element cladding (0.5 to 0.7 mm). Lastly, no results relating to irradiating environments are currently available.

7.2.5.2 Second barrier

In normal operation, the second barrier includes the reactor vessel, the pipes that carry hot water to the turbine and the pipes that return cold water to the vessel. Span sections of the reactor vessel are 45 cm thick. If the primary circuit isolation valves are closed, this barrier includes only the reactor vessel and the pipe sections between the vessel and the valves.

As the vessel only comes into contact with cold coolant, operating at a temperature similar to that of a PWR, the HPLWR designers intend to adopt the same material for the vessel as in the PWR. However, for the steam line (which operates at a steam temperature of 500°C), a P91 (X10 CrMoVNb 9-1) ferrito-martinsitic steel is being considered (cf. the paper in reference 88).

IRSN stresses that the thickness of the vessel may limit the scope for in-service inspection, as well as potentially reducing in-service inspection capabilities.

7.2.5.3 Third barrier

The third barrier (see Figure 53) consists of a cylindrical prestressed concrete containment designed to withstand an internal pressure of 0.5 MPa. This containment houses, in particular, the reactor vessel and wet and dry chambers designed to decrease pressure in the event of a feedwater line break or supercritical steam line break.

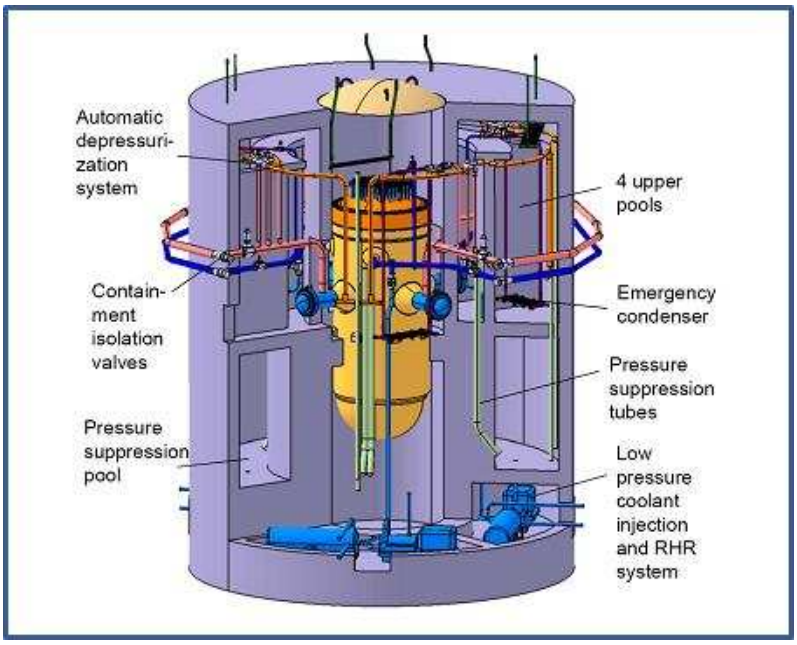

Figure 53: HPLWR reactor containment

The reactor containment is penetrated by four feedwater pipes and four outgoing pipes carrying supercritical steam at 500°C. Each pipe is fitted with two isolation valves: one inside the containment and the other outside. This reactor containment design is very similar to that adopted for the Generation III KERENA BWR concept developed by AREVA. There will also be a ventilation and filtration system.

7.2.5.4 Fourth barrier

The fourth barrier consists of the reactor building, which contains the cylindrical containment and various decay heat removal systems. The purpose of this barrier is, firstly, to limit environmental releases in the event of a severe accident resulting in a failure of the third barrier, and secondly, to protect safety-classified equipment against external hazards.

7.2.6 *FUEL ASSEMBLY HANDLING AND STORAGE*

The intended fuel handling and storage systems are similar to those used in BWRs.

7.3 CONTROL OF SAFETY FUNCTIONS

7.3.1 *REACTIVITY CONTROL*

Feedback coefficients

Most SCWR projects are thermal-spectrum reactors. In this configuration, the void effect is negative, as draining the water causes the nuclear reaction to shut down. However, reactivity is affected by the complex route travelled by the circulating coolant, which passes through the core several times. Detailed studies will be necessary in order to demonstrate proper control over the reactivity control function. Furthermore, the existence of areas with different enrichment rates complexifies nuclear calculations, particularly in the case of localised variations in temperature and by implication, density.

Fast-spectrum concepts, such as the Japanese "Super Fast Reactor" described in Section 7.1.2, are obtained by decreasing the quantity of water circulating through the core. This type of reactor is able to breed plutonium by using fertile assemblies placed in the middle of the fissile assemblies. A proposed core optimisation would ensure that the void effect associated with any fissile assembly would always have a negative effect on reactivity. This would be achieved, in particular, by inserting a moderating material (zirconium hydride - ZrH) into fertile assemblies (see Figure 54): the ZrH layer slows the neutrons exiting the fuel assembly before they penetrate the fertile assemblies; slowing them decreases the number of fast fission events and increases the rate of neutron capture in the fertile assembly, helping to decrease the local void effect.

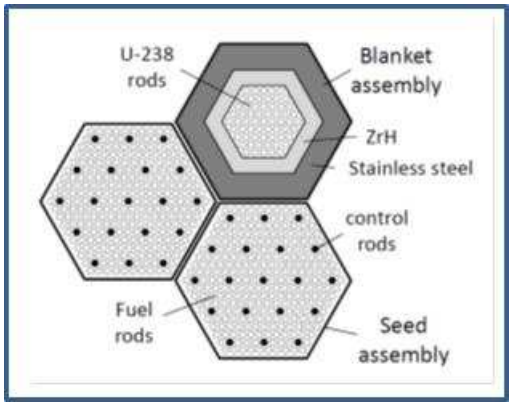

Figure 54: Diagram of the "Super Fast Reactor" core

Means of reactivity control

The control rods are of a similar design to those used in PWRs. In the HPLWR, as in BWRs, boric acid cannot be used to adjust fluctuations in reactivity during operation, but a boron injection system is included for use in case of accident. Consequently, the anti-reactivity potential of the control rods is greater than in a PWR.

The inadvertent control rod withdrawal or rod ejection accident case is therefore likely to be more penalizing (see Section 7.4.2.1).

7.3.2 *DECAY HEAT REMOVAL*

An emergency injection system (the "coolant injection system" – see Figure 45), very similar to that adopted for the BWR, is designed to cool the core if the normal cooling systems are unavailable (if the feedwater pumps are unavailable or the containment isolation valves are closed). The vessel can be depressurised by the "Automatic Depressurisation System" (ADS), in which cases the steam is condensed in a pressure suppression pool before being reinjected into the reactor vessel.

The HPLWR designers indicate that in case of loss of all feedwater pumps, rod drop is initiated by the resulting high pressure signal; the reactor must then be rapidly depressurised, as the loss of the feedwater pumps leads to a loss of flow circulating through the core within a few seconds. **Note that the water inventory in the vessel is not considered to be an essential safety parameter, in the absence of a closed cooling loop: the primary aim of the strategy is to maintain a flow rate through the core, using the water inventory present inside the containment.**

After around ten minutes, reactor cooling ceases unless the safeguard systems are activated. To increase the grace period in which to start the active systems, the SCWR designers are seeking to design passive systems, although the relevant research is still at a very early stage.

For the HPLWR, two passive systems have been proposed:

- One system, inspired by the system adopted for the BWR (shown on the left in Figure 55), actuates a turbopump that reinjects water initially present in a pool into the reactor vessel.
- A second system (shown on the right in Figure 55) features a loop in which steam from the vessel condenses in pools located in the upper part of the containment and is then reinjected into the core. The quantity of water may be increased using steam injectors (not shown in Figure 55). No detailed studies of this system have yet been conducted, but it is similar to the isolation condenser systems implemented in some existing BWRs.

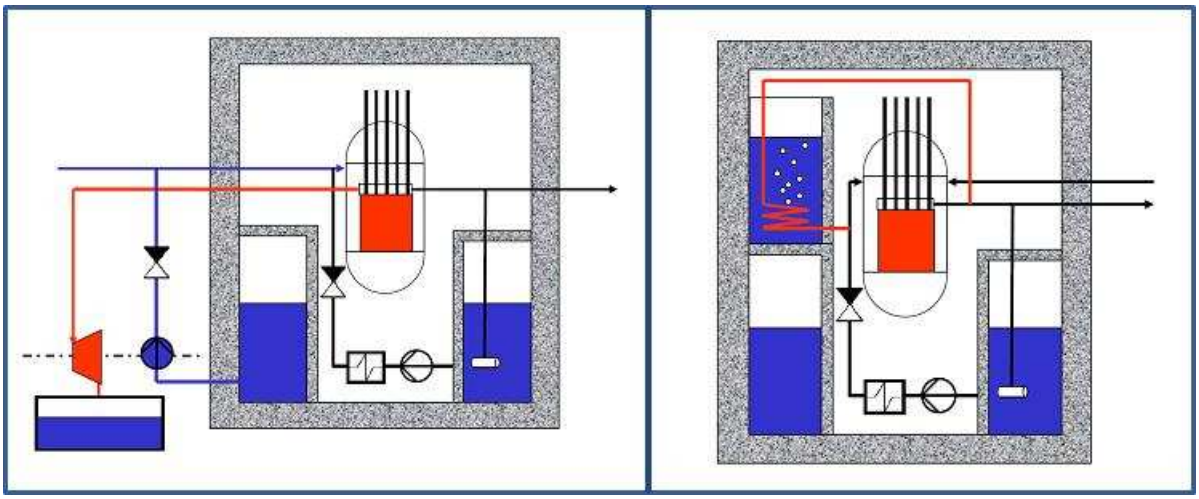

Figure 55: Passive decay heat removal systems proposed for the HPLWR

7.3.3 *CONFINEMENT OF RADIOACTIVE MATERIALS*

The foregoing information shows that the overall radioactive material confinement strategy is very similar to that implemented for BWR reactors. The containment space is inerted using nitrogen. The reactor building is presented as being a fourth barrier, unlike in current BWRs.

IRSN notes that the limited volume of the containment would cause the integrity of the third barrier to be compromised quasi-concomitantly with the loss of integrity of the second barrier in the event of loss of all decay heat removal systems in the containment.

7.4 RISK ANALYSIS

7.4.1 *RISKS INHERENT TO THE USE OF SUPERCRITICAL WATER*

Coolant-cladding heat transfers

Heat transfers between the coolant and cladding remain a major research topic, due to the significant changes in the thermodynamic properties of water in the so-called pseudo-critical domain. Around pseudo-critical conditions, significant variations in heat transfer between the fuel and water can occur under certain specific thermal-flux and mass-flow rate conditions.

Much research exists on the subject, showing that heat exchanges deteriorate at high thermal fluxes or low mass flow rates. Thus, it can be assumed that the consequences of an abnormal event leading to an increase in thermal flux or a decrease in mass flow rate could be amplified if the values that trigger the heat transfer deterioration phenomenon were reached. This phenomenon may occur abruptly, leading to an equally sudden rise in cladding temperature. While the extensive research has established a certain number of laws that can be used to characterise the appearance of this phenomenon and its consequences, development efforts are being pursued on this subject, due to the complexity of the SCWR reactor core design. The aim is to explore in detail the geometrical configurations and mass flow rate distributions that could be encountered. Modelling the heat exchanges associated with depressurisation transients is even more complex.

Researchers in Canada and Japan have built models to study the change in heat transfer coefficients in various pin lattice configurations.

Water chemistry and material corrosion

Supercritical water is a very powerful oxidising agent, used in industry as a solvent. This phenomenon diffuses oxygen atoms into the base metal, particularly at higher temperatures.

Numerous studies are currently being conducted with the aim of better understanding the specific chemistry of this medium, informed by experience feedback relating to coal-fired power plants. The corrosion resistance of various steels in a supercritical water medium has been tested. If the pressure increases, water can accelerate the metal dissolution process due to physicochemical properties that increase the solubility of ionic species in the medium. This would prompt inter-granular corrosion, in extreme cases resulting in cracking in carbon steels.

Furthermore, irradiated supercritical water can cause radiolysis phenomena different to those observed in existing water reactors.

7.4.2 *ACCIDENT SEQUENCES*

7.4.2.1 Inadvertent control rod withdrawal or rod ejection

Inadvertent control rod withdrawal and control rod ejection accidents were examined as part of the HPLWR safety studies.

Inadvertent control rod withdrawal

The consequences of inadvertent withdrawal of a control rod assembly containing five control rods without reactor trip were studied in order to determine the maximum temperatures reached by the fuel pellets and cladding. Depending on the position of the rod assembly in the core, the fuel may reach temperatures significantly above its melting point, and the cladding temperature can reach 1000°C, well above the temperature at which buckling can occur (850°C).

Control rod ejection

The temperature reached by the fuel is very close to its melting point and the cladding reaches 860°C, very slightly above the temperature at which buckling can occur.

7.4.2.2 Loss of coolant accidents

Loss-of-coolant accident (LOCA) studies are particularly important from a safety perspective, due to the high pressure of the supercritical water, low mass in the reactor vessel and unusual depressurisation sequence. Considerable uncertainties also exist in relation to the value of the cladding-water heat transfer coefficient during the depressurisation phase. Lastly, there is a risk of flow inversion in some core cooling channels.

The studies focusing on a guillotine rupture in a reactor vessel water inlet or outlet pipe assumed that the isolation valves were closed, the control rods dropped and the emergency injection system activated. The most severe condition is a guillotine rupture in an inlet pipe, leading to cladding temperatures of around 850°C within a few seconds of the onset of the transient.

7.4.2.3 Severe accident

Research carried out to date has focused on design-basis accidents. Severe accidents have not been studied. The main assumptions relating to the severe accident behaviour of the SCWR are described below, but are subject to confirmation by detailed studies.

The SCWR should behave similarly to current LWRs in the event of a severe accident involving corium falling on the reactor bottom vessel head and (rapid or delayed) vessel failure, unless it is cooled from the exterior (as in the AP1000 or the Finnish reactor at Loviisa) and the effectiveness of such cooling measures demonstrated. As the control rods are located above the core, a core catcher similar to the one used in the EPR could be designed.

The interaction between superheated steam and steel structures will produce hydrogen. However, hydrogen production should be more limited than in a PWR, due to the absence of Zircaloy.

The risk of a steam explosion also exists with the SCWR, possibly with specific implications, in view of the possible physical states of the water at the time of interaction (i.e. supercritical or not).

Lastly, in the case of the fast-spectrum SCWR, the risk of recriticality of the molten core must be taken into consideration when designing severe accident management provisions.

7.5 ENVIRONMENTAL IMPACT AND RADIATION PROTECTION

Primary coolant composition

Corrosion resistance test revealed that some elements in the tested alloys are released from the material in greater quantities than others. Such elements include manganese (Mn), molybdenum (Mo), tungsten (W) and aluminium (Al), in the case of Alloy 625 and Hastelloy. Some of the tested alloys contain molybdenum. Molybdenum dissolving into the reactor coolant could form technetium isotope 99 m.

These chemicals as well as activation products relating to these dissolved elements and tritium (see below), will have to be given due consideration when determining the composition of the liquid waste to be processed and assessing the potential releases in accident conditions.

Radiation protection

Using a direct energy conversion cycle entails strict requirements in terms of fuel cladding leaktightness (as in current BWRs), in order to limit worker exposure. However, the extent of the second barrier in normal operation, which includes the turbine, is unfavourable from this point of view. For example, the OECD report in reference 87 states that the annual professional collective dose for a BWR is approximately double that for a PWR.

Concerning maintenance operations, equipment accessibility and actual performance of maintenance operations should be broadly comparable to those for a PWR or BWR. However, due to the corrosive nature of supercritical water, such operations may be more frequent on an SCWR, which is unfavourable from a radiation protection perspective.

Note that in PWRs and BWRs, most of the tritium produced by ternary fission reactions remains trapped on the inner surface of the Zircaloy cladding, in the form of zirconium hydride. This will not be the case for the SCWR, based on the cladding materials currently envisaged. This point is particularly significant in the SCWR design, as water enters the turbine after passing through the core.

Lastly, as for the other processes, minor actinide transmutation in the reactor will have a negative impact on radiation protection.

Liquid and gaseous waste management

The source of effluents in an SCWR and the related isotope compositions should be comparable to those for the PWR or BWR, apart from the special case of tritium, as explained above. This also applies to accident-related releases.

7.6 MATURITY OF CONCEPT AND R&D REQUIREMENTS

7.6.1 *OPERATING EXPERIENCE FEEDBACK*

No SCWR nuclear reactors have been built to date, but extensive experience feedback relating to fossil-fuel power plants operating with supercritical water is available. However, although this experience feedback is valid for the conventional part of the plant, and has yielded advances in terms of turbine technology and corrosion prevention provisions, it cannot be directly transferred to a nuclear reactor (very different water tube geometry, no irradiation-induced radiolysis, pressure and temperature differences over the cycle, etc.).

Experience feedback acquired from light water reactors, particularly BWRs, may also be used, not least in relation to the design of the safeguard systems.

7.6.2 *R&D REQUIREMENTS*

7.6.2.1 Reactor core design

Designing an SCWR core (whether thermal-spectrum or fast-spectrum) is highly complex, owing to the need to mix the coolant multiple times in order to limit the cladding temperature in the hottest channel and the strong coupling between the nuclear and thermal-hydraulic studies.

This design difficulty is also noted in the CEA document reference 6. It will undoubtedly make the safety demonstration more difficult.

7.6.2.2 Cladding material

To date, only cladding materials compatible with maximum cladding temperatures in the region of 550°C could be used for the SWCR, given the stress levels and the corrosive effects of supercritical water, but for the HLWR, for example, the target maximum cladding temperature is 630°C. Furthermore, the behaviour of such materials when subjected to irradiation remains to be studied. This issue is also noted in the CEA document reference 6.

Additionally, as stated in Section 7.2.4, increasing the maximum cladding temperature to 700°C would enable the core design to be simplified by limiting the number of mixing areas in the core. To this end, various austenitic, ferrito-martensitic and oxide dispersion-strengthened (ODS) steels are currently being tested in Japan, Canada, China and Europe. Some aluminium-containing ODS steels appear promising (cf. the paper in reference 89).

IRSN also notes that the presence of certain elements in the cladding material will require a better understanding of the relevant nuclear data.

An experimental loop featuring fuel rods representative of the HPLWR concept is in the process of being installed in the LVR-15 reactor at the nuclear centre in Řež (Czech Republic), as part of the EC's FP7. The purpose of this loop includes studying the effects of radiolysis on water chemistry and structural corrosion.

7.6.2.3 Accident modelling

Modelling depressurisation transients in the primary circuit is very complex and requires detailed models of the heat transfer coefficient between the cladding and the water.

Although some transients show that cladding failures or fuel melt are a possibility, no studies describing the behaviour of the fission products in the containment exist. The same applies to studies of molten fuel behaviour in the reactor vessel; furthermore, the need for a core catcher has not been examined. Nevertheless, the various phenomena are likely to be similar to those observed with the BWR.

7.7 CONCLUSION

The supercritical water-cooled reactor is presented as being an upgrade to current light water reactors. Its main benefit is economic, as the planned operating temperatures make a target efficiency of around 45% a possibility. It is the only reactor shortlisted by GIF that uses water as its coolant.

It benefits from the very good heat capacity of supercritical water, which limits the mass flow in the core and hence the required pumping power. Using supercritical water as a heat transfer fluid also avoids any liquid-vapour phase change problems, such departure from nucleate boiling and dry-out, which are limiting factors for the PWRs and BWRs in operation.

Although supercritical water has been used for many years in coal-fired thermal power stations, using it in a nuclear reactor raises numerous issues.

Firstly, the chemistry of supercritical water must be better understood, and in particular its behaviour when subjected to a neutron flux (radiolysis effect). In addition, the very unusual behaviour of water in the "pseudo-critical" region — with significant variations in its thermodynamic properties depending on the heat flux in the fuel and its mass flow rate — would also require considerable research, in particular regarding the normal reactor startup and shutdown transients. Special attention and significant modelling efforts will be required with regard to depressurisation accidents, which would cause the water and vapour phases to separate and potentially cause large variations in the heat transfers, depending on the steam quality of the mixture. The complexity of the core and the strong coupling between the thermal-hydraulic and nuclear aspects add to the difficulty in demonstrating the safety of the concept.

In addition to these considerations, it will not be possible to demonstrate the feasibility of the concept until certain technological difficulties have been overcome. The main challenge relates to the materials used for the fuel cladding and reactor internals. Lastly, the strongly oxidising nature of supercritical water must also be taken into consideration.

It should be noted, however, that as evolutionary reactors, they benefit to some extent from experience feedback from the PWRs and BWRs, in particular regarding safety system design.

In contrast, the R&D programs overseen by GIF are innovative and, even if they do not lead to an industrial project in the short or medium term, might yield improvements to the current PWR and BWR concepts, for example in the areas of cladding materials (without Zircaloy), core design and/or passive systems (steam injectors, etc.). Accordingly, reactors with optimised designs (compared to current reactors) may be developed in the medium term.

Based on current knowledge, IRSN is unable to issue a verdict regarding the ability of the SCWR to achieve a significantly higher safety level than that targeted by the Generation III reactors currently under construction. In particular, the strong coupling between the nuclear and thermal-hydraulic aspects may result in core instability, which is particularly undesirable in fast-spectrum concepts, when the water is not maintained in a supercritical state.

8. COMPARISON OF THE SIX REACTORS SELECTED BY GIF

Adopting a thematic approach, this chapter places the concepts described in the previous chapters in perspective. In order to define which themes to examine, IRSN focused on the possible safety orientations adopted for reactors deriving from these systems. Although it would be premature to set overall safety objectives for the various reactors, the lessons learned from the Generation III reactor safety analysis and from operating experience feedback (particularly feedback relating to the Fukushima accident of 11 March 2011) have already revealed a number of areas and themes where improvements are necessary. IRSN has selected some of these themes as the basis for a cross-concept examination of the nuclear systems shortlisted by GIF.

Most of the chosen points of comparison are qualitative in nature, and subject to caution, due in particular to the differences in design between the various systems and the disparities in terms of the related scientific understanding, research and studies.

The following chapters focus on aspects relating to system fuel cycles and ability to transmute high activity, long-lived radionuclides.

This chapter opens with a concise summary of the key features of each of the six reactor concepts, and then describes possible orientations for Generation IV reactors and the themes adopted for the analysis.

8.1 OVERVIEW OF REACTOR FEATURES

The main operating characteristics of the reactors associated with the nuclear systems selected by GIF are presented in the two tables below. The equivalent features of the Flamanville EPR (EPR FA3) have been added for the sake of comparison.

Table 3: Key features of the reactor concepts examined by GIF

	SFR	**VHTR**	**GFR**	**LFR**	**MSR**	**SCWR**	**PWR (EPR FA3)**
Neutron spectrum	Fast	Thermal	Fast	Fast	Fast or Thermal	Fast or Thermal	Thermal
Power density (MW/m^3)	300	5 to 10	100	100	330 (for fast spectrum)	100	92
Moderator	-	Graphite	-	-	Graphite for thermal spectrum	Water or heavy water for thermal spectrum	Water
Coolant	Sodium (liquid)	Helium (gas)	Helium (gas)	Lead or LBE (liquid)	Liquid salt (which is also the fuel)	Supercritical water	Water
Experience feedback	Approx. 20 reactors worldwide (400 reactor-years)	7 HTR already operated and 2 HTR under construction	None	No civilian reactors A few submarine propulsion units	Two thermal-spectrum reactors built	None	272 PWRs in operation worldwide, including 58 in France

Fuel							
	SFR	**VHTR**	**GFR**	**LFR**	**MSR**	**SCWR**	**PWR (EPR FA3)**
Fissile/fertile materials	$^{235}U/^{238}U$ $Pu/^{238}U$	$^{235}U/^{238}U$ $Pu/^{238}U$ $^{233}U/^{232}Th$	$Pu/^{238}U$	$Pu/^{238}U$	$^{233}U/^{232}Th$ $^{235}U/^{232}Th$ $Pu/^{232}Th$	$Pu/^{238}U$	$Pu/^{238}U$ $^{235}U/^{238}U$
Physical state	Pellets	Particles	Pellets	Pellets	Liquid	Pellets	Pellets
Chemical compound	Oxide Metal	Oxycarbide Oxide	Nitride Carbide	Oxide Nitride Carbide	Fluorides	Oxide	Oxide

Table 4: Coolant specifications

	SFR	**VHTR**	**GFR**	**LFR**	**MSR**	**SCWR**	**PWR (EPR FA3)**
Maximum coolant pressure (MPa abs)	≈ 0.6	≈ 7	≈ 7	≈ 1.5	≈ 0.5	≈ 25	≈ 15.5
Boiling point (°C)	880 at 0.1 MPa	-	-	1745 for Pb 1670 for LBE at 0.1 MPa	≈ 1800 at 0.1 MPa	-	345 at 15.5 MPa
Solidification point (°C) at 0.1 MPa	98	-	-	327 for Pb 125 for LBE	≈ 560	-	-
Coolant temperature range in core (°C)	400-550	250-1000	400-850	400-480	700-770	280-500	295-330

8.2 ANALYSIS THEMES

8.2.1 *SAFETY ORIENTATIONS FOR GENERATION IV REACTORS*

IRSN first describes the safety objectives adopted by GIF for the purpose of selecting so-called Generation IV nuclear systems (reference 13) as well as the safety orientations proposed by the "Comité programme de préparation du future" (CPPF) under the three-way R&D cooperation agreement between Areva, CEA and EDF, with a view to developing a French SFR process (reference 9).

Based on this information and its current thinking, IRSN goes on to describe the principal safety orientations that it considers should be targeted for reactors slated for construction by 2050.

ASN's basic position on this subject is set out in a statement (reference 4). ASN *"considers that […] the process selected for the development of a fourth generation of reactors in France,[…], should deliver a significantly higher level of protection of the interests stated in Section L.593-1 of the Environmental Code than Generation III reactors."*

Safety objectives adopted by GIF

As stated in the introduction to this report, GIF has defined a number of objectives with the aim of selecting a limited number of Generation IV nuclear systems and coordinating the relevant R&D activities. These objectives are as follows:

- Pursue the improvements in competitiveness and safety achieved with Generation III water-cooled reactors.
- Make better use of uranium resources.
- Decrease radioactive waste, especially long-lived high-level waste (LLHLW).
- Enhance protection against malicious acts and attempted theft or misuse of nuclear materials.

GIF has adopted the following three basic rules regarding safety and radiation protection:

- Systems shall deliver very good performance in terms of safety and reliability;
- They shall cause very little damage to the core and the risk of occurrence of such damage shall be very low.
- There shall be no need for action to protect the off-site population.

More detailed objectives were subsequently proposed in reference document 13:

- Apply the As Low As Reasonably Possible (ALARP) principle, in particular by harnessing technological innovations.
- Adopt safety objectives equivalent to or stricter than those for Generation III reactors.
- Adopt accident prevention rather than mitigation measures, in particular by prioritising a strategy that strives for "practical elimination" of conditions in which environmental releases may occur, rather than significantly enhancing confinement performance.
- Define a robust safety architecture, based in particular on good intrinsic system characteristics and appropriate technical options to cover the first four levels of defence in depth and ensure that they are mutually independent.
- Implement very reliable safety systems, making extensive use of passive systems and safety-enhancing behaviours intrinsic to the concept (such as negative neutron feedback, gravity effects and thermal inertia).
- Limit human intervention by using innovative measurement and monitoring systems.
- Enhance the robustness of the safety demonstration by exhaustively defining the risks associated with the installation.

Safety orientations proposed by the RNR-Na Project

The report reference 9, issued in 2009, sets out "*the general safety principles and objectives for future reactors*" defined by CPPF. The report states "*Regarding the safety objectives applicable to future reactors, and the SFR in particular, the objectives adopted for the EPR™ in France are considered to be already very ambitious, ensuring a very high standard of protection. Stipulating an additional reduction in risk compared to this level, in particular in terms of probability of occurrence, would be unwarranted and might even be counterproductive … but this is not incompatible with the definition of more ambitious R&D orientations, by GIF for example.*" A number of potential areas for improvement with respect to the EPR have been identified, however: "*The potential for improving risk prevention will have to be examined, with particular focus, within the studied scope, on managing uncertainties, checking that threshold effect margins are adequate and minimising common-mode phenomena."* The report also states: *"In particular, […] 'forgiving' solutions that provide grace periods in which to take corrective action or undertake repairs shall be preferred.*"

In the light of technical discussions with IRSN in 2013 (see meeting report reference 10), the RNR-Na Project updated these orientations (reference 11). An overview of these updated orientations is given below, although they do not refer specifically to the SFR concept:

- Prevention of core melt: each sequence potentially leading to generalised core melt shall be made very improbable. The overall melt frequency target, allowing for uncertainties and all types of failures and hazards shall be equivalent to that adopted for Generation III reactors.
- Management of the consequences of severe accidents: the stated objective is to limit the consequences associated with severe accidents to a level "*equivalent to or better than the EPR™*", i.e.: "*the scope of any measures to protect the population as a consequence of the maximum conceivable releases* [associated with the considered core melt conditions] *shall be very limited"* and return the installation to an end state in which the safety functions are durably performed. This objective should be pursued for Generation IV reactors, regardless of the concept adopted. Only generalised core melt conditions for which the designer is unable to identify reasonable mitigation provisions shall be "practically eliminated" by implementing specific risk prevention provisions.
- Consideration for natural hazards: suitable provisions shall be implemented to protect resources that perform functions enabling the avoidance of significant or early radiological consequences, including in the event of a severe accident or reference beyond design-basis natural hazard.
- Autonomous operation and grace periods; the design shall result in:
    - grace periods appropriate to the internal capabilities of the site, enabling the correction of "drift" that might otherwise lead to unacceptable consequences;
    - capacity for autonomous operation consistent with the capabilities of off-site emergency response teams in the event of a beyond design-basis external natural hazard;
    - means of working on a defective system, even if the installation is in a degraded state;
    - instrumentation delivering key information enabling the state of the installation to be assessed while it is in a degraded configuration.

IRSN position

IRSN approves the general objectives adopted by GIF in relation to the safety of Generation IV reactors, and in particular the goal of encouraging development of reactors whose safety is partially based on intrinsically safe characteristics (neutron feedback, thermal inertia, no risk of high-energy phenomena, etc.). This choice should ultimately enable simpler reactors to be designed, as fewer, less complex systems should in theory be necessary in order to achieve a given safety standard. This should have a positive impact on operational safety (in terms of long-term compliance, maintenance operations, etc.). The goal of limiting consequences to levels requiring no specific action to protect the off-site population should also be pursued at this stage. In particular, it would appear to be a useful reference for guiding future R&D initiatives in the context of the development of Generation IV reactors.

IRSN does not, however, share the objective of giving preference to provisions that aim to "practically eliminate" severe accidents over provisions to mitigate their consequences. The Institute also considers that provisions intended to enhance core melt prevention and severe accident mitigation provisions should be implemented, minimising the number of conditions that must be "practically eliminated".

Concerning the orientations adopted by CPPF and subsequently described by the RNR-Na Project in 2013, further discussion will be required, in a framework yet to be determined; IRSN considers that the safety orientations identified thus far should, after being more fully defined, be adopted in the ASTRID project, which will in turn provide inputs for defining the safety objectives for a future Generation IV SFR process. In their current form, these orientations are not consistent with the ASN position in favour of a significantly higher safety level than that adopted for Generation III reactors. Based on current understanding and the available research and studies, it is too early to judge whether a safety level "significantly higher" than that targeted for the EPR pressurised water reactor is realistically achievable.

IRSN considers that an assessment should be undertaken, to identify the areas liable to yield significant safety gains, taking into consideration the upgrades already implemented for Generation III reactors. Although it is not easy to conduct such an assessment without referring to a particular reactor, it is nevertheless possible to identify areas for improvement. Generation III reactors incorporate significant advances in terms of safety, in particular concerning the consideration of multiple-failure conditions, internal and external hazards and severe accidents; these advances should be treated as prerequisites for Generation IV reactors. Furthermore, based on experience feedback yielded by the safety analysis centred on Generation III reactors as well as the various lessons arising out of the Fukushima accident, new areas for improvement should be identified and, when the time comes, implemented in a manner appropriate to each individual Generation IV concept. Already, new changes to the safety approach are expected, in particular in terms of protecting nuclear power plants against natural and man-made external hazards, mitigating the short-, medium- and long-term consequences of accidents, and more generally, addressing conditions that, although highly improbable, might result in massive environmental releases.

In this context, we should strive to make safety system architectures more robust, for example through appropriate combinations of active and passive systems, ensuring that safety systems deliver a high level of reliability while being less sensitive to disruption of off-site resources such as the power grid or heat sink. To this end, an iterative design optimisation approach should be adopted, with the aim of ensuring that design choices are sound. Significantly upgrading provisions relating to the "radioactive product confinement" function (for liquid

discharges as well as atmospheric releases) – for example by building a geotechnical barrier, should also yield major gains in accident mitigation terms.

Lastly, more effectively including organisational and human factors in the design is also a major area for improvement. The facility should be designed not as a technical system but as a sociotechnical system, the correct operation of which is dependent on interaction between technical processes and individual and collective human activities. This aspect further underscores the importance of selecting concepts with intrinsically safe characteristics.

In conclusion, IRSN considers - from a safety perspective - that when selecting a Generation IV process, preference should be given to concepts that feature:

- intrinsically safe characteristics;
- long grace periods, enabling appropriate corrective action to be taken for all foreseeable accident conditions;
- a robust safety system architecture that ensures that the various levels of defence in depth are properly independent and not overly reliant on the availability of external sources (cooling, electricity, etc.), in particular by including appropriate combinations of active systems and passive systems;
- ability to operate autonomously for several days in case of accident, enabling the safety functions to be performed pending the arrival of off-site emergency response teams, including in the event of external hazards that are very severe in terms of both intensity and duration; in this respect, the risk of domino effects across the site must be given proper consideration;
- enhanced protection against external hazards;
- design provisions to enable safe operation of the facility (optimised design, instrumentation, etc.);
- limiting the short-, medium- and long-term impacts of the installation on humans and the environment, both in normal operation (limiting effluent and waste volumes and activity levels, etc.) and in accident operation, including in multiple-failure conditions and severe accidents. This will require a notable improvement to the (chemically and/or radiologically toxic) hazardous substance confinement function, as well as upgraded accident prevention provisions; environmental impacts, radiation protection for workers, decommissioning issues and waste will all need to addressed, taking all stages of the life cycle into consideration.

During the technical assessment process, the RNR-Na Project indicated that its objectives for a Generation IV reactor were in line with those envisaged by IRSN. The Project also stated that the objectives will guide the French R&D efforts already underway for the SFR process and that it will advocate for them in the international arena, in particular through GIF Risk and Safety Working Group.

8.2.2 *THEMES ADDRESSED BY THE CROSS-CONCEPT ANALYSIS OF THE REACTORS ASSOCIATED WITH GENERATION IV NUCLEAR SYSTEMS*

In view of the foregoing and the fact that the designs currently being researched might not necessarily be those ultimately adopted as Generation IV reactors, IRSN sought to assess:

- the safety characteristics of the various reactors;
- the hazard potential of the systems, taking into account the substances used and/or produced during the operating life of a facility;
- the technological difficulties identified to date, which appear to rule out the transition to industrial scale by the target date of 2050 and the topics in need of further research for the purpose of the safety demonstration.

Concerning the safety characteristics, most of the chosen themes relate to the aforementioned safety orientations and reflect the nature of the studied concepts. IRSN therefore selected themes that focus on the positive or negative impact of the characteristics of the different reactors with regard to most of the desired orientations.

Ultimately, IRSN selected the following themes:

- nuclear characteristics and reactivity control (Section 8.3);
- sensitivity to loss of cooling (Section 8.4);
- hazardous material confinement function (Section 8.5);
- operational safety (Section 8.6);
- in-service inspection (Section 8.7);
- severe-accident behaviour (Section 8.8);
- toxicity of products used (Section 8.9);
- sensitivity to hazards (Section 8.10), specifically seismic hazards, as the means of protecting installations against other external and internal hazards are not directly linked to the reactor concept;
- maturity of the concepts and the related technological challenges (Section 8.11);
- issues requiring additional research for the purpose of the safety demonstration (Section 8.12).

8.3 NUCLEAR CHARACTERISTICS AND REACTIVITY CONTROL

First and foremost, it is important to reiterate that the purpose of GIFs objectives is not only to improve reactor safety and energy efficiency, but also to optimise natural resource use and minimise waste (particularly long-lived waste). The latter objectives require the complete fuel cycle including front-end installations and back-end installations of the reactor to be taken into consideration. The nuclear aspects of each studied concept cannot be dissociated from the objectives in terms of fuel depletion, plutonium recycling, transmutation and breeding. Comparing the safety of the various concepts, from a reactivity control perspective, is only relevant if the compared concepts have similar objectives in this area.

8.3.1 *SPECIAL CONSIDERATIONS RELATING TO FAST REACTORS*

Of the six concepts identified by GIF, four are fast reactors (SFR, LFR, GFR and MSR[40]); the SCWR concept is suitable for either fast-spectrum or thermal-spectrum operation; and the VHTR is a thermal-spectrum concept. The choice of a fast-spectrum concept is driven by the objective of optimising fissile resources. From the point of view of the "reactivity control" safety function, comparing concepts is only really meaningful within a particular family: thermal-spectrum reactors or fast-spectrum reactors. Few thermal-spectrum concepts have been selected by GIF as they do not, by themselves, enable the objectives in terms of optimising natural resource use and minimising waste to be achieved. Consequently, the remainder of this section focuses mainly on reactivity control in fast reactors.

Before addressing the characteristics of the various concepts from a reactivity control perspective, the significant quantities that influence core behaviour should be restated: **these are intrinsic characteristics that may have either positive or negative safety consequences, where applicable necessitating specific design provisions.**

Significant quantities

Regarding fast-neutron spectra, the "harder" the neutron spectrum in the core (i.e. the higher the proportion of fast neutrons) the greater the ratio between fissions and captures, and the more neutrons are produced by fission. This creates favourable conditions for fission by all actinides (U, Pu, Am, Np, Cm) and enables high breeding rates to be achieved. As a result, concepts with a hard spectrum will be used to consume a large quantity of plutonium and in some cases minor actinides (MA). Fission reactions involving plutonium isotopes generate more neutrons but also a smaller proportion of delayed neutrons than with uranium. This results in:

- a lower effective delayed-neutron fraction, and therefore a lower prompt criticality threshold;
- a lower average prompt-neutron lifetime and therefore a lower reactor period (hence potentially faster-developing prompt-critical power excursions).

In addition, hardening the spectrum decreases the Doppler effect and control rod worth.

There are two possible routes for designing fast reactors. The first seeks to maximise the burn-up in order to optimise consumption of the available fissile material (derived from spent fuel from water-cooled reactors or stockpiled or surplus military materials, for example). The second route aims to optimise the neutron balance to maximise the energy produced by unit of mass of natural uranium; this is achieved by harnessing the 238U to produce fissile plutonium. In the first case, the design features an incinerator-reactor able to produce energy from "available" transuranium elements (Pu and MAs). Concerning the neutron balance, this approach generally leads to a more significant reactivity swing over the operating cycle than with the second route, requiring relatively "high-worth" control rods to be inserted into the core at the beginning of the cycle. The risks associated with an inadvertent rod withdrawal accident are therefore greater than in a core in which the rods would remain largely withdrawn while the reactor was at power. The second route requires more frequent fuel reprocessing, generating additional risks at fuel cycle facilities. The nuclear characteristics of U-Th cycles are notable for the unusual effect of protactinium (Pa), which has a long half-life that must be taken into consideration, as well as the very limited minor actinide production.

[40] This applies, in particular, to the MSFR, which has been adopted as the reference MSR concept.

Using fuels with a large proportion of transuranium elements would require special efforts in order to ensure a good level of safety in normal operation and during accident transients. Introducing a large proportion of plutonium and minor actinides into the core tends to degrade the neutron feedback coefficients and the effective fraction of delayed neutrons, and increase decay heat. Furthermore, the fuel isotope composition would be subject to significant variability, influenced by multiple fuel cycle-related parameters. Such uncertainties will have to be included from the core design stage, failing which, core compliance with the safety studies may be compromised.

The neutron balance of the core is influenced by leaks, as well as fission reactions and captures. Geometry has a major effect on leaks. The greater the core surface-to-volume ratio, the greater the leaks. As a result, concepts with poor fission/capture ratios must minimise leakage, which can result in large cores. From a nuclear perspective, large cores have disadvantages in terms of monitoring and power distribution uniformity.

For a given power level, the greater the power density, the more "compact" the core, which makes for uniform power distribution and simplifies monitoring. Furthermore, in a fast reactor, compaction of the fissile material tends to increase reactivity. Thus, if the core geometry is already "compact" in normal operation, the risk of an increase in reactivity if the geometry is disrupted (due to degradation, earthquake, etc.) is more limited.

The type of fuel also influences the core nuclear characteristics. Oxide fuels are low-density, resulting in large cores and a less hard spectrum. Dense fuels make for more compact cores and a harder spectrum.

The "neutron transparency" of the coolant also affects the spectrum. The more neutron-transparent the coolant, the less susceptible it will be to density and temperature variations and the concomitant effects on core reactivity. On the other hand, a neutron-transparent coolant does not protect core structures from damage by irradiation, which tends to be particularly severe with a fast spectrum.

Lastly, it should be noted that, from a nuclear perspective, fast-spectrum reactors offer a few benefits over thermal-spectrum reactors, and in particular, a smaller reactivity swing over the operating cycle and the absence of neutron poisoning effects (such as xenon effects in water-cooled reactors).

<u>"Recriticality" issue</u>

The normal geometric configuration of a fast reactor is not its most reactive[41]. This means that disrupting reactor geometry might prompt an increase in reactivity. In particular, in degraded-core conditions, with accumulated quantities of molten fuel, criticality can be reached with even a relatively small mass of fuel. Note that, assuming the fuel characteristics to be equivalent, the larger the core, the greater the risk of recriticality in degraded configurations, due to the lower neutron leakage rate. **This recriticality issue is one of the key safety challenges for fast reactors. If the core becomes degraded, it is important to give consideration to the quantity and evolution of the power released by fission (in addition to the decay heat), in order to demonstrate the resistance of the barriers.**

[41] This concept does not apply directly to MSRs, due to the nature of the fuel salt. This aspect is addressed in relation to the MSR in Section 8.3.2.

IRSN attaches particular importance to preventing the risk of recriticality during accident transients. This requires the implementation of reactivity control mechanisms that are effective not only in normal conditions but also in degraded conditions, including in the most reactive core configurations. Furthermore, these mechanisms must be combined with core monitoring instrumentation that remains effective in all operating conditions.

Interpretation of reactivity measurements

Core reactivity may be influenced by a large number of parameters. For example, changes in the geometry or temperature of core components, reactor operating conditions or coolant circulation may all affect core reactivity. With fast-spectrum reactors, reactivity changes occur as the sum of multiple individual phenomena of comparable intensity but in some cases having opposing effects. Some effects cancel each other out, and interpreting a reactivity measurement, which by definition represents an integral core quantity, can be complex. In particular, it is not always immediately possible to identify the disruptive factor responsible for a change in reactivity. Such phenomena would be more marked in a large core, where localised reactivity effects are a possibility.

Note all the different core monitoring measurement systems implemented to provide information about core state and reactivity, as well as the spatial distribution of the neutron flux. This aspect is a particular challenge for large reactors.

8.3.2 *BASIS FOR COMPARING THE SAFETY OF NUCLEAR CONCEPTS - NUCLEAR ASPECTS*

In SFRs, which so far have always been built for breeding purposes, the sodium coolant, which is not totally neutron-transparent, can capture neutrons and slightly moderates the spectrum. As explained above, this is an advantage in terms of protecting structures and increasing the Doppler effect. However, any fluctuation in the density of the sodium in the core, even if it is very localised, influences the neutron spectrum, potentially resulting in positive coolant density feedback.

Turning to LFRs, the characteristics of lead - from a nuclear perspective - are relatively similar to those of sodium. Lead has a slightly more limited moderating effect, however.

Furthermore, the difference in “core compactness” between a lead-cooled core and a sodium-cooled core results in different nuclear characteristics. A lead-cooled core must be designed with large coolant cross-sections to limit the coolant velocity (see Chapter 5 devoted to the LFR). Accordingly, a lead-cooled core is a relatively “hollow” system: this is a disadvantage in terms of power distribution uniformity and stability, nuclear monitoring and the coolant density feedback effect.

In addition, some designers have considered installing steam generators inside the main reactor vessel, due to the lack of high-energy chemical interaction between lead and water. In such a configuration, an accident scenario involving water insertion caused by a steam generator line break must be considered in the safety analysis. Lastly, inserting control rods into lead might be problematic, due to the high density of lead.

The principle underpinning the design of the VHTR core, from a nuclear perspective, consists in ensuring that the power coefficient remains sufficiently negative, in all possible operating states, to ensure that the fuel remains intact without the need for active cooling. This objective can be achieved as a result of:

- the nuclear characteristics of graphite;
- limiting the density of the fuel particles in the core;
- optimising the quantity of fissile material in fuel particles.

The temperature feedback coefficient of graphite is strongly negative. As a result, any change in core temperature results in a significant variation in power. This phenomenon makes evaluating local power in the core of a VHTR more complex than in an SFR or LFR core (the relevant studies have not yet been conducted for the GFR). As stated in Section 3.6.1, operating experience feedback has revealed some difficulties in controlling local power output in large HTR cores (Fort Saint-Vrain and THTR). These problems are less challenging for the VHTR concept, which has a relatively small core.

Helium is a neutron-transparent coolant. In the GFR, the accident scenarios involving a change in coolant density have little effect on core reactivity.

Helium gas does not remove the heat produced in fuel as efficiently as a liquid metal. Consequently, the power density of the GFR core must be limited (to approximately 100 MW/m^3). As a result, a GFR core is larger than an SFR core of equivalent power, which is a disadvantage in terms of nuclear monitoring and power distribution stability and uniformity.

Some nuclear characteristics of the MSFR are unusual, compared to the other concepts. One of the strengths claimed by the designers relates to a fast-responding and strongly negative overall feedback coefficient, which should ensure that core behaviour is self-stabilising, enabling it to tolerate significant reactivity insertions (up to ~1000 pcm in 1 second[42], based on preliminary analyses), despite having only a small fraction of delayed neutrons.

The notion of compaction does not apply directly to an MSR, due to the incompressible nature of the salt; however, rapid positive reactivity insertion may occur in the event of an increase in the volume of fuel salt in the central cavity (expansion of structures, failure of the fertile blanket, etc.). Positive reactivity insertions may occur, at a slower rate, in the event of an uncontrolled addition of or change in the local concentrations of fissile material, a change in salt temperature or density, or an enhanced contribution to the chain reaction by delayed neutrons. In any case, based on current knowledge, no scenarios causing the core to become prompt-critical (~1000 pcm in 1 second) are foreseeable.

In addition, studies concerning the moderating effect of water on the fuel salt reveal that the criticality risk should be easily manageable within the subcritical area, in the event of interaction between the salt and water in the pool.

[42] Level of reactivity insertion that would make the reactor prompt-critical, taking feedback into account.

However, in view of the complexity of the phenomena involved and the strong coupling between the nuclear, thermal-hydraulic and thermochemical aspects, robustly demonstrating the safety of the concept will require a special effort. The nuclear parameters are wholly dependent on fuel circulation, which makes predicting them extremely complicated.

The preliminary studies, which highlighted the useful nuclear characteristics of the fuel salt, will also have to be confirmed through additional analyses.

Concerning the issues relating to nuclear monitoring and power distribution uniformity, for an MSR, the link with power density cannot be assessed in the same way as for the other concepts, making any comparison difficult. Little information on this topic is available.

Lastly, not enough research into the SCWR concept has yet been conducted to enable any analysis from a nuclear physics perspective. A few observations may nevertheless be made:

- reactivity is affected by the complex route travelled by the circulating coolant, which passes through the core several times. Accurate studies will be required in order to ensure that reactivity is properly controlled;
- with the fast concept, the void effect may be strongly positive, due to:
    - the steel used for the cladding, resulting in lower capture counts when the spectrum is hardened;
    - the large core;
- materials able to withstand the ambient temperature conditions have yet to be developed, meaning that the corresponding nuclear data is probably subject to significant uncertainty;
- the behaviour of supercritical fluids in irradiating conditions is unknown.

8.3.3 *CONCLUSION*

Any comparative analysis of the “reactivity control” safety function across the various Generation IV concepts can only be very limited, as the nuclear behaviour of the cores is largely dependent on the objectives pursued in terms of use of the fissile material and available natural resources.

The concepts are at very different stages of maturity, and the nuclear characteristics of the various cores are very sensitive to the isotope compositions of the constituent material of the fuel (which are in turn dependent on the fuel cycle). Consequently, it would appear to be difficult to assess whether a particular concept is more or less able than another to implement effective design provisions to offset any intrinsic characteristics that are less favourable from a safety perspective. Regardless of the concept, it is important to note that from a nuclear physics perspective, a high-power core does not enhance safety.

In any case, a few elements can be assessed for the various concepts.

Concerning the nuclear physics of fast reactors, the main points to be taken into consideration are as follows:

- absence of neutron poisoning effect (xenon effect for water-cooled reactors) and a more limited reactivity swing over the operating cycle than with thermal-spectrum reactors;
- more limited sensitivity to the insertion of neutron absorbers compared with thermal-spectrum reactors, due to the smaller effective cross-sections of the absorbers;
- greater sensitivity to geometry changes than with thermal-spectrum reactors; in the event of core

degradation, the criticality risk is therefore greater;

- a relatively small proportion of delayed neutrons, compared with a thermal-spectrum reactor, due to the spectrum and the high proportion of plutonium;
- reactivity change measurement-based core state detection rendered difficult if reactivity effects are offset, particularly in large reactors.

More generally, reactivity control is a bigger safety factor for fast-spectrum reactors than for thermal-spectrum reactors.

Of the analysed concepts (SFR, LFR, GFR, MSFR), note that:

- For SFRs and LFRs, the potentially positive coolant expansion feedback coefficient is a design constraint.
- LFRs and GFRs are more subject to nuclear monitoring and power distribution uniformity than SFRs, due to their lower power density, resulting in larger cores for a given power output.
- LFRs are subject to a risk of water insertion into the core if steam generators are mounted inside the reactor vessel; additionally, inserting control rods into the core might be problematic owing to the high density of lead.
- The MSR appears to have very stable nuclear behaviour, even after significant reactivity insertion (of several dollars in a second), and is not subject to a risk of recriticality in the subcritical area. However, predicting nuclear parameters is extremely complex; the results yielded by current research will have to be corroborated by more in-depth studies.

8.4 SENSITIVITY TO LOSS OF COOLING

The Fukushima Daiichi accident highlights the safety importance of securing extended grace periods in total loss-of-cooling conditions, to enable operators to respond before the worst-case scenario (i.e. successive loss of barriers) occurs and establish adequate reserves (fuel oil, compressed air, water, etc.) to maintain the safety functions of the facility until off-site emergency response teams arrive, even if access routes are heavily degraded or when confronted with extreme climate conditions. In addition to the information presented in Chapters 2 to 7 regarding safety performance, and decay heat removal in particular, IRSN wishes to highlight a few intrinsic reactor characteristics that have a major impact on their sensitivity to loss-of-cooling accidents, in particular with regard to grace periods and autonomous operation of facilities.

The thermal inertia of the primary circuit and the ability to establish stable natural convection conditions in the core and the emergency cooling systems play a major role in reactor behaviour in the event of loss of cooling. These aspects are addressed in Sections 8.4.1 and 8.4.2. Depending on the reactor, the state of the facility may be degraded by an absence of reactor structure cooling. This issue is covered in Section 8.4.3. Lastly, IRSN examined the scope for on-site and then off-site emergency responses in the event that the pre-existing systems of the facility cannot be restored (Section 8.4.4).

8.4.1 *THERMAL INERTIA IN THE PRIMARY CIRCUIT*

As an initial approximation, in the event of reactor trip, the increase in coolant temperature over a given period depends on the ratio between the energy produced by the core and the thermal inertia of the complete primary circuit (i.e. the core, coolant, moderator and structures).

Table 5 shows the orders of magnitude of the predicted coolant temperature increases respectively one hour and ten hours after reactor trip (value annotated T_{adia}) due to loss of cooling, for five of the six examined concepts. For SCWRs, IRSN has no data with which to estimate the gradient of the coolant temperature increase but the HPLWR designers have indicated that, in the event of depressurisation due to loss of the feedwater pumps, water must be injected within 10 minutes, to prevent core degradation. If, however, the water inventory is maintained, the SCWR may be cooled autonomously, using semi-passive loops or condenser systems.

The values shown in Table 5 were determined assuming that the reactor temperature changes adiabatically in accordance with a point model. The table identifies the leading factor in thermal inertia (i.e. the coolant or the moderator). IRSN adopted the same Superphenix-type law for evaluating decay heat versus time for all the concepts, with the exception of the MSFR, for which a proprietary decay law was used; this assumption may be considered valid for an initial approach. IRSN verified that the order of magnitude of heating phenomena remained the same for reactors with different power outputs (verification for the Phenix and Superphenix reactors, for the SFR concept, and the BREST-300 and BREST-1200 projects for the LFR concept).

Table 5: Evaluation of coolant temperature increases 1 hour and 10 hours after reactor shutdown, assuming that the temperature changes adiabatically in accordance with a point model

	SFR	VHTR	GFR	LFR	MSFR
Thermal inertia of the primary circuit	High / due mainly to the mass of sodium: 3300t for 3000 MWth	High / due mainly to the high mass of graphite: 800t for 600 MWth	Very low (8t helium for a 150t)	High / due mainly to the mass of lead: 26,500t for 3000 MWth (BREST-1200 project)	Very low / low mass of salt (70t)
ΔT_{adia} at t=1h	53°C	30°C	1700°C	64°C	1600°C*
ΔT_{adia} at t= 10 h	308°C	200°C	-	375°C	-

(*): This value must be viewed in perspective, as thermal inertia is provided by an alternative means to the coolant (see below).

The ratio of decay heat to the thermal inertia of the core is more favourable for the VHTR than for the other concepts. This is because a VHTR core (see Chapter 3) demands a large mass of graphite, as this material is a less efficient moderator than hydrogen-containing materials. This accounts for the high thermal inertia of the core. Furthermore, the power density of the core is voluntarily limited to ensure that the maximum fuel temperature does not compromise its integrity in case of loss of cooling. Operators therefore benefit from very long grace periods.

The heating calculations for the LFR and SFR point to a period of at least several hours before reaching conditions that degrade the first or second barrier are reached (assuming initial coolant temperatures in a range between 400°C and 650°C).

On the other hand, it would be unwise to count on an extended grace period based on the thermal inertia of the gas used in the GFR; IRSN considers this to be a major drawback of this design, despite the fuel theoretically being able to withstand temperatures approaching 2000°C.

As already stated, the case of the MSR is very specific: due to the very low thermal inertia of the salt (ΔTadia = 1600°C after 1h) and the high operating temperature (approx. 700°C), the salt cannot be kept in the fuel system (see Chapter 6). Therefore, in the MSFR, the fuel salt must be rapidly drained in order to control the decay heat removal function. To be consistent with the other concepts for which reactor trip was included in the calculations, the thermal inertia of the MSFR should be assessed once the fuel salt has been drained to the subcritical area; it is the thermal inertia of the ensemble formed by the fuel salt located in the subcritical area and the water contained in the pool surrounding this area that should be taken into consideration. The pool is what provides the thermal inertia of the system. IRSN does not have any quantitative data relating to this aspect, however.

In terms of thermal inertia, the VHTR offers a major benefit compared with the other reactors. Reactors that use liquid metal as their coolant also have favourable properties and are therefore less sensitive than gas- or water-cooled reactors to loss of cooling, at least in the short term. It should, however, be noted, that a prolonged loss of means of cooling would result in structural damage. With the MSFR concept, the thermal inertia is not attributable to the properties of the coolant but to the pool in which the subcritical area sits. Potentially quite long grace periods may be available. The GFR and SCWR concepts, however, are unable to harness the thermal inertia of the primary circuit for the purpose of managing loss-of-cooling conditions.

8.4.2 *NATURAL CONVECTION-BASED DECAY HEAT REMOVAL*

Natural convection in the core and primary circuit can be harnessed to provide a passive means of removing decay heat. It maintains coolant circulation, which in most concepts is essential for core cooling. A few actions requiring modest energy inputs or manual actions may sometimes be necessary in order to prime the natural convection process.

The ability to establish natural convection-based circulation in the core depends on the physical characteristics of the coolant and the pressure drop in the core, as well as design provisions such as the relative altitudes of the top of the core and the power removal systems.

It would appear that natural convection-based cooling of the core and primary circuit may be considered for SFRs and LFRs, subject to appropriate design. The performance of such an operating mode will have to be verified through experimentation in the reactor, as natural convection must be established both in the primary circuit and in the decay heat removal systems.

For the GFR, decay heat removal requires forced convection to be maintained in the core. The gas flow through the core must be adapted to suit the ambient pressure in the primary circuit. The aim is to use natural convection-based decay heat removal after a certain period, subject to eliminating the risk of core bypass (see Chapter 4).

In the case of the SCWR, cooling may be achieved through natural convection, with the aid of the isolation condenser systems.

The MSR and VHTR are, once again, different to the other concepts studied by GIF, inasmuch as the fuel can be cooled with no need for the coolant to circulate. Coolant circulation is even considered to be undesirable[43].

For the VHTR, compliance with the maximum fuel particle temperature criterion (1600°C) is in principle achieved without the coolant circulating through the core, as decay heat is removed from the primary circuit essentially by conduction and radiation, and the reactor vessel and reactor cavity are themselves cooled by active or passive systems.

For the MSR, the fuel salt is drained in accident conditions. In view of the very low thermal inertia of the fuel salt, it does not appear feasible to remove decay heat entirely using natural convection while the salt is still present in the fuel system. Natural convection will nevertheless help to limit the rate at which the salt is heated, extending the available period in which to drain the salt. In the case of the MSFR, when the salt has been drained into the subcritical area, the pool that cools the drained fuel is cooled by natural convection (see Chapter 6).

Ultimately, decay heat can therefore indeed be removed by natural convection in these two concepts, even if the coolant does not circulate.

All the examined reactors can benefit from natural convection-based cooling, whether the natural convection is established in the primary circuit (as is the case of the SFR, LFR and SCWR) or in the systems used to cool the primary circuit from outside (in the case of the VHTR and MSR). However, with the GFR, forced convection is required in order to cool the core during the initial phases of an accident.

[43] In the VHTR, natural convection forms hot spots in the upper part of the primary circuit.

8.4.3 *REACTOR STRUCTURE COOLING*

In the event of loss of heat sinks or electric power, the grace periods before cooling must be initiated may be determined not by the temperature of the fuel or cladding, but by the rise in temperature of the reactor structures. This applies in particular to the SFR and LFR. Note that these concepts all operate at higher temperatures than existing reactors, and that the creep threshold of the metallic materials may be exceeded - in some cases rapidly - if the cooling systems fail to operate.

With the VHTR, reactor structures are cooled by systems located outside the reactor vessel (with the possible exception of the control rod mechanism cooling system). This also applies to the MSFR (systems located outside the subcritical area). This option makes for simpler cooling system design, monitoring and maintenance.

Externally-located reactor vessel cooling systems are also possible for the SFR, and indeed existed at the Phenix and Superphenix facilities (see Section 2.3.2). They may also be envisaged for the LFR.

8.4.4 *AUTONOMOUS SITE OPERATION AND OFF-SITE EMERGENCY RESPONSE*

Thermal inertia and support for natural convection are not the only factors to influence the sensitivity of the concepts to total loss of electric power and heat sinks; such factors include the ability to use diversified (water and air) heat sinks and mobile emergency resources (pumps, generators, etc.). Adequate resources (stockpiled water, gas, fuel oil, etc.) must also be available at the site, to enable the safety functions to be performed until off-site responders arrive, even if access routes are heavily degraded or when confronted with rare and severe climate conditions.

Providing a means of injecting coolant in the event of loss of reactor coolant is a key issue for several concepts, including the GFR, SFR and LFR. In this respect, GIF notes (in reference document 2) that it would not possible to inject water into these reactors to cool the core. Experience has shown that the ability to inject water can be very useful in severely degraded conditions.

8.4.5 *CONCLUSION*

The VHTR concept would appear to be the least sensitive to total loss of cooling.

The SFR and LFR also have high thermal inertia and the ability to remove decay heat by means of passive or semi-passive systems (requiring low-power energy sources), which should yield extended grace periods. However, in the event that these systems were to be unavailable or ineffective, there would be a risk of structural collapse. Note that efforts have been made at the design level to enable such conditions to be considered “practically eliminated”.

The MSFR also ultimately has good thermal inertia, subject to draining the fuel salt to the subcritical area.

The GFR has negligible thermal inertia and active emergency systems must take over from the normal systems within an as-yet undefined time limit. The effectiveness of cooling systems and their ability to operate autonomously also depend on a wide range of technical conditions (e.g. actuating isolation valves, maintaining pressure, etc.). IRSN considers that the grace period for this reactor is very short.

8.5 CONFINEMENT FUNCTION

As stated in Section 8.2.1, IRSN considers that an effort should be made to improve the radioactive material confinement provisions for Generation IV reactors over Generation III reactors. This applies to releases in all foreseeable forms, including atmospheric releases and liquid discharges.

Such an improvement may be obtained, in part, by better defining the boundaries of the barriers and related requirements, in particular in terms of leakage rates:

- The first barrier consists of the fuel element cladding in the SFR, GFR, LFR and SCWR, by the fuel particle coating in the VHTR, and by various structures in the MSR (fuel envelope in the case of the MSFR). Note that the robustness of the first barrier is a key factor for the VHTR safety demonstration.
- The second barrier comprises various structures and components, including the reactor vessel and systems carrying potentially radioactive fluids other than coolant. Note that for the SFR and LFR, the systems that control the cover gas above the coolant surface are considered to be part of the second barrier and are therefore a potential source of direct, immediate or delayed atmospheric releases.
- Little information is available regarding the boundary of the final barrier for the various studied systems.

SFRs and LFRs are subject to a risk of containment bypass via the gas cover space control system. This risk will have to be taken into consideration when designing Generation IV reactors. IRSN considers that an effort should also be made to enhance the leaktightness of the second barrier for these reactors.

The second barrier in the reactors that use helium as a heat transfer fluid (VHTR and GFR) are subject to a significant leakage rate in nominal operation. This results in uncontrolled diffuse releases in normal operation, in particular due to the presence of graphite particles in the VHTR, which does not seem acceptable for a Generation IV reactor.

Some of the systems selected by GIF use or generate products listed as hazardous substances in the course of their operation. Provisions must therefore be adopted, both at the design stage and during operation, to limit releases of such substances in both normal and accident operation, in a way consistent with the provisions adopted for radioactive material confinement (cf. Order mentioned in reference document 19). This issue primarily concerns three reactors: the SFR, LFR and MSR.

The modes whereby the confinement function may be lost in the event of a severe accident are described in Section 8.8.2.

8.6 SAFETY IN OPERATION

The intrinsic characteristics of a reactor have an impact on the ability to operate it safely. In this section, IRSN focuses on:

- real-time installation status monitoring capabilities (Section 8.6.1);
- operating constraints arising out of characteristics intrinsic to particular concepts (Section 8.6.2);
- installation control (Section 8.6.3);
- safety system architecture (Section 8.6.4).

In-service inspection of reactor structures and components is discussed in Section 8.7.

8.6.1 *REAL-TIME INSTALLATION STATUS MONITORING*

Operators must be able to precisely monitor the state of the installation, rapidly detect any abnormal developments and make a prompt diagnosis in the event of an incident or accident. This requires instrumentation that is suited to the physical phenomena potentially encountered and provides easily-interpretable measurements (whenever possible with direct measurement of phenomena, measurement sensitivity suited to requirements, minimal related uncertainties, fast response times, etc.). The complexity of the phenomena involved and reactor characteristics (in terms of coolant type, core, size, etc.) may complicate the task of installing instrumentation that delivers outputs easily interpreted by operators.

In particular, "real-time" characterisation of the state of the reactor entails global and local power measurements, the characteristic parameters representing the state of the fuel, coolant (core inlet and outlet temperature, etc.) and barriers (especially the first barrier).

Core state monitoring

Overall, as stated in Section 8.3, IRSN considers that large fast-spectrum cores are not conducive to uniform power distribution and make monitoring nuclear parameters more complex. Heterogeneous make this monitoring even more difficult. In both cases, the main challenge from a safety perspective relates to the detection and management of hot spots. The RNR-Na Project does not share this view, stating that operating experience feedback from the French SFRs did not reveal any specific difficulties deriving from the size effect.

With the SFR, the power of fuel assemblies is evaluated on the basis of temperature measurements, although it could also be based on nuclear measurements or flow rate measurements. In this respect, SFRs should be able to benefit from new techniques currently under development (neutron chambers in the reactor vessel and ultrasonic sodium flow rate measurements). Implementing equivalent measures in the LFR may be problematic, on the other hand, due to the high density and corrosive action of lead.

For the VHTR concept with a pebble-bed core, the fuel is randomly distributed in the core, which raises monitoring issues. In addition, the local helium flow rates in the core cannot be accurately calculated, due to bypasses in surrounding structures (see Chapter 3).

The uniformity of core temperatures is also a subject of concern with the MSR, which is subject to unmeasurable localised variations in the concentration of fissile material. Hydraulic dissymmetry problems can result in hot spots that can lead to structural damage.

In general, identifying the source of abnormal phenomena is made more difficult when nuclear, thermal-hydraulic, mechanical and/or other phenomena are coupled. This applies in particular to the MSR and SCWR concepts.

Monitoring of the first barrier

For the SFR, the monitoring solutions adopted for the first barrier, as already implemented on French reactors, are able to rapidly detect a degradation in the cooling function or cladding leaktightness (by detecting delayed neutrons emitted into the sodium - DND), thereby effectively protecting the core and keeping any sodium contamination to a low level. It may be possible to apply the approach adopted for the SFR to the GFR (using optical systems to monitor the fuel assembly outlet temperatures) and the LFR. However, for the GFR, it will be necessary to verify that helium (on account of its low density) is able to capture and convey delayed neutron emitters as effectively as sodium.

The complexity of the SCWR core would not appear to be conducive to fuel element cladding temperature monitoring. Monitoring the state of the installation, and the first barrier in particular, does not appear to be an easy task.

The difficulty with the VHTR relates to the ability to monitor the quality of the fuel particles before they are introduced into the core rather than to in-service monitoring.

8.6.2 *OPERATING CONSTRAINTS*

The risks specific to each concept are described in Chapters 2 to 7. Managing these risks requires the imposition of operating constraints of varying severity, some of which result in installation safety relying to a significant extent on operating procedures. This applies to the LFR in particular: the risk of structural corrosion requires the dissolved oxygen concentration to be maintained in a very narrow range, in order to prevent precipitation of lead oxide (which can potentially form clumps able to obstruct fuel assemblies) and protect structures throughout the primary circuit. The quantity of lead oxide produced and the thickness of the iron oxide layer depend on the temperature; managing changes in localised oxygen contents is likely to be difficult, due in particular to the heterogeneous temperature distribution in the primary circuit. As indicated in Chapter 5, managing such changes would even appear to be impossible in a large reactor.

As with the LFR, the corrosive nature of the coolant used in MSR and SCWR reactors creates some specific operational constraints, if only in terms of higher structural and component inspection and/or replacement frequencies. The operating procedures implemented to limit the corrosion product inventory in the reactor coolant will have to be particularly strict.

With some concepts, a high coolant freezing point also severely constrains operating temperatures:

- In the MSR, the salt has a crystallisation temperature of 560°C; due to the risk of damage to structures and components, the maximum temperature is limited to 700°C (subject to obtaining materials qualified for such temperatures). The operating temperature range is therefore relatively narrow.
- In the LFR, the high freezing point of lead (327°C) is a problem, in particular during reactor shutdown phases for fuel loading and unloading operations.

The operating constraints associated with managing the risk of coolant freezing in SFRs are less severe, as sodium freezes at a much lower temperature (98°C) and fuel handling operations do not require the reactor vessel to be opened.

The SFR's main operating constraints relate to the risks of reactions between the sodium and water, air and fuel (if the fuel is in oxide form). These risks are taken into consideration at the design stage; the operating constraints essentially concern the need to monitor the numerous primary and auxiliary fluid systems (carrying sodium, argon, nitrogen, etc.). In this respect, it would appear necessary in the light of the relevant experience feedback to enhance the reliability of risk detection provisions in order to avoid having an excessively high number of spurious detection signals, and to implement systems able to promptly detect all types of leak, including minor leaks occurring underneath thermal insulation.

Lastly, the nature of the coolant may lead to particular constraints in terms of product confinement and local working procedures. The chemical toxicity of the products used in the six examined concepts is addressed in Section 8.9.

8.6.3 *PLANT CONTROL*

A reactor installation is easier to control when a simple, direct relationship exists between a change in power and a change in a dominant, directly measurable parameter (core inlet temperature, reactivity, etc.). This is generally the case for reactors that are controlled by varying the control rod insertion level and the coolant inlet temperature (by controlling the secondary pumps or the steam flow rate for system monitoring purposes). This is the case of the SCWR, GFR, SFR and LFR. Furthermore, when these concepts are compared with the PWR, the absence of neutron absorbers in the coolant simplifies control and removes some risks associated with the use of liquid neutron poisons (corrosion, dilution risks, etc.). Additionally, fast-spectrum reactors benefit from the absence of xenon effects. IRSN expects controlling the aforementioned four reactors to be relatively easy. This view is supported by positive core control operating experience feedback from the French SFRs. It is subject to confirmation for the SCWR, however, due to the complexity of the core in this concept.

In principle, the VHTR is simpler to control than the aforementioned reactors, due to its "forgiving" nature, attributable in particular to core thermal inertia and good nuclear stability. The reactor can be controlled throughout its authorised power range just by adjusting the coolant flow rate, and reactor shutdown can be initiated by shutting down the blower units. The xenon effect must be managed, however.

Lastly, controlling the MSR in operation is theoretically simple, with the reactor automatically stabilising at the requested power level. Starting the first cores might prove to be complicated, however, and involve some risks (relating to the approach to criticality and the removal of power generated by the fuel salt during the filling phase).

8.6.4 *SAFETY SYSTEM ARCHITECTURE*

A goal of simplifying the safety systems is being pursued for several Generation IV concepts. This applies to the MSR (automatic salt draining and no control rods) and in particular to the VHTR, which essentially has a safety cooling system outside the reactor vessel (but no cooling system connected to the primary circuit and no emergency blower system).

A high level of reliability must be ensured for the decay heat removal systems in the SFR and LFR, due to the risk of structural collapse. This can make the safety system architecture complicated. The RNR-Na Project has indicated that the SFR decay heat removal systems are designed to be as simple as possible. The risk of a sodium-water reaction in the SFR requires an intermediate system to remove power during normal operation, further complexifying the overall architecture.

The GFR, in its most recent state of development, features particularly complex decay heat removal systems.

The SCWR may use safety systems similar to those adopted for BWRs, featuring turbopumps and passive steam condensing systems; the intrinsic characteristics of this reactor do not appear to offer any particular benefits from a safety system architecture perspective.

Concerning shutdown systems, the Generation IV concepts have retained the current principles, supplementing them with passive systems triggered by temperature (SFR, VHTR) or changes in reactor coolant flow rate (LFR). Only the MSFR concept does not feature a neutron absorber-based shutdown system.

8.7 IN-SERVICE INSPECTION

IRSN considers the ability to perform in-service inspections of structures and components to be an important safety requirement, compliance with which is a prerequisite for reactors earmarked for construction by 2050. This requirement is all the more important as the coolant used in some reactors (i.e. the LFR, MSR and SCWR) is particularly corrosive and the operating temperature conditions are particularly severe for structures and mechanical components, due to the ambitious plant efficiency objectives (resulting in temperatures approaching the material creep temperatures and large temperature differentials over the conversion cycle).

In-service inspection is difficult in reactors that are not designed to allow the reactor coolant to be drained (other than in exceptional conditions) and operate with “optically opaque” fluids; this applies to the sodium in SFRs and even more so to the lead in LFRs, for which inspection techniques will largely be based on ultrasonic measurements. In such reactors, however, it is essential to be able to inspect the core support structures and the second barrier, which is relatively complex. Regarding this aspect, the RNR-Na Project stressed that the safety vessel can be inspected without difficulty, and that the sodium-filled parts of the main vessel are not complex and can all be inspected from outside. For IRSN, the difficulty relates in particular to the ability to inspect the core support structures and the reactor top head, on the primary circuit side.

In addition to the difficulties described for the SFR, the LFR is subject to major drawbacks relating to the activation of lead, its high density (similar to that of the steel in the vessel) and the presence of an oxide layer on metal surfaces that degrades coolant wetting of the walls, decreasing ultrasonic transmission quality. In addition, the lead must be maintained at a temperature above 327°C to prevent any risk of freezing.

In-service inspection for the SCWR, VHTR and GFR (with transparent helium) should not pose any particular problems, subject to making allowance for the need for inspections at the design stage.

The MSR concept has not been defined in sufficient detail to enable IRSN to adopt a position regarding in-service inspection. For the MSFR, it must be possible to inspect the fuel system after having drained the fuel salt during the reactor shutdown states. The subcritical area can be inspected when salt is stored in the storage area. The means of introducing the necessary monitoring equipment remain to be defined. The chosen measurement techniques will also need to make allowance for any salt residues or films that may remain on equipment surfaces after the fuel salt has been drained . During extended shutdowns, it will be possible to remove and flush the heat exchangers located between the fuel system and the intermediate system. In-service inspection of MSR structures and components would appear to be a complex task that might represent a sticking point for this concept.

In conclusion, the in-service inspectability issue would appear to be a weakness of the LFR and potentially the MSR, as well as, to a lesser extent, the SFR. Note that it is also important to be able to repair defective components and structures during the life of the plant; this requirement must also be taken into consideration at the design stage.

8.8 SEVERE ACCIDENT BEHAVIOUR AND RELEASE ROUTES

8.8.1 *INTRODUCTION*

The severe accident concept generally taken into consideration for PWRs is not applicable to the reactors examined in the context of this report. The term “severe accident” was therefore redefined such that it may be applied to all the studied reactors (Section 8.8.1.1). After defining this key term, IRSN examined what approach could be adopted to attempt to place the severe accident behaviour of these reactors into perspective (Section 8.8.1.2).

8.8.1.1 Definition of a “severe accident” applicable to all reactor types

The IRSN publication in reference document 15 offers the following definition for the term “severe accident”: *“The term ‘severe accident’ or ‘core melt accident’ refers, in a pressurized water reactor, to an accident during which reactor fuel is significantly degraded with some degree of reactor core melt.”*

Since the Fukushima accident, there is an acceptance that the possibility of a severe accident should not be limited to reactor cores, but should also include the pools used for spent fuel storage. This expanded definition, including fuel storage facilities, is well suited to light water reactors, but poses a problem when one attempts to apply it to other reactor concepts, such as:

- VHTRs: on the basis of current studies, fuel melt is considered highly improbable, or even impossible, in view of the design of the TRISO fuel particles (see Chapter 3).

- MSRs, in which the normal core state is liquid; the concept of fuel degradation is therefore difficult to define. The concept of severe accident must therefore be expanded.

It should also be noted that, with SFRs or LFRs, structural collapse (of the reactor vessel and core support structures, etc.) may in some conditions, and in particular in the event of prolonged loss of cooling, occur before core melt (with structures being damaged by creep as a result of the temperatures reached). In this context, a severe accident is defined as an accident involving core melt or structural collapse; the designers intend to "practically eliminate" the possibility of structural collapse.

IAEA and NRC have suggested a broader definition than that generally used for LWRs.

IAEA Severe Accident Definition (reference 16): "*A beyond design basis accident comprises accident conditions more severe than a design basis accident, and may or may not involve core degradation. Accident conditions more severe than a design basis accident and involving significant core degradation are termed severe accidents.*"

NRC Severe Accident Definition (reference 17):

1. ***"Beyond design basis accident**: this term is used as a technical way to discuss accident sequences that are possible but were not fully considered in the design process because **they were judged to be too unlikely**.*
2. ***Severe accident**: a type of accident that may **challenge safety systems** at a level much higher than expected."*

These two more general definitions can be applied to VHTRs and MSRs. However, for next-generation reactors, it is acknowledged that accidents involving "fuel" degradation must be included at the design stage, rather than being considered "beyond design-basis" events.

Another definition is given, indirectly, in the international nuclear event scale (INES). A severe accident corresponds to levels 6 and 7 on this scale. The TMI2 accident involving partial core melt was rated as level 5, and as such was not a severe accident according to this scale.

For VHTRs, based on the current state of research, the design considers the risk of fuel melt as a "residual risk"; however, large quantities of fission products may be released into the reactor vessel if high temperatures are reached, at least locally, for extended periods (a few hours at 1800°C rather than 1600°C). A severe accident might therefore result from exceeding a certain fuel particle "coating" failure threshold.

MSRs are harder to position, not least because the fuel, in some models, is liquid in normal operation. However, by analogy with the VHTR, a severe accident might correspond to a failure of the first barrier.

In light of the foregoing considerations, IRSN has adopted the following definition for the purpose of this report: **A severe accident in a nuclear reactor is an accident during which the nuclear fuel radioelement confinement function is significantly degraded, regardless whether the fuel is inside the reactor, being handled or in a storage area.**

This definition does not cover the major chemical accidents liable to occur in some reactors. However, a severe nuclear accident may include radiological and chemical releases. This definition applies to this report; it is not intended for use as a general reference definition.

Note that:

- This definition intentionally makes no reference to the state of the core: there is no need for a specific definition for MSRs, which use fuel salt.
- Severe accidents do not necessarily result in severe core degradation, the distinctive criterion being the potential releases. A VHTR is always subject to a certain degree of radiological contamination of the primary circuit, even in normal operation, and a primary circuit leak may result in significant releases if it occurs in parallel with high-energy phenomena liable to affect the other barriers.
- This definition would cover collapse of the structures forming the second barrier, which might occur before generalised core melt in some reactors such as the SFR or LFR.

The notion of "significantly degraded" will have to be defined for each reactor when the relevant studies have yielded precise criteria. For example, for the EPR, a severe accident is considered to be initiated by a core outlet temperature in excess of 650°C or by a high dose rate measurement inside the containment.

The RNR-Na Project considers that the definition of a severe accident should "*include the notion of a radical change in the physical phenomena involved in a severe accident, compared with those involved in other accidents addressed in the design.*" IRSN acknowledges that such a radical change exists with some reactors (such as PWRs) but argues that the proposed notion does not apply to all reactors.

8.8.1.2 Approach adopted by IRSN in determining the Institute's position regarding the risk of severe accident for the studied systems

At the current stage of development of the various systems, it does not appear possible to compare comprehensive deterministic scenarios initiated by a particular event and developing into a "severe accident" as a result of safeguard system failures and/or human error. Such a comparison is only possible for sufficiently well-developed concepts (which is not the case of the majority of the reactors studied by GIF, with the exception of the VHTR and SFR) for which control procedures and the related instrumentation have been at least partially defined. Extensive research has been conducted for the SFR and VHTR, as a result of which the accident risks associated with them are better understood than those associated with the other concepts. This situation is conducive to errors of perspective and is a potential source of confusion.

The only reasonable attitude in terms of adopting a position regarding the severe accident risk (without seeking to identify the specific accident scenarios potentially resulting in such conditions) appears at this stage to be as follows:

- Consider very general possible "accident configurations" liable to result in releases of radioactive and possibly toxic elements.
- Attempt to assess the risks associated with such configurations, based on available expert opinion and a few very general principles.

Note that the configurations liable to lead to significant releases will require measures to prevent or mitigate their consequences, which at least in theory almost always exist (but may be very costly). Such provisions are generally not defined for non-finalised reactor concepts.

Initially, IRSN sought to identify the various plausible modes of loss of the confinement function based on the classification adopted for light water reactors (α, ß, γ modes, etc.) in the light of the Rasmussen study. The possible release routes yielded by this approach are described in Section 8.8.2.

IRSN then provides an overview of the foreseeable severe accident configurations for the various reactors, based on the information already presented in Chapters 2 to 7 of this report (Section 8.8.3).

8.8.2 *RELEASE ROUTES*

This chapter focuses primarily on the possible radiological material release routes for a reactor in operation. Naturally, accident conditions relating to shutdown states, fuel handling and transfer operations as well as on-site fuel storage prior to core loading or after unloading will also have to be considered when developing a new reactor process. These conditions are not addressed in this chapter, owing to the lack of available information.

The European ASAMPSA2 project (reference 18) identified the possible release routes for four reactors (VHTR, SFR, LFR and GFR) by transposing the α, ß, γ, δ and ε confinement function loss modes, as defined for light water reactors in the WASH-1400 report. The same approach has been adopted here. The α, ß, γ, δ and ε modes are defined as follows:

- α mode corresponds to a steam explosion in the reactor vessel or reactor cavity, rapidly leading to containment failure.
- ß mode corresponds to containment bypasses:
    - either via systems at the interface between the interior and the exterior of the containment, such as the steam generators;
    - or via containment penetrations.
- γ mode corresponds to combustion-related phenomena inside the containment liable to rapidly lead to its loss (typically, a hydrogen explosion in light water reactors).
- δ mode corresponds to a loss of the confinement function following a gradual pressure increase inside the containment.
- ε mode corresponds to a loss of containment due to basemat melt-through.

Table 6 shows the proposals issued by the ASAMPSA2 project, with a few modifications. These proposals are supplemented for the two concepts not examined by the project.

Reactivity accidents in fast-spectrum reactors have been assigned to the α mode category.

ß mode mainly covers heat exchanger ruptures (in concepts featuring indirect energy conversion cycles) and steam line breaks (in concepts featuring direct energy conversion cycles), combined with a containment isolation failure. For reactors with an intermediate circuit, the risks depend on the position of the secondary heat exchanger (i.e. inside the containment or outside the containment).

The interaction between molten core and concrete would not necessarily melt through the basemat (ε mode) but would generate combustible gases, including, at the least, hydrogen formed as rebars are oxidised by steam released as the high temperature causes the concrete to decompose. Subsequent inflammation or detonation of such gases might damage the containment (γ and ð modes).

The reference SCWR project (HPLWR) cannot easily be included in this framework, due to the existence of a fourth barrier, the leaktightness requirements for which are undocumented. When producing the table it was assumed that loss of the third barrier (i.e. the cylindrical prestressed concrete containment) would result in environmental releases.

Note that a systematic study of reactor release routes may be undertaken at the design stage, yielding a robust design at all levels of the defence-in-depth approach. Put differently, provisions intended to mitigate the consequences of the most severe accidents will be more effective if defined at the start of the design process, rather than being added at the end.

Table 6: Containment loss modes for each reactor concept

	SFR	GFR	LFR	VHTR	MSR	SCWR
α mode (explosion)	- Release of mechanical energy during a reactivity insertion accident - Sodium-fuel interaction	Release of mechanical energy during a reactivity insertion accident	- Steam explosion due to rupture of one or more steam generator tube - Release of mechanical energy during a reactivity insertion accident	Graphite dust explosion	- Possible thermodynamic interactions requiring study - Any release of energy during a reactivity insertion accident would be very limited	Steam explosion
B mode (bypass)	Containment bypassed via cover gas space pressure control system (argon system)	Heat exchanger rupture and failure to close containment isolation valves	- Steam generator tube rupture and failure to close containment isolation valves - Containment probably also bypassed via the argon system as in the SFR (although no data relating to this system is available)	Heat exchanger rupture and failure to close containment isolation valves[44]	Low risk, as the intermediate heat exchanger is inside the containment	Steam line break and failure to close containment isolation valves

[44] According to the RNR-Na Project, the bypass risk in the VHTR differs from that in the other concepts inasmuch as the safety strategy consists in venting reactor coolant helium to the exterior of the building in order to remove a potential source of pressure that might otherwise diffuse radioactive products over the long term when the increase in fuel temperature risks compromising its leaktightness. IRSN considers that the bypass risk should be limited to the fullest possible extent.

	SFR	GFR	LFR	VHTR	MSR	SCWR
γ mode (combustion)	- Reactor coolant sodium fire - Combustion of gases formed by molten core - concrete interaction (MCCI)	- Combustion of H2 / CO produced in case of steam ingress into core with silicon carbide cladding - Combustion of gases formed by MCCI	Combustion of H2 / CO formed by MCCI	- Combustion of H2 / CO produced in case of steam ingress into core with a graphite moderator - Explosion of dust (may also be α mode)	Combustion of H2 and CO formed by MCCI	Combustion of H2 and CO formed by MCCI
δ mode (slow pressurisation)	- Sodium vaporisation (following loss of ultimate heat sink) - Gradual heating associated with release of fission products into containment	Containment pressurisation by gases formed by CCI	Containment pressurisation by gases formed by CCI	- Graphite fire - Gradual heating associated with release of fission products into containment	Containment pressurisation by gases formed by MCCI	- Containment pressurisation by gases formed by MCCI - Reactor vessel depressurisation
ε mode (basemat melt-through)	MCCI	MCCI	MCCI	Very probably irrelevant	MCCI	MCCI

8.8.3 *SEVERE ACCIDENT PHENOMENOLOGY*

Risk of recriticality in severe accident conditions

The phenomenology of reactivity accidents in fast-spectrum reactors is complex, as a reactivity insertion accident does not necessarily end with the dispersion of fuel triggered by the power surge, if the dispersion is not sufficiently intense to prevent potentially recritical agglomerations of any remaining fuel. The relatively high enrichment rate of the fuel enables materials to be redistributed in all critical configurations, making further power excursions a possibility. These are referred to as “secondary excursions”, in contrast to the “primary excursion”, which is the initial reactivity excursion. This secondary phase poses special challenges to fast-spectrum reactor designers, in particular because it is very difficult to reliably predict the geometric configuration of the fuel after the primary excursion.

SFR designers are currently working to “eliminate” the risk of secondary excursions by designing routes to channel molten fuel out of the core to a core catcher designed to prevent any return to criticality. The effectiveness of such provisions remains to be demonstrated, however.

The GFR designers are not yet researching secondary excursions, but there does not appear to be any reason to rule them out.

Regarding the LFR, the designers consider that molten MOX fuel is certain to be floatable as it is lighter than the reactor coolant (lead). This remains to be demonstrated. The fuel may only be partially floatable if the corium phases separate. Even if the fuel was demonstrated to be floatable, it is not unreasonable to imagine that molten fuel could accumulate at the surface of the liquid lead in a reactive geometric configuration;

For the MSFR, the risk of secondary excursions is at this stage considered to be “eliminated”, as it is already the case of the primary excursion.

Fuel cooling in severe accident conditions

This section focuses on cooling of severely degraded cores. The studied configurations correspond to a molten, relocated core, for the concepts for which such conditions are foreseeable. They may occur, for example, after a reactivity accident, after plugging and subsequent degeneration, or in the event that all decay heat removal systems fail to operate, resulting in the failure of the reactor vessel.

Concerning the VHTR, the core geometry may be unaltered but in a scenario involving excessive heating of the fuel particles, fission products would be released and potentially transferred into the containment. Note, however, that the safety demonstration for these reactors should show that such sequences fall under the residual risk.

For the MSFR, most of the severe accident configurations currently considered relate to scenarios involving a failure of the first barrier (in either the critical area or the subcritical area). The first scenario must be studied with due consideration for the possible reactor configurations, in order to assess the consequences (fuel salt flowing into the subcritical area or into the subcritical area cooling pool, risk of return to criticality due to the moderating effect, etc.) The risk of penetration of the subcritical area is addressed in the design:

- Burnable poisons stored on the pool floor are diluted to prevent a return to criticality.
- Molten fuel spread over the pool floor should be cooled by the pool water.

In the event of prolonged failure of the decay heat removal function, only the VHTR may have a relatively favourable long-term behaviour, with the fuel envelope remaining intact. With the MSFR, the second barrier should remain intact, significantly limiting environmental releases.

For the remaining four concepts, fuel, in the form of corium, would be relocated or dispersed outside the active area of the core. Depending on the reactor, the reactor vessel may be fail if no means of decay heat removal is restored.

The SCWR should behave similarly to current LWRs in the event of core melt resulting in (rapid or delayed) vessel failure, unless, perhaps, it is cooled from the exterior (as in the AP1000 or the Finnish reactor at Loviisa).

The GFR concept is particularly unfavourable regarding the ability of the vessel to withstand core melt: in the event of a delay in restoring decay heat removal, it would probably not be possible to save the core if a certain (undoubtedly relatively small) “threshold” mass of corium is relocated.

Lastly, for the SFR and LFR, there would be a relatively long grace period before vessel melt-through ; it should be possible to avoid a vessel melt-through, even in the event of a delay in restoring decay heat removal and the relocation of significant masses of corium. Further research is required, including for the SFR, to confirm this point.

For each concept concerned by the risk of vessel rupture as a result of core degradation, an appropriate solution must be implemented to mitigate the consequences of an accident (such as the core catcher on the EPR, for example).

Risks relating to high-energy interactions

Apart from the risk of molten core-concrete interaction, which exists with all the concepts, risks of high-energy interactions in severe accident conditions may exist as a result of:

- thermodynamic interactions between coolant and molten corium, or between coolant and a “fluid” entering the primary circuit;
- chemical interactions.

The two concepts known to be subject to a risk of thermal-hydraulic interaction between the reactor coolant and corium are the SFR and the SCWR. Further research and development work is necessary, however, in order to explore the subject in greater depth. Note that, for the SFR, the energy released during such an interaction is taken into consideration when determining the containment design basis. For the other four concepts, subject to confirmation by targeted R&D, the risk would appear to be relatively limited.

Chemical reactions may:

- be strongly exothermic, like the zirconium oxidation reaction in light water reactors. This raises the issue of removing the surplus heat;
- produce gases, some of which may be flammable or explosive; gases may damage the containment either by explosion or by gradual pressurisation.

Two different situations must be considered, depending whether the gases are produced:

- in the reactor core (vessel), following ingress of a liquid or gaseous fluid into the primary circuit;
- in the containment, following a loss of reactor coolant or vessel rupture (resulting in molten core-concrete interaction). MCCI produces:
    - hydrogen, carbon monoxide and dioxide, when water vapour released by the concrete degradation process oxidises the metallic phases in the corium;
    - hydrogen and carbon dioxide, as water vapour released by the concrete degradation process oxidises the rebars. The latter point is important inasmuch as this reaction generates a large quantity of hydrogen, even if the corium does not react with water.

The main sources of the identified risks of high-energy chemical interactions are as follows:

- in the SFR, contact between sodium and water or air; this risk mainly concerns the steam generators, or the reactor building if sodium is ejected onto the slab;
- in the VHTR, ingress of water or air into the primary circuit, creating a risk of an oxidation reaction with graphite. The risk of a graphite fire in such conditions cannot be ruled out. However, only a fairly limited portion of the core is likely to be heavily degraded in almost all foreseeable accidents. Nevertheless, even if the degradation is only localised, it may still have significant consequences (including core collapse, for example, if the degradation is located at the base of the prismatic assemblies) depending on where it occurs;
- in the SCWR, hydrogen may be produced by the interaction between superheated steam and steel structures. Significant hydrogen production should be anticipated in accident conditions before the SCWR vessel fails .

For the other concepts, it should be possible to consider the risk of high-energy interaction as improbable, subject to implementing appropriate design provisions (for example, in response to the risk of air or water ingress into the primary circuit in the GFR, or the risk of a lead oxidation reaction upon contact with water or air). The problems associated with salt-related chemical interactions are poorly understood, making it difficult to issue an opinion regarding the energy levels of water-fuel salt reactions.

Impact of coolant type on the radiological source term

The reactor radiological inventory depends on the concept and the core history. No comparisons can therefore be drawn at this stage, other than regarding the toxicity of the products involved. However, when “comparable” inventories are assumed, some coolants have an impact in limiting the radioactive releases.

There are three scenarios in this respect:

- The coolant is an inert gas (helium). In the VHTR and GFR, there are no possible chemical interactions between the coolant and the various fission products, and hence no benefit to be obtained in terms of trapping fission products. In the VHTR, in view of the very large number of particles present in the core

(several billion), some particles are acknowledged to be "defective" (i.e. subject to a partial loss of integrity); this situation is tolerated. Consequently, the primary circuits of VHTRs will always be subject to a certain level of contamination by fission products, even though a system to continuously purify the helium is planned for such reactors. These fission products will be major contributors to the radiological source term.

- The coolant is supercritical water (SCWR concept): conditions will be similar to those in the light water reactors currently in services.
- The coolant readily combines with a wide range of fission products. The formation of chemical compounds has a positive impact in terms of fission product retention: this applies to the remaining three concepts, which respectively use sodium, lead (or lead-bismuth eutectic) and molten salts.

8.9 TOXICITY OF CHEMICALS

The chemical substances – specific to each process – that may be present at a facility are described in Chapters 2 to 7.

It should, however, be noted that the chemical inventory for each process cannot be considered exhaustive, based on current knowledge; it will need to be supplemented in the light of future understanding and as operating experience feedback involving the same substances is acquired by experimental facilities and laboratory research.

Furthermore, in the absence of information regarding the quantities of each such chemical or element present in each process, risks cannot be evaluated and it is therefore not possible to produce "toxicity" assessments for the various concepts.

However, the following lessons can still be learned, based on the available information:

- Concerning the SFR, sodium oxides produced when sodium burns in air are transformed - due to the presence of moisture and carbon dioxide in the air and – primarily into sodium hydroxide (NaOH) and sodium carbonate (Na_2CO_3), which are toxic to health. No toxicological reference values exist for either sodium hydroxide or sodium carbonate. Acute toxicity threshold values have been defined, however, in particular for the purpose of determining radiuses for the Creys-Malville off-site emergency plan. For example, the threshold value adopted by EDF in 2010 when determining the off-site emergency planning zone (corresponding to the threshold for irreversible effects) was 5 mg/m^3 for a 60-minute exposure, whereas a value of 250 mg/m^3 had previously been used. INERIS recommends (reference 20) using the same toxicity value (5 mg/m^3) in an emergency situation for the purpose of defining measures to protect the population in the event of an accident involving releases consisting of a mixture of sodium hydroxide and sodium carbonate aerosols. Concerning the rise of toxic release, the confinement strategy adopted for the SFR will have to be improved with respect to past practices.
- The LFR uses lead, a highly toxic substance. Conventional industry is seeking to systematically eliminate lead from industrial processes. Lead or LBE may also contain impurities (such as silver, cadmium, copper, chromium, indium, tin or tungsten), some of which are also highly toxic. In addition, LBE is strongly activated, generating ^{210}Po, known for its very high toxicity. It should be noted, however, that the requirements applicable to the reactor coolant purification function are necessarily particularly stringent, considering the potential consequences of impurities in the coolant.

- The MSFR uses molten fluoride salts (lithium fluorides). The purpose of the related fuel reprocessing plant is to recover uranium and separate minor actinides from fission products. The extraction and separation processes are based on acid-base, redox and complexing and decomplexing reactions. To this end, chemicals are introduced into the process, including fluorine (F_2) and hydrogen (H_2), which are used to extract and recover 99% of the uranium. These reactions produce a number of substances, including particularly toxic hydrofluoric acid (HF), which is produced when lanthanides are oxidised. Hydrofluoric acid and fluorine (F_2) are toxic by inhalation and by ingestion. It will not be possible to ascertain the chemical toxicity of the reactor coolant until the actual composition of the salt in terms of the fluorinated molecules formed is known. Furthermore, it should be noted that molten salts create unusual physicochemical conditions, in particular in terms of redox potential. Thus, reactor coolant toxicity is due not only to its composition but also to its chemical reactivity, potentially facilitating the formation of other highly toxic substances.

In conclusion, LFR and MSR operation would appear to use or generate particularly toxic radioactive and chemical substances. Although the quantities of hazardous substances produced are not known (as they will depend on future design choices relating to purification and effluent management processes, for example), this aspect is a major disadvantage of these concepts; particularly efficient purification or filtration systems will be required. This issue also concerns the SFR, albeit to a lesser extent. Note that the acute toxicity threshold value ultimately adopted for the purpose of assessing the consequences of chemical releases will be a decisive factor for the design of sodium-related accident prevention and mitigation provisions.

8.10 HAZARD SENSITIVITY – SEISMIC HAZARDS

This section refers to a rare and severe earthquake, significantly stronger than the design-basis earthquake.

Effects on reactivity

In broad terms, a powerful earthquake could have very significant consequences in terms of reactor safety if it were to distort the core, preventing control rod drop; in response to this risk, articulated control rods were developed for the SFRs operated in France. Furthermore, with SFRs, LFRs or GFRs, an earthquake might cause core power output to increase by moving fuel assemblies closer together during horizontal tremors (compaction phenomenon), thereby inserting reactivity.

In a VHTR, a powerful earthquake would theoretically have little effect on the reactivity of a core that uses blocks. This type of core is not very compactable due to the solid nature of the moderator and the prism stacking. The ability of the graphite blocks to withstand an earthquake, particularly if they are irradiated, should nevertheless be verified. For a pebble-bed core, on the other hand, the effects of compaction will have to be analysed, to demonstrate that they would be offset by neutron feedback. Furthermore, a gravity absorber-ball injection system, whose effectiveness would be little altered by core distortion, is planned in some VHTR projects.

Potential effects on the strength of structures and systems

In the SFR, multiple sodium leaks from intermediate circuit loops and sodium fires representing hazards to important equipment and premises may potentially occur as a result of a very severe earthquake, significantly more powerful than the design-basis. The strategy with regard to such a situation could be to rely on draining the damaged systems, subject to availability of sodium dump tanks and one or more robust means of decay heat removal (not connected to the intermediate sodium loops). The core support structures may also be subjected to heavy loading, given the mass of sodium in the reactor vessel in the integrated concepts.

In the case of the GFR, given that the refractory core and internals are lighter than in the SFR, the inertial forces transmitted to the core support structures would be weaker, which is favourable from a safety perspective. However, it is essential for the GFR reactor to trip without delay, in view of the risk of loss of primary circuit leaktightness in the event of an earthquake. Also, in the event of a large break in the primary circuit, forced convection in the core would initially be indispensable to remove decay heat, hence the need for an electric power source and means of injection designed to cope with a severe earthquake.

For the LFR, the considerable mass of lead in the reactor vessel could impose large loads on some structures, especially given that an earthquake could cause lead displacements with wave effects. IRSN notes that LFR earthquake behaviour is a sticking point for the safety demonstration, and considers that the development of the concept will be limited to small reactors as a result.

In terms of earthquake resistance, the behaviour of the thermal-spectrum SCWR is basically the same as that of existing BWRs.

Lastly, IRSN is unaware of any studies relating to the seismic resistance of MSRs and it is therefore difficult to take a position on this issue at this stage. However, the small mass of the combined fuel and coolant compared with SFRs, LFRs and HTRs should be a favourable aspect, from an earthquake resistance perspective. Conversely, the coupling between the power generating unit and the salt processing unit may be unfavourable.

In conclusion, it would appear that the LFR is the concept most sensitive to seismic activity, by far. The (pebble-bed) VHTR and SFR concepts are also, to some extent, sensitive to this hazard.

8.11 MATURITY OF CONCEPT AND TECHNOLOGICAL CHALLENGES

The shortlisted concepts are at very different stages of maturity. The SFR is the concept for which the most extensive experience feedback is available, representing approximately 400 reactor-years of operation at around 20 reactors worldwide, some of which are still in service. This data spans a range of power ratings. A 500 MWe and a 800 MWe reactor are in the final phase of construction in India and Russia, respectively.

The VHTR concept benefits from experience feedback gained from the HTR reactor (with the coolant temperature limited to 850°C), seven of which have been built since the mid-1960s. This experience feedback amounts to approximately 60 reactor-years of operation. Two 200 MWth modules are currently under construction in China.

Little or no experience feedback is available for the remaining concepts:

- No prototype gas-cooled fast reactors have ever been built, although numerous studies have been carried out.
- No civilian lead- or LBE-cooled reactors have been built to date; LBE-cooled reactors have been built in the Soviet Union, for use as military submarine propulsion units.

- Only very limited experience feedback is available for the MSR (two uranium-fuelled experimental thermal-spectrum reactors); the MSRE project has nevertheless yielded valuable lessons relating to controlling corrosion of structural materials and reactor behaviour during transients.
- No SCWR projects have been built. Note, however, that the thermal-spectrum SCWR concept is an evolutionary design based on BWR reactors, and as such benefits from much of the experience feedback relating to the BWR, in particular regarding safety system design. This concept also benefits to some extent from the decades of experience feedback gained from supercritical water-cooled fossil-fuel power plants.

Regarding the GFR and LFR, the technological challenges must be overcome before an industrial project can be envisaged. Due to the very ambitious specifications of the GFR, in particular in terms of the desired operating temperatures, a number of significant technological problems remain to be resolved. For the LFR, controlling corrosion and erosion is the principal problem. IRSN considers the construction of a low-power demonstrator to be an essential step in the development of both concepts.

The feasibility of the MSR concept remains to be proved. The development roadmap for this type of reactor should include a detailed development plan featuring studies and tests to validate various technological options and assess their feasibility. In this context, selecting and validating a material suitable for high-temperature operation in a saline environment containing fission products, and designing appropriate intermediate heat exchangers will be key factors in demonstrating the viability of the concept. IRSN considers that the development of this process will require the construction of a low-power demonstrator, which is not a realistic short-term prospect.

Lastly, the SCWR concept is at a very early stage of development. IRSN mainly examined a thermal-neutron SCWR concept using MOX fuel; the Institute is not in a position to issue an opinion regarding the feasibility of a fast SCWR. For the thermal-neutron SCWR concept, extensive design and safety studies will be necessary in order to ensure that the concept is viable, before proposing a demonstrator in the medium-term.

Ultimately, the principal technological difficulties identified to date – compromising the prospect of a transition to industrial-scale projects in the first half of the 21st century – relate mainly to:

- the high temperatures anticipated in some reactors (VHTR, SCWR, MSR and in the GFR in accident conditions);
- the design of the core and fuel assemblies: this applies to the GFR (requiring the development of a refractory fuel able to withstand high temperatures), the SCWR (cladding material resistant to supercritical water; core design) and the VHTR (fuel particles designed to remain leaktight at temperatures exceeding 1600°C);
- managing the corrosive effects of coolant on structures; this applies in particular to the LFR, SCWR and MSR concepts;
- performance of safety functions:
    - performance of the decay heat removal function in the GFR (requiring pressure to be maintained);
    - performance of the MSFR emergency fuel salt draining function, from the critical area to the subcritical area.

8.12 OVERVIEW OF ISSUES REQUIRING ADDITIONAL RESEARCH FOR THE PURPOSE OF THE SAFETY DEMONSTRATION

Table 7 shows the main issues requiring further study before the safety demonstration can be undertaken. Note that the list of issues depends to a large extent on the level of understanding of each concept. In particular, the issues listed for the SFR, which result from a detailed safety analysis of existing projects and facilities may largely be transferrable to the other concepts.

As a basic principle, the safety demonstration will have to be based on experimentally-qualified tools that are sufficiently representative of the modelled installations and physical phenomena.

A satisfactory safety demonstration may be more difficult for concepts in which the nuclear aspects are strongly coupled with thermal-hydraulic (in the SCWR) or thermochemical aspects (in the MSR), given that a more robust safety demonstration will be required for future reactors.

Table 7: Issues requiring additional research for the safety demonstration

Concepts	Issues requiring additional research for the safety demonstration
SFR	- Control and nuclear monitoring of localised reactivity insertion accidents - Research into fuel assembly blockage accident risks - Establishing natural convection in the core and decay heat removal systems - In-service inspection - Molten fuel behaviour (relocation, debris bed formation, interaction with molten steel and sodium, fuel dispersion) and effectiveness of mitigation provisions (in particular, core catcher design to ensure subcriticality and cooling) - Management of phenomena liable to release significant quantities of mechanical energy in case of core melt - Determination of releases in the event of a severe accident (fission product transfers and trapping from the fuel to the sodium, distribution in the primary circuit and transfer to the gas cover space) - Long-term management of severe accidents
VHTR	- Management of the confinement function (primary circuit contamination, particulate waste management); source term in normal operation and accident conditions - Earthquake behaviour (balance of graphite structures) - Thermomechanical impact of severe transients on fuel - Definition of a severe accident, release limitation and monitoring strategy in case of severe accident, release assessment - Research into air ingress, water ingress and dust and/or hydrogen explosion (water thermolysis) accident risks

GFR	- Control and nuclear monitoring of localised reactivity insertion accidents - Control of the decay heat removal function - Fuel element degradation mode(s) - Primary circuit depressurisation transients - Severe accident management (core catcher, risk of secondary excursions, etc.)
LFR	- Management of structural corrosion risks - Control and nuclear monitoring of local reactivity insertion accidents - Research into fuel assembly blockage and air ingress accident risks - Control of fluid-structure interaction phenomena (during seismic events, for example) - Natural convection in the core and decay heat removal systems - Feasibility of in-service inspection - Severe accident behaviour, in particular with regard to the ability of molten fuel to float to the surface of the lead and the effectiveness of the mitigation provisions - Management of the toxicological consequences of environmental release of lead - Management of the lead freezing risk, in particular in shutdown states - Determination of releases in the event of a severe accident (fission product transfers and retention from the fuel to the lead, distribution in the primary circuit and transfer to the gas cover space) - Management of phenomena liable to release significant quantities of mechanical energy in case of core melt - Limiting worker exposure to ^{210}Po
MSFR	- Management of structural corrosion risks - Reactivity management - space-coupling and time-coupling between the nuclear aspect and thermal-hydraulic, physicochemical and mechanical aspects - Approach to criticality during first loading and startup phases - Management of fissile material and power distribution in the fuel salt - Reliability and performance of the fuel salt emergency draining system - Feasibility of in-service inspection - Management of the salt crystallisation risk - Definition of a severe accident – risk of high-energy phenomena (thermodynamic reaction / chemical reaction between salt and water) - Management of releases of toxic products such as hydrofluoric acid and salts bearing radioelements from the reactor - Limiting worker exposure during maintenance operations

SCWR	- Management of fuel cladding and structural corrosion risks - Effectiveness of mixing in the relevant core areas and stability of flow rates through the various fuel assemblies - Nuclear stability and control of reactivity arising from the coupling with thermal-hydraulic aspects - Behaviour of supercritical water when subjected to neutron flux (radiolysis) - Water behaviour in the pseudo-critical region - Management of depressurisation accidents - Core behaviour in severe accident conditions - Management of phenomena liable to release significant quantities of mechanical energy in case of core melt

9. FUEL CYCLES ASSOCIATED WITH THE SYSTEMS SELECTED BY GIF

This chapter provides a brief overview of current knowledge concerning the fuel cycles that can be associated with the six Generation IV nuclear energy systems chosen by GIF. It examines the maturity of the processes and technologies currently in development and explains the main safety and radiation protection issues identified so far. The overview also covers aspects related to waste management. Based on the knowledge currently available, the deployment strategies and capabilities of each system are also examined (particularly availability and management of the nuclear materials) along with their impact on fuel cycle management. This chapter discusses in more detail:

- the fabrication and reprocessing processes for fuels usable in SFRs, GFRs and LFRs (Sections 9.1 to 9.5);
- fuel cycle-related issues for MSRs (Section 9.6), VHTRs (Section 9.7) and SCWRs (Section 9.8);
- the constraints associated with transporting fuel (Section 9.9).

For its evaluation, IRSN has used information from the "RNR-Na" project and other documents that will be mentioned later in the text.

Concerning fuel cycle facilities safety, it is not possible to refer to safety objectives defined for the design and construction of the next generation of facilities, as is the case for reactors. However, it would be desirable for the general safety objectives associated with Generation IV systems to take into account the associated fuel cycle facilities, in a manner yet to be defined. In particular, IRSN considers that the general safety orientations listed below should be followed at the very least:

- improving the implementation of the defence-in-depth principle is in these facilities;
- strengthening the measures for preventing and limiting the consequences of postulated accidents (consideration of accidents that could lead to significant releases into the environment, internal and external hazards, etc.);
- improving the robustness of safety demonstrations (quantification of margins, consideration of cliff-edge effects, etc.);
- improving safety and radiation protection during operation (limiting the doses received by workers in normal and accidental situations, etc.).

The general assessment made of the safety of future fuel cycle plants depends heavily on how well developed the fuel cycle processes are and how well defined the associated risks are. It is already well known that some innovative or less well-tested processes (particularly pyrochemical processes, etc.), will require entirely new safety approaches, and difficulties with demonstrating that some risks are properly controlled cannot be ruled out.

9.1 FUELS USABLE IN SFR, GFR AND LFR SYSTEMS

SFR, GFR and LFR fast reactors, which are associated with a "closed" fuel cycle, are capable of recycling plutonium and uranium on an unlimited basis. They therefore make much more efficient use of natural uranium than ordinary water reactors. Once the quantities of plutonium required for their deployment have been stored, "isogenerating" fast reactors can operate without any more fissile material being added externally and are fed solely with depleted or reprocessed uranium. The uranium enrichment stage is therefore no longer necessary. Moreover, fast reactors can either be breeder reactors, enabling the fleet to be deployed fairly quickly, or burners, maintaining the plutonium inventory at a desired level or else reducing stocks at the end of fleet life (cf. CAPRA concept: Consommation Accrue de Plutonium dans les RNR (Increased plutonium consumption in FRs)). This increased fuel consumption must, however, remain compatible with the reprocessing option (feasibility of fuel dissolution).

The use of a Th-^{233}U cycle can also be considered in fast reactors, though the breeding performance is not as good as that obtained with a U-Pu cycle (because plutonium has better neutron qualities than uranium-233 in a fast neutron spectrum). The U-Pu cycle is quicker to deploy than the Th-U cycle because the later requires a long transition phase to produce the initial uranium-233 stock needed (two fuel cycles would have to be run in tandem at least at the start, it would be necessary to use thorium-MOX fuels for example, materials management would be quite complex, etc.). The need to produce the solid fuels containing uranium-233 in shielded cells (because of the presence of uranium-233 and its descendants, which are high energy gamma emitters) is one of the major drawbacks of the closed fuel cycle with thorium, though the use of a hydrometallurgical process could be envisaged (the THOREX process, for which a first operating experience feedback exists on a pilot scale), provided that a significant amount of R&D takes place to demonstrate the safety of this process. Finally, in equilibrium, the Th-U cycle produces far fewer minor actinides than the U-Pu cycle (minimising the inventory in a geological repository).

At the current stage of development, there are several types of fuel that could be used in these three types of reactor, according to their specific nature: ceramic fuels (oxide, carbide or nitride), and metal alloys (U, Pu, Zr). In particular, the metal alloys and the carbide and nitride ceramic fuels, which are denser and offer higher thermal conductivity than oxide, are of particular interest, notably for the rapid deployment of a fleet of fast reactors (FRs). They would enable the plutonium inventory of a reactor core, and its volume, to be significantly reduced and would deliver high regeneration gains. However, there is very little industrial experience with these fuels as regards fuel cycle operations. Deploying these fuels in a fuel cycle on an industrial scale would require a major R&D effort in a wide range of fields. Anyway, oxide fuel is still relevant for deploying an FR fleet at a slower rate.

Concerning oxide fuels or even nitride or carbide fuels, closure of the fuel cycle is considered using a hydrometallurgical process based at a centralised plant associated with several reactors. However, for metallic fuels, pyrometallurgical processing is envisaged in units that could be integrated into the reactors. In any case, the main issue at stake is reliably providing a continuous fuel supply to the FRs.

9.2 OXIDE FUELS

9.2.1 *FABRICATION ISSUES*

The fabrication of oxide fuels, particularly for SFRs, should be based on the COCA process developed at Cadarache (optimised ball milling and blending of UO_2 and PuO_2 powders), taking account of the vast amount of experience acquired with fabrication of the Phenix and Superphenix fuels (more than 100 tonnes produced at Cadarache).

However, in view of the specific features of the future MOX fuels for fast reactors (particularly their increased Pu content compared with PWR fuels), this experience, especially as regards radiation protection, has shown that developments in the fabrication process will be necessary for new fuels, notably to limit the doses received by operators and achieve the capacities required on an industrial scale (some 400 tonnes per year for a fleet of 60 GW fast reactors at equilibrium). This will require significant optimisation of the process (simplification of the stages by using coprecipitated powders $(U,Pu)O_2$ from the COEX^{TM} process, improvement of pellet quality (mixing, pressing, sintering), development of new organic additives more resistant to irradiation, reduction of fabrication scraps, etc.) and technological innovations (equipment compactness and capacity, control automation, maintenance, minimisation of retentions, etc.). Consequently, a vast amount of research, including the technology validation stage, will have to be carried out. Particular consideration will have to be given to whether the processes are installed in glove boxes, because of the radiation protection constraints, as this will affect the structure of this type of plant. For the most problematic operations in terms of doses, alternative solutions (shielded cells, etc.) will have to be considered.

From a safety point of view, the design of future fabrication units (ventilation and cooling systems, etc.) should not require any significant technological innovations. However, particular attention will have to be paid to the management of slow accumulations of fissile materials within containment enclosures. Finally, the quantities of technological waste not suitable for surface disposal produced by future fabrication plants should be consistent with those produced by the MELOX plant at equivalent production levels. However, special decontamination processes may need to be developed to limit the amount of fissile materials in this waste.

9.2.2 *IRRADIATED FUEL REPROCESSING ISSUES*

The standard process for reprocessing spent MOX-FR fuels (with a “closed” cycle) is the hydrometallurgical process. These fuels are soluble in nitric acid and the PUREX process is industrially proven. In principle, pyrochemical reprocessing could be considered, especially the Dimitrovgrad Dry Process (DDP) mentioned in references 90 and 91. However, this process, which is associated only with the fabrication of fuel pins using the vibro-packing technique, has its drawbacks and some peculiarities (it is difficult to obtain a uniform distribution of fuel in the pellet, the technique cannot be used to make annular fuel pellets, the fission product contamination factors of the U and Pu are very poor, etc.) making it incompatible in principle with the SFR core and fuel element concepts developed so far in France.

Experience with reprocessing around 25 tonnes of fuel from Rapsodie and Phenix reactors at the Marcoule and La Hague plants has shown that it should be possible to use the PUREX process as the basis for a future industrial reprocessing plant for spent MOX-FR fuels. However, because of specific characteristics of these fuels and the industrial timescales contemplated, special adaptations and implementation conditions will have to be developed. This will also profoundly affect some of the safety analyses and associated measures (greater criticality risks linked to bigger plutonium streams, higher insoluble residues contents, higher decay heat, etc.). Moreover, new operations will be needed for dismantling the fuel pins and removing the spacer wire around them. Thus there is still a vast amount of research to be done, particularly to define conditions for dissolving the irradiated fuel that give the expected solubilisation yields ("digestion" of the insoluble residues, etc.) and to develop technologies for dismantling the fuel assemblies, separating the fuel from its cladding if necessary (to overcome problems associated with cladding corrosion in the dissolution medium), dissolving the fuel continuously, clarifying the dissolution medium and co-converting the resulting purified solutions of uranium and plutonium . This series of developments should make it possible to achieve an industrial-scale capacity for reprocessing several hundred tonnes a year for a fleet of fast reactors at equilibrium.

CEA considers that current research and development, and R&D planned for the next few years, should bring the process to a sufficient level of maturity by 2040-2050 and allow the necessary adaptations to be defined. The fuel cycle facilities associated with the ASTRID reactor will aim to qualify these developments on a significant scale.

9.3 CARBIDE AND NITRIDE FUELS

Since the 1960s, major research and development programmes have been run on the fabrication of carbide and nitride fuels. However, until now, no experience feedback is available on an industrial scale. Only India has experience in the fabrication, use in an SFR, and reprocessing of carbide fuel (FBTR experimental reactor – see Chapter 2 and Appendix III). In the context of the current GIF projects, carbide and nitride fuels are associated with GFRs and LFRs respectively.

9.3.1 *FABRICATION ISSUES*

The standard process currently used for the fabrication of both these types of fuel is carbothermal reduction from uranium and plutonium oxides. Because of the constraints associated with these materials (pyrophoricity, reactivity with oxygen and water vapour) and experience acquired with processes using "powders", the operations involved in fabrication should be performed in containment enclosures with an inert, controlled atmosphere and will require significant maintenance. Particular attention will need to be paid to the safety of these processes (containment, pyrophoricity of the materials, etc.) and to radiation protection-related aspects (similar to those associated with the fabrication of oxide fuels, discussed earlier). Additionally, the prevention of criticality risks needs to be a particular focus for development because the intrinsic reactivity of these materials is higher than that of the ceramic oxides (higher densities, moderating and reflective effect of the carbon, etc.). Qualification of the computing methods also requires consolidation (particularly by developing benchmarks). In any case, a large amount of research is needed to develop and optimise these fabrication processes, for both the standard process (control of the carbothermic reduction stage and synthesis conditions, quality of the ceramic) and the main alternative processes studied (oxalate co-precipitation, sol-gel processes).

Finally, it seems in principle that the quantities of technological waste produced that are not suitable for surface disposal should be similar to those produced by plants fabricating oxide fuels. The pyrophoricity risks associated with their reprocessing should also be studied particularly carefully.

In conclusion, the industrial feasibility of the fabrication of carbide and nitride fuels does not appear to have been demonstrated satisfactorily at present. In view of the specific risks linked to the nature of these ceramic fuels, the design principles of future plants to be used for the fabrication of several hundred tonnes of these fuels per year will require detailed analysis when the time comes, if the decision is made to develop this type of fuel.

9.3.2 *IRRADIATED FUEL REPROCESSING ISSUES*

The standard process used for reprocessing spent carbide and nitride fuels is the hydrometallurgical process (PUREX process). These fuels are soluble in nitric acid. Throughout the world there has been a large amount of R&D in this area, and India especially has solid experience on a pilot scale.

The feasibility of pyrochemical processing (electrorefining, salt/metal reductive extraction), which could also be considered, has not been demonstrated on a laboratory scale, and certain process-induced constraints could prove insuperable, particularly in the case of carbide fuels. Thus there is a lot of work to be done on this process, though some of the areas for development are identical to those identified for reprocessing metallic fuels (see Section 9.5 below). Pyrochemical processing can therefore only be envisaged in the long term.

Because these fuels are particularly pyrophoric, mechanical processing operations (dismantling, shearing or destructuring, cladding/fuel separation) have to be carried out in an inert atmosphere. These operations could cause specific difficulties, particularly in the case of GFR fuel elements (which have refractory cladding).

The solubility of irradiated carbide and nitride fuels and the amounts of insoluble residue recovered after dissolution should be comparable, to a first approximation, with those for irradiated oxide fuels. Finding out this information will be essential for analysing the safety of the dissolution and clarification stages, particularly as regards criticality risks.

However, carbide fuels differ from oxide fuels because of the formation of carboxylic and aromatic acids when they dissolve. These compounds can be powerful complexing agents for plutonium and can cause considerable extraction cycles malfunctions (Pu loss into the raffinate, retention in the discharged solvent, etc.). Mineralization treatments using highly oxidising reagents have been developed to limit the presence of these organic complexing agents. However, these treatments, which complicate the process and have significant disadvantages in terms of safety and waste management, have not proven to be sufficiently effective. This therefore continues to be a sensitive point in demonstrating the feasibility of reprocessing these fuels.

Meanwhile, when nitride fuels are dissolved, on the one hand large quantities of ammonium ions in solution are formed, which can cause an explosion risk at the vitrification stage (though adjusting the dissolution conditions can limit the quantities produced), and on the other hand very large amounts of carbon-14 are released if the nitride fuel is not sufficiently enriched beforehand with nitrogen-15 (the performance level to be achieved as regards the recovery and trapping of such large quantities of carbon-14 is not possible at the current stage of development). This is a major drawback when it comes to closing the nitride fuel cycle, because recovering the nitrogen-15 and recycling it seem to be necessary from a technical/economic point of view. To overcome these drawbacks, some indirect processes have been studied, based on the conversion of irradiated carbide and nitride fuels to oxides by oxidation prior to the dissolution step (oxidation in oxygen or CO_2 at high temperatures, pyrohydrolysis, etc.). Although performing this preliminary step does not appear to pose insuperable problems from the point of view of the process chemistry, the associated risks should be carefully investigated (hydrogen or ammonia formation, etc.). A suitable process for treating the gases would in any case be necessary (release of gaseous or volatile fission products such as ^{85}Kr, ^{3}H, ^{129}I, ^{106}Ru, ^{137}Cs, etc.). This "indirect" process would mean adding an extra stage at the head-end plant, which could cause technological problems on an industrial scale.

In any case, significant research and development is necessary to demonstrate the feasibility of the industrial reprocessing of spent carbide and nitride fuels, either directly or indirectly. Consequently, it appears that the use of these fuels, which would require major changes to the current fuel cycle, could only be envisaged in the longer term.

9.4 MANAGING PROCESS WASTE FROM THE REPROCESSING OF OXIDE, CARBIDE AND NITRIDE FUELS

As regards management of the waste from the reprocessing of oxide, carbide and nitride fuels for FRs, studies done by CEA show that, compared with the management process defined for the current fleet of PWRs:

- The quantity of fission products produced in an FR is smaller than the quantity produced in a PWR (for example, 106 kg/TWhe of fission products are produced with a MOX-FR fuel irradiated at 123 GWd/t instead of 133 kg/TWhe for a UOX fuel in a PWR irradiated at 45 GWd/t). This variation is due mainly to the better energy efficiency of fast reactors.
- The quantity of minor actinides formed depends on the isotopic composition of the plutonium and the burn-up rate. Thus for example, with an FR fuel at the first reload, where the plutonium comes from reprocessing MOX-PWR fuel irradiated at 45 GWd/t, the quantity of minor actinides for a burn-up rate of 123 GWd/t is estimated to be 7.2 kg/TWhe, whereas for an FR fuel at equilibrium, the quantity is only 3.8 kg/TWhe. For information, with UOX-PWR and MOX-PWR fuels, where the burn-up rate is around 45 GWd/t, the quantities of minor actinides produced are 3.6 kg/TWhe and 17 kg/TWhe respectively (or 4.6 kg/TWhe for the PWR/UOX+MOX fleet).
- The ratio of the mass of the structural waste (hexagonal tube, head with upper neutron shielding, head , spacer wires, cladding, etc.) of an SFR (or even LFR) fuel assembly to TWhe is around three times higher than in the case of a PWR fuel assembly. With GFRs it is difficult to say what the ratio is because there is no consolidated design for GFR fuel assemblies.

The greater mass of insoluble elements (platinoids) and cladding corrosion products in the solutions to be vitrified could increase the number of vitrified waste packages produced compared with the number produced when reprocessing UOX-PWR fuels. However, the number of packages could be reduced by changing the current specifications for these packages, or even by defining a new glass matrix or deploying a separate process for managing the dissolution fines (conditioning in a metal matrix or special glass). Significant R&D activity is planned by CEA to look into this in the case of oxide fuels.

There is also research to be done to reduce the mass of metallic structures waste from SFR fuel elements (dismantling of assemblies, reuse of the upper neutron shielding, etc.), to limit the activation levels of these elements and to develop alternatives to the current compacting process used to reduce the volume of waste produced and its contamination levels (melting with slag, cold crucible melting and pulling, etc.). These goals have been taken into account by CEA in its research programmes. Similar processes and technologies could be envisaged for structural waste from LFRs.

9.5 METALLIC FUELS

IRSN's assessment is based on the documents listed as references 21, 92, 93 and 94.

Metallic fuels consist of U or U-Pu metal generally as an alloy with zirconium.

IRSN indicated in Section 2.6 on SFRs that it would be worth looking into the respective advantages and disadvantages of MOX fuel and metallic fuel in terms of safety, especially as regards controlling reactivity and behaviour during a serious accident, so as to justify the choice eventually made.

Hydrometallurgical and pyrochemical processes have been developed for reprocessing spent metallic fuels.

The hydrometallurgical process seems to be particularly problematic in terms of both its technological feasibility and its safety (the metal is insoluble in nitric acid so concentrated hydrofluoric acid has to be used, the presence of sodium bonds is incompatible with aqueous reprocessing technology, a stage has to be added for the reduction of the recovered uranium and plutonium oxides to metal, etc.). The pyrochemical electrorefining process in molten chloride medium, developed as part of the American IFR (Integral Fast Reactor) project, coupled with a vacuum melting and injection casting process for fabricating fuel rods, seems much more suitable. These pyrometallurgical processes are currently the standard international processes (developed in the United States, Japan, South Korea and India). However, they still require further major technological developments to be deployed on an industrial scale. Where metallic fuel is reprocessed using electrorefining techniques, recovering the uranium does not present any difficulties. But multiple recycling of uranium and plutonium on their own seems to be difficult, with grouped separation of the actinides appearing to be inevitable in the processes being considered at the current stage of development. In addition, decontamination of the uranium and plutonium from fission products appears to be limited. The fuel rods will therefore have to be produced in a shielded cell, since radiological constraints and problems related to the release of heat can also increase with further recycling. This seems to be a tricky aspect of demonstrating the feasibility of pyrochemical reprocessing. Consequently, major development programmes are still needed before an industrial-scale process can be defined (turning a discontinuous process into a continuous process, improving the decontamination of the finished products (uranium

and plutonium), overcoming technological problems with corrosion and with managing metallic waste and contaminated salts, etc.).

Similarly, major research is still required to develop the aforementioned fabrication process on an industrial scale (to control the fabrication parameters, minimise fission product and actinide volatilisation phenomena, develop a continuous melting process, minimise the amount of technological waste, etc.). Additionally, measures to prevent criticality risks, which need to be more stringent for metallic materials, need to be examined particularly carefully. However, it should be possible to control the risks associated with using this process.

In conclusion, it seems that the use of metallic fuels on an industrial scale can only be envisaged in the long term. The use of these fuels would constitute a major shift from the current technologies for reprocessing spent fuels (hydrometallurgical process) in France, and it may be necessary to have both of the technologies mentioned above (the PUREX process and the pyrochemical process) running in tandem during the transition phase (where PWRs fed with MOX fuels and SFRs fed with metallic fuels are operated alongside one another).

9.6 MSRs

Fast spectrum MSRs of the MSFR type are breeder reactors (see Chapter 6). They use molten eutectic alloy-type salts (lithium fluoride) as both liquid fuels and coolants. MSRs are associated mainly with a thorium fuel cycle (^{233}U-Th). A fuel reprocessing unit is integrated into the reactor to maintain the chemical integrity of the salt (extraction of the soluble lanthanides) and the redox potential of the U^{4+}/U^{3+} couple in the salts (see Figure 36 in Section 6.2.1). This unit reprocesses several dozen litres a day. The daily reprocessing of this small volume is only possible because of low core poisoning levels (fast spectrum).

As far as is currently known, the MSFR has a number of advantages (incineration capacity, breeding capacity, saving of natural resources, recycling of the actinides, etc.). The concept was tried out in the United States (MSRE experiment), where the conclusion was reached that it was viable. However, the experiments were very limited. So the feasibility of the concept and the associated reprocessing has not yet been established. This is particularly significant given that some very specific problems are associated with the concept (containment of liquid fuel, choice of materials, corrosion by the molten salts, reprocessing of the salt, processing of the ultimate used salts, etc.). It therefore seems that these prospective technologies, which represent a complete break with the current technologies, will probably not be accessible until at least the second half of this century given the major developments and technological breakthroughs needed.

CNRS has presented an “embryonic” scenario for the installation of a fleet of MSFRs, which includes the doubling of the currently installed nuclear power capacity of the PWR reactors (to 120 GWe), in order to test capability for deploying this type of reactor in challenging conditions. IRSN does not currently consider this to be a realistic scenario. However, new scenarios for deploying MSFRs at a constant installed capacity (60 GWe) are currently being developed to take account of the context in France. If the decision were made to continue research into this type of reactor to evaluate whether it was genuinely feasible and how relevant it would be to implement, a vast amount of R&D would have to be done on these reactors and the associated reprocessing technologies. Prior to any MSFR demonstration reactor going into operation, besides resolving the inherent problems of the reactor (resistance of the materials, reliability of the technology, etc.), numerous studies and experiments would have to be done to find out the information needed to define reprocessing schemes (performance of the system for

removing noble gases and aerosols from the fuel salt by helium bubbling, behaviour of the volatile metal fluorides, iodine and tritium, problem associated with the non-extraction of the zirconium, separation performance of the alkalis, alkaline earths and lanthanides, etc.) and to validate the technological solutions under real operating conditions (loop of molten salts under forced circulation, sparging system, technologies and resistant materials in a highly corrosive environment, etc.).

Lastly, a significant amount of R&D will be necessary to define the processes for reprocessing and conditioning the waste produced ("metallic" waste from the liquid filter in the bubbling system, residual lanthanides and minor actinides extracted using a pyrochemical process, spent fuel salts, the many items of equipment and heat exchangers replaced during the life of the reactor, technological waste, etc.).

9.7 VHTRs

IRSN's assessment is based mainly on references 50, 93, 95, 96, 97 and 98.

The VHTR is a thermal neutron spectrum system that uses TRISO particle fuels (see Chapter 3). It is notable for its great flexibility with regard to the fuel used. Because of its refractory fuel in the form of coated particles, it is able to use all types of fissile and fertile materials (low enriched uranium (less than 20% uranium-235), MOX, plutonium, uranium-238, uranium-233 and thorium).

The concept uses an open fuel cycle because its fuel is by nature difficult to reprocess. The VHTR does not allow natural resources or waste to be managed efficiently in the longer term, and savings in natural uranium are very limited, particularly with a low enriched uranium fuel. VHTRs, compared with light water reactors, are potentially good "burners" of plutonium (U.S. "Deep Burn" project). They relate more to a dedicated plutonium consumption concept than a type of reactor designed to generate power and destined for large-scale deployment. These reactors could "incinerate" the plutonium in MOX-PWR fuels, thus stabilising the plutonium inventory in the cycle or significantly reducing it in the context of a nuclear phase-out, or even adapting it to the requirements of a future deployment of a few fast reactors.

Based on experience acquired since the 1960s with fabrication of TRISO and BISO particle fuels on a laboratory and pilot scale (Peach Bottom, DRAGON, Fort Saint-Vrain, AVR, THTR notably using oxide and carbide fuels based on low enriched and highly enriched uranium as well as thorium), it should be possible to demonstrate the industrial feasibility of this stage for VHTRs (mainly for fuels based on enriched uranium) with processes and technologies currently being developed (internal and external gelation processes). However, lack of experience with the fabrication of plutonium-based VHTR fuels means that specific R&D will be required, and the safety of an "all-plutonium" core will also need to be demonstrated. Research programmes are currently in progress to design fuel particles that can reach even higher burn-up rates and irradiation temperatures (development of an oxycarbide phase instead of oxide, coating with zirconium carbide instead of silicon carbide, etc.). Controlling the manufacture of very large numbers of particles, as regards their containment characteristics, could be something of a technological challenge (use of automated non-destructive methods, etc.). More generally, as well as requiring the adaptation particularly of existing enrichment facilities if low enriched uranium (with a uranium-235 content below 20%) is used, the fabrication of fuels from particles based on enriched uranium or plutonium will also necessitate the construction of a specific facility. It should also be possible to demonstrate the safety of the fabrication processes without too much difficulty based on processes currently known. However, the risks

associated with the pyrophoricity of some materials (e.g. carbide matrices) and the criticality risks (high plutonium content, enrichment with uranium-235 to more than 6.5%, used in a liquid medium, presence of carbon as a moderator, etc.) should be investigated in detail. In particular, the resulting criticality safety constraints could be incompatible with the imperatives of an industrial process, notably in terms of throughput.

With the "open cycle" option, disposal of the structural waste and fuel as it is (pebbles or prismatic fuel assemblies and compacts) seems in principle to be problematic in terms of waste volume. The graphite to be disposed of would, as a rough estimate, come equally from the fuel elements and reflectors used (total production of around 3500 tonnes of graphite per year for a 10 GWe HTR fleet, including the quantities produced during dismantling). This waste and materials could be placed in a near surface disposal facility. The quantities involved should lead to alternative management solutions being proposed, consisting of separation of the compacts from the prismatic assemblies (the fact that these fuels are "sealed" by design is in principle compatible with this) and of reducing the activity of the structural waste due to its activation (limitation of impurities), and finding solutions for decontaminating and recycling the graphite. Various research programmes which form part of the VHTR development plan drawn up by GIF are being conducted along these lines. In any case, demonstrating that the management of waste from the VHTRs is under control will be an important factor in any decision-making. In particular, acceptable long-term behaviour of the VHTR fuel when disposed of in a geological formation will need to be demonstrated, together with appropriate conditioning, which still remains to be defined.

Lastly, the industrial feasibility of a "closed" cycle for VHTRs has not been demonstrated, with some major technological breakthroughs necessary, particularly as regards the removal of the layers of coating on the particles to access the fuel (e.g. a mechanical, thermal or pyrometallurgical process for destroying the structure of the layers). Attempts to achieve a high burn-up rate for this type of fuel, particularly for plutonium-based cycles, are further reducing the interest of recycling this actinide (low residual quantities and poor isotopic quality). The development of an industry for reprocessing fuel from VHTRs operating as thermal neutron spectrum systems therefore seems very complex where interest is so low. However, using a thorium cycle based on a mixed fuel (233U-Th) in the longer term could make the possibility of a closed cycle more credible. Fertile BISO fuel particles (which have no refractory layer made from silicon carbide) are in principle not as resistant as fissile TRISO particles and would therefore be easier to reprocess. A significant amount of R&D is still needed, however, to demonstrate the feasibility of these processes (sorting through particles, selective dissolution, etc.), but deploying this type of cycle would still be a long and complex process (see Section 9.1).

9.8 SCWRs

The SCWR constitutes an advance on current water reactors, mainly from the point of view of its thermodynamic efficiency. It operates with a thermal neutron spectrum and is offered with an open fuel cycle. Demonstration of the feasibility of a breeding fast-spectrum core and multiple plutonium recycling in a closed cycle is a long way off as research currently stands. Potential savings in terms of natural uranium consumption with an SCWR fleet compared with the current PWR reactors would as a general estimate be directly proportional to the thermodynamic efficiency of the reactor. From the point of view of fuel fabrication, there does not appear to be any insuperable barrier based on industrial experience with fuels for light water reactors. However, the "open cycle" option would necessitate direct disposal of the spent fuels, requiring an increase in the underground footprint of the repository because of the decay heat of the packages.

Mono-recycling of uranium and plutonium in SCWRs could prove attractive because of the high residual uranium-235 content in the spent fuels, and reprocessing them should not present any insuperable problems, provided that the fuel assemblies can be designed to make them easy to dismantle and to facilitate the mechanical treatment carried out at the head-end plant.

9.9 TRANSPORT

Transport for fuels from SFR, GFR, LFR, VHTR and SCWR reactors should not require any major technological breakthroughs in relation to the means of transport currently in use and could benefit from feedback from previous experience (transportation of dismantled SFR fuel pins and HTR pebbles that has already taken place, etc.). Studies and research will still be needed to accommodate the specific requirements of these fuels and prevent these requirements from becoming insuperable problems. The dismantling and cutting up of some types of assemblies (e.g. LFR) may be necessary depending on their characteristics.

On the other hand, transporting irradiated molten salts or salts containing actinides from the reprocessing of PWR or MOX-Th fuels may require that new types of packaging have to be developed to take account of thermal and radiation protection constraints and the physical and chemical characteristics of the salts being carried.

9.10 CONCLUSION

In the current state of the art based on the available research, only Generation IV nuclear systems based on multiple recycling of uranium and plutonium in fast reactors seem likely, in the current French context and in the medium term, to offer prospects compatible with the criteria taken into account by GIF (sustainable management of nuclear materials, limited quantities of final waste, etc.).

In view of the factors discussed above (significant experience feedback with French SFRs and their fuel cycle, planned R&D programmes, ASTRID demonstrator project and the facilities associated with its fuel cycle, etc.), the implementation of a scenario in which SFRs are deployed to replace the entire PWR fleet by the end of this century seems eminently possible.

Of all the potential fuels for the different types of Generation IV fast reactors examined, oxide fuel, with which there is already vast experience feedback, seems to be the most mature. The corresponding fuel cycle could be deployed the most quickly, with the completion of studies and accomplishment of technological innovations that have already been partially identified. CEA's timetable in its report listed as reference 12, submitted in

December 2012, appears in principle to be technically compatible with beginning the deployment of an SFR fleet and its associated fuel cycle by 2040-2050.

However, if deployment had to be postponed, it might be necessary to review the merits of the different Generation IV nuclear systems in the light of newly-acquired knowledge. Also, the emergence of new nuclear power generation scenarios, taking account of industrial constraints (for example CEA and the companies AREVA and EDF are currently looking at (very) gradual FR deployment scenarios to partially replace the current PWR fleet) and phase-out scenarios, could influence current assessments of these systems. Because of sometimes very different timescales for deployment, major differences in maturity and wide variations in levels of knowledge of the different concepts examined, it is difficult to make a balanced assessment of the nuclear systems chosen by GIF. The insights given in this report should therefore be viewed with some caution at present.

Lastly, it should be borne in mind that any decision to deploy a particular type of Generation IV reactor on an industrial scale, if such a thing is decided, will be based on its advantages not only in terms of the operation and safety of the reactors, but also in terms of the coherence and performance of the associated fuel cycle, including aspects related to safety, radiation protection, materials management and minimisation of the quantities of radioactive waste produced, without prejudging the overall economic competitiveness of the nuclear system. Ultimately, the choice of system must be made as part of an integrated approach, based on studies that cover multiple criteria and all the aspects mentioned above.

10. TRANSMUTATION OF LONG-LIVED RADIOACTIVE ELEMENTS

At the request of the French Nuclear Safety Authority (ASN), IRSN produced the notice listed as reference 5, based on the assessment of the progress report concerning an initial review of studies of technical and economic scenarios conducted since 2007 as part of the work on the separation and transmutation of long-lived radioactive elements in spent fuels from nuclear power plants. This report has since been completed by a second volume (reference 12) submitted to the relevant ministers in December 2012 to meet a deadline set by the French Act of 28 June 2006 on the sustainable management of radioactive materials and waste, which requires a review of this research. On the basis of IRSN's notice, ASN in turn stated its position on this matter in its notice listed as reference 99.

Furthermore, CEA has also submitted information concerning the separation and transmutation capacities of the Generation IV systems selected by GIF.

IRSN based its assessment of the value of having a process for managing radioactive waste that involves the separation and transmutation of long-lived radioactive elements on all of these documents. This assessment for the most part restates the conclusions in the IRSN notice mentioned above.

10.1 BASIC PHYSICS OF TRANSMUTATION

Transmutation consists in using a neutron flux to convert radionuclides with a long or very long half-life into stable or short-lived elements. Transmutation is therefore the action by which the absorption of a neutron by the radioactive nucleus of a long-lived radionuclide changes its radioactive characteristics, or if the nucleus is fissile, causes a potential fission reaction.

Thus the isotope technetium-99 (99Tc), a long-lived fission product with a half-life of 210,000 years, is converted by means of neutron capture into technetium-100 (^{100}Tc), which has a short half-life (15.8 seconds). Through radioactive decay, this turns into ruthenium-100 (^{100}Ru), which is stable.

As regards radionuclides with heavy nuclei, the actinides are all the elements in the periodic table above actinium. Among these actinides, uranium (U) and plutonium (Pu) are the main elements in nuclear fuel. These elements containing fissile isotopes can be recycled in nuclear reactors. The so-called minor actinides are mainly neptunium (Np), americium (Am) and curium (Cm). All the isotopes of these radionuclides are α emitters with long or very long half-lives, and some are also ß emitters. The masses of the other actinides in spent fuel (actinium, thorium, protactinium, berkelium, californium, einsteinium and fermium) are low enough for them to be considered as traces. The minor actinides are currently considered to be waste and are incorporated with the fission products into a glass matrix. The vitrified waste containers produced in this way are intended, once cooled, to be sent for disposal in deep geological formations.

Unlike with fission products, the neutron capture reaction in the case of minor actinides can lead to the formation of other minor actinides potentially with a longer half-life. However, if the reaction leads to a fission reaction, the minor actinide splits into two (or sometimes three) fragments, most of which have a shorter half-life (less than 30 years). Consequently, this should be the favoured process for the transmutation of minor actinides. Moreover,

the fission reaction produces additional neutrons that can be used for the transmutation of other nuclei or to participate in maintaining the chain reaction.

Based on a fundamental analysis comparing effective capture and fission cross-sections in thermal, epithermal and fast reactors, it appears that the transmutation of long-lived fission products is effective in the epithermal neutron spectrum, while for minor actinides it is in the fast spectrum. This analysis reveals the intrinsic advantage of fast reactors for transmuting minor actinides. On the other hand, studies done by CEA show that the separation and transmutation of long-lived fission products on an industrial scale is "very difficult, or even impossible."

In the case of fast reactors, two modes of transmutation are being studied: homogeneous transmutation and heterogeneous transmutation. In homogeneous mode, the minor actinides are incorporated in small proportions into the fuel in the reactor core (uranium and plutonium), while in heterogeneous mode, the minor actinides are concentrated in specific UO_2_based target fuel (minor actinide-bearing blankets - CCAMs) or are mixed with an inert matrix in targets, and these assemblies or targets are placed in core periphery zones.

10.2 CONTEXT

For the scenarios (which stop arbitrarily in 2150), CEA is considering a gradual deployment of a fleet of fast reactors from 2040, to replace the current PWRs and the future EPRs installed during a transition phase. It restricted its detailed studies of scenarios to SFRs in which the plutonium is systematically recycled, considering the following configurations: no minor actinide transmutation (reference scenario in which only the plutonium is recycled), and homogeneous or heterogeneous transmutation of all minor actinides or americium only.

IRSN only evaluated the SFRs (MOX fuel), GFRs (carbide fuel, (U,Pu)C) and dedicated systems (Accelerator-Driven Systems) included in the scenarios presented.

This evaluation showed that:

- The Accelerator-Driven System scenario for 2040 was unrealistic because of the number of Accelerator-Driven Systems required, the technical feasibility of the reactors (design, control, fuel) and the operations in the cycle, which would not be established for several decades, and because of the associated transport flows, which appeared to be prohibitive (extremely high thermal output from the fuel).
- The deployment by 2040 of a fleet of GFRs was unlikely (no operational experience, resistance of the fuel and structural materials to high temperatures, need for large quantities of plutonium for deployment).

However, the choice of scenarios seems to depend heavily on the start date of deployment of the SFR fleet (2040), which requires that the solutions being considered have reached a certain level of maturity. If the deployment date were to be pushed back, studies necessary to take account of newly-acquired knowledge about the Generation IV systems, which have currently been ruled out, would have to be reinstated.

10.3 EVALUATION OF THE VALIDITY OF THE SCENARIOS

Assessments of the different options chosen as regards their fuel cycles are generally based on scenario studies, the purpose of which is to evaluate the merits of these scenarios by, on the one hand, establishing the situation of the fleet at equilibrium[45] and, on the other, shedding light on the transitional period between the current situation and the eventual situation. This period can be lengthy and difficult to manage as regards the availability of materials, reactor deployment capability and capacity for recycling or accumulating stored materials.

10.3.1 *REFERENCE SCENARIO*

As explained in the previous chapter, implementing the reference scenario (multiple recycling of plutonium only) appears to be feasible with the completion of studies and accomplishment of technological innovations that have already been partially identified.

10.3.2 *TRANSMUTATION SCENARIOS FOR SFR SYSTEMS*

The scenario studies concern the multiple recycling either of all minor actinides or of americium only, in either homogeneous or heterogeneous mode, though heterogeneous mode can only use minor actinide-bearing blankets (CCAMs) or americium-bearing blankets (CCAms). The studies reveal that the scenarios with multiple recycling of neptunium (Np) or curium (Cm) are of little interest (negligible inventory reduction factors, curium recycling is highly problematic because of thermal and radiation protection constraints, etc.).

The transmutation of minor actinides from spent MOX-PWR fuels (reprocessing planned from 2035) should be considered in the scenarios studied, regardless of the start date of deployment of the fast reactor fleet. Gradually implementing transmutation once the fast reactors had been deployed without including the minor actinides from spent MOX-PWR fuels, which would then be destined for disposal, would minimise the inventory reduction factors for the minor actinides and make transmutation of very limited interest.

The scenarios studied should also take account of the initial stocks (minor actinide inventory in high-level and long-lived waste (HLW-LL) packages from the reprocessing of spent PWR fuels up to 2035) and of the final inventory in the cycle in 2150 (~1400 t of plutonium and minor actinides). The influence of these initial stocks on the reduction of the minor actinide inventory and the thermal load of the HLW-LL waste (factors affecting the size of the geological repository) should be examined, and the available options for managing the final inventory, and the associated timescales, should be made clear. If this final inventory were conditioned in HLW-LL waste packages destined for disposal in a geological repository, the benefits (reduction of the underground area and excavated volume of the repository, etc.) achieved by using minor actinide transmutation from 2040 to 2150 would be drastically reduced, and would be wiped out entirely if the spent fuel assemblies and minor actinide-bearing blankets unloaded during the period in question went straight into the repository.

[45] A stable situation in terms of both the type of reactors in the fleet and the inventory and isotopic characteristics of the materials present within the reactors and fuel cycle facilities.

The studies carried out show that, for all scenarios that include minor actinide transmutation, the inventory reduction factor for these actinides, including the initial and final inventories mentioned above, though significant, would remain below 3.

The general scale of the quantities of materials destined for disposal in geological repositories is therefore not substantially altered (unless the scenarios are extended beyond 2150). Thus transmutation does not appear to make a decisive contribution in terms of inventory reduction.

Moreover, according to CEA studies, it would take around two centuries to incinerate the final inventory (plutonium and minor actinides) in "burner"-type fast reactors ("phase-out" scenarios). Thus the total duration of the scenario that includes both the "phase-out" for significantly reducing the final inventory and the FR "isogenerating" phase solely for transmutation of the minor actinides could be several centuries. This would mean that a nuclear industry would have to be maintained for the whole of this period. For scenarios that do not include these two phases, the reductions to be hoped for in terms of geological repository capacity would be minimal or even non-existent.

10.4 INFLUENCE OF TRANSMUTATION ON FACILITIES AND TRANSPORT

10.4.1 *REACTOR*

Implementing transmutation does not affect overall safety in SFRs, regardless of the transmutation mode used. However, with homogeneous mode, major uncertainties due mainly to nuclear factors affect core neutron counter-reactions.

With heterogeneous mode, the high decay heat of new and used minor actinide-bearing blankets (CCAMs) means that special storage areas have to be designed for these (internal storage in the reactor or external storage in sodium or gas) before their loading in the reactor core and their unloading from the core.

10.4.2 *FUEL CYCLE FACILITIES*

Introducing minor actinides into fuels or blankets and targets significantly increases the thermal output and radiation levels of sources of neutron and gamma radiation in radioactive materials being handled during fabrication and reprocessing operations.

The industrial feasibility of fabrication operations in shielded cells has not been established (thermal constraints/size of batches of powder, throughputs, etc.).

Because of the nuclear characteristics of the radioactive materials being used, it would be necessary to upgrade the safety systems (cooling, ventilation, appropriate radiation protections).

Introducing additional, more harmful, radioactive materials into the fuel cycle means that additional separate storage for fuels and materials is needed and also brings an increase in risks, accidental situations to be considered, and source terms. In addition, the doses received by operators are potentially higher in accidental situations, and waste management is more complex.

10.4.3 *TRANSPORT*

The feasibility of transport looks in principle to have been established for homogeneous transmutation. For heterogeneous transmutation, feasibility depends on major technological innovations in the form of the development of new packaging. For some scenarios, the use of packaging concepts similar to those currently in use would lead to consignments being split, particularly in the case of new or used blankets. In this case, transport flows would increase significantly.

10.4.4 *IMPACT OF TRANSMUTATION ON GEOLOGICAL DISPOSAL*

The radiological impact of geological disposal, at a repository considered to be identical to the one in Meuse/Haute-Marne, is due mainly to “mobile” fission products and long-lived activation products, in both normal and degraded situations (denser disposal, more permeable area). However, the actinides would remain contained in the near field because of the strength of their retention in the geological medium, so the level of activity escaping from the host formation would be negligible. Transmutation of the minor actinides would therefore be unlikely to alter these conclusions.

The reduction factors of the underground footprint (2 to 4.6) and the excavated volume of the HLW-LL repository (30 to 50%) are only significant after prior storage for 120 years, which would allow the decay of the curium-244. These benefits should also be assessed in the light of the total footprint of the geological repository, the initial inventory and the option chosen for managing the final inventory. Furthermore, the overall impact on safety and radiation protection of long-term storing HLW-LL waste before disposal remains to be demonstrated (monitoring, recovery of the packages, associated transport, etc.).

However, transmutation of minor actinides or americium only would significantly reduce the time taken by the disposal cell walls to reach maximum temperature (from a few centuries to a few decades). On the one hand this would allow the first disposal modules to be monitored during their “thermal phase” while the repository was still in operation, and on the other it might reduce the length of time during which the containers would have to be leaktight to protect the glass from possible water ingress (as long as the package temperature remains above 50°C).

10.5 CONCLUSION

Transmutation of the minor actinides would require the handling of more highly radioactive materials in all the fuel cycle facilities, which would have consequences for safety and radiation protection, and would also require the development of new processes and the design of suitable new facilities and means of transport.

The feasibility of the fuel cycle processes with fissile assemblies and minor actinide-bearing blankets has not yet been established on an industrial scale, and the characteristics of the materials or items to be reprocessed or fabricated could lead to strong impositions on the design of equipment, safety systems and facilities, while increasing risks and dosimetric constraints.

The anticipated benefits in terms of reductions in the capacity and excavated volume of the geological repository are limited. In addition, the initial and final inventories are not taken into account in the scenarios studied by CEA.

The radiological impact of the repository is not altered by the deployment of minor actinide transmutation. However, the reduction in the duration of the thermal phase would improve the safety of the repository.

At present, for SFRs, the studies examined confirm that the anticipated benefits of minor actinides transmutation in terms of safety, radiation protection and waste management are not decisive, particularly in view of the constraints it places on the cycle, the reactors and transport. This assessment can also be extended to GFRs and LFRs.

In this regard, in its notice listed as reference 99, ASN states that "*potential for the separation and transmutation of long-lived radioactive elements should not constitute a decisive factor when choosing between the Generation IV technologies under examination. The different reactor technologies studied should be compared mainly based on the prospects for improving protection of the interests mentioned in Article L.593-1 of the Environment Code in relation to Generation III EPR reactors....*"

SCWRs and VHTRs, as thermal spectrum reactors, are not capable of the efficient transmutation and multiple recycling of minor actinides.

On the contrary, fast-spectrum MSRs such as the MSFR seem to offer excellent minor actinides recycling and transmutation capabilities. Besides the fact that they can operate in breeder, isogenerator and burner modes, the reprocessing units associated with these reactors do not separate actinides, which are therefore continuously recycled and transmuted. In addition, the final molten salt load remaining at the end of the reactor life can either be used as the initial load of a new reactor or can be burned in a fast-spectrum burner MSR.

11. CONCLUSION

On the basis of its examination, IRSN considers the SFR system to be the only one of the various nuclear systems considered by GIF to have reached a degree of maturity compatible with the construction of a Generation IV reactor prototype during the first half of the 21st century. Furthermore, the scenario in which PWRs would be phased out and replaced by SFRs by the end of the century is plausible given the ready supply of plutonium in the initial stages of the deployment scenario and the use of a closed nuclear fuel cycle with proven oxide fuel. Before this scenario can be rolled out, however, further progress must be made in research and technological development in areas that are already identified.

However, while it seems possible for SFR technology to guarantee a safety level equivalent to that targeted for Generation III pressurised water reactors, IRSN is unable to determine whether it could significantly exceed this level, in view of design differences and the current state of knowledge and research. The nuclear system associated with the SFR has been the focus of considerable R&D, both in France and elsewhere. One example is the ASTRID reactor project led by CEA in collaboration with AREVA and EDF. A review of the safety options report for this reactor should lead to a more accurate assessment of the various technological solutions studied in terms of feasibility and safety.

The VHTR system could bring about significant safety improvements compared with Generation III reactors, especially regarding core melt prevention, but only by significantly limiting unit power. A precise assessment of the risks relating to graphite dust would also be required if this option were selected, and it would also be necessary to develop fuels and materials capable of withstanding high temperatures; the operating temperatures currently envisaged of around 1000°C are close to the transformation temperature of materials commonly used in the nuclear industry. Moreover, this system does not optimise natural resource and waste management in the long term; storing structural waste and spent fuel is not a viable permanent solution.

There is no operating experience feedback from the other four systems studied that can be used directly. The technological difficulties involved rule out any industrial use of these systems within the time frame considered. Nevertheless, a distinction can be made between the LFR and GFR systems on the one hand, for which small reactors could be built during the first half of this century, and the MSR and SCWR systems on the other, where it seems hard to imagine any reactor being built before the end of the century.

IRSN also draws attention to the fact that the Generation IV systems selected by GIF are intended for different national contexts. The selected systems can be associated with different fuel management modes (e.g. open or closed cycles, plutonium breeding or burning) and are therefore not all suited to the energy context in France. Some criteria such as sustainable and optimised management of natural resources and waste, which are particularly associated with fast reactors, are not necessarily compatible with a significant improvement in reactor safety. This is largely because of high operating temperatures and the toxicity and corrosive nature of most coolants considered.

Regarding SFRs, and possibly GFRs and LFRs, IRSN restates its position on research into minor actinide transmutation, namely that this option offers only a very slight advantage in terms of inventory reduction and geological waste repository volume when set against the induced safety and radiation protection constraints for fuel cycle facilities, reactors and transport. On this point, ASN has recently announced that minor actinide transmutation would not be a deciding factor in the choice of a future reactor system.

IRSN considers that there is still much R&D to be done to develop the Generation IV nuclear systems; the amount depends on the system chosen. Not enough is currently known about most of the systems to judge the level of safety that could ultimately be achieved.

GIF emphasises (in reference 2) that technical development of the systems it has chosen has not progressed equally in the last 10 years, and is dependent largely on priorities set nationally. Vast resources have been committed by GIF members to developing SFR and VHTR systems, while investment in the other systems has been relatively limited. GIF believes this is mainly due to historic efforts put into developing these technologies.

It confirms the choices made in 2002 but points out that taking on board the lessons learned from the Fukushima accident raises yet more issues requiring detailed examination, particularly as regards other light water reactor concepts:

- Most of the systems studied by GIF use coolants other than water.
- The operating temperatures and power densities are higher than for PWR or BWR reactors.
- In some cases, fuel reprocessing units or facilities that present the risk of toxic releases could be built in the immediate vicinity of reactors.

Whatever the case, the assessments made will have to be reviewed once the systems have made further progress and new knowledge has been acquired. This is particularly true if the deployment of Generation IV reactors is delayed and postponed until the end of the century. Similarly, current assessments of these systems could be reconsidered due to the emergence of new, more realistic nuclear power scenarios that take into account industrial conditions and make allowance for the decommissioning of the current reactors. A well-balanced safety and radiation protection assessment of these systems is impossible at present owing to deployment time frames, which can be very different, significant gaps in degrees of maturity, and the fact that the state of knowledge varies considerably according to the system. Consequently, the indications given in the IRSN report should be viewed with caution.

Lastly, it should be borne in mind that any industrial deployment of a Generation IV reactor system in France will be linked to its advantages, not only regarding reactor fleet operation and safety, but also in terms of the coherence and performance of the associated fuel cycle. This concerns all aspects relating to safety, radiation protection, material management and efforts made to minimise the quantities of radioactive waste generated, without overlooking the overall economic competitiveness of the nuclear system. Ultimately, the choice of system must be made as part of an integrated approach, based on studies that cover multiple criteria and all the aspects mentioned above.

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APPENDIX I: ASN REQUEST

RÉPUBLIQUE FRANÇAISE

AUTORITÉ DE SÛRETÉ NUCLÉAIRE

DIRECTION DES CENTRALES NUCLEAIRES

Montrouge, le 9 décembre 2013

Réf. : CODEP-DCN-2013-064453
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Le Directeur général de l'Autorité de sûreté nucléaire

à

Monsieur le Président du Groupe permanent d'experts chargés des réacteurs nucléaires (GPR)

Objet : Saisine relative à l'examen des systèmes nucléaires de quatrième génération

Réf. : Annexe

A. Les réacteurs de quatrième génération

A.1. Le contexte international

Le développement des réacteurs de quatrième génération[1] fait l'objet de discussions entre les organismes de recherche dans le cadre de coopérations internationales au sein du GIF (« Generation IV International Forum »). En 2002, le GIF et le ministère américain de l'énergie (US DOE) ont proposé une feuille de route (référence [1]) visant à développer de nouveaux concepts d'installations nucléaires (réacteurs et installations du cycle du combustible associées), dits de quatrième génération. Pour ces organismes, les objectifs poursuivis pour ces réacteurs de quatrième génération sont les suivants :

1) développement durable :
   - limiter les énergies polluantes en favorisant des systèmes nucléaires capables d'utiliser efficacement et de façon durable le combustible nucléaire ;
   - minimiser la quantité de déchets radioactifs et réduire les inconvénients liés à leur gestion à long terme de façon à améliorer la protection des populations et de l'environnement ;
2) disponibilité et sûreté :
   - être performant en termes de disponibilité et de sûreté ;
   - avoir un très faible niveau d'endommagement du cœur du réacteur associé à une très faible probabilité d'occurrence de cet endommagement ;
   - supprimer la nécessité des actions de protection des populations à l'extérieur du site nucléaire en cas d'accident ;
3) protection physique et risque de prolifération :
   - faire en sorte que les systèmes nucléaires considérés offrent le plus de garanties possibles au regard des possibilités de détournement ou de vol de matières en vue de leur utilisation pour la fabrication d'armes nucléaires tout en améliorant la protection physique de l'installation contre les actes terroristes ;

[1] La « 4ème génération » de réacteurs constitue une nouvelle génération par rapport aux réacteurs actuellement disponibles pour un renouvellement du parc nucléaire comme les réacteurs EPR ou AP1000, nommés « 3ème génération »

4) économie :
   - montrer des avantages économiques en regard des autres sources d'énergie ;
   - présenter un risque financier comparable à ceux d'autres projets de production d'énergie.

Ce forum international s'est donné comme objectif de mutualiser les efforts de recherche et développement (R&D) et de maintenir ouvert le choix des possibilités de développement industriel parmi les six concepts de réacteurs potentiellement aptes à répondre aux objectifs indiqués ci-dessus :

1) les réacteurs à neutrons rapides refroidis au sodium (RNR-Na ou SFR),
2) les réacteurs à neutrons rapides refroidis au gaz (RNR-Gaz ou GFR),
3) les réacteurs à neutrons thermiques à très haute température (VHTR),
4) les réacteurs à neutrons rapides refroidis au plomb (RNR-Pb ou LFR),
5) les réacteurs à sels fondus (RSF ou MSR) pouvant être soit à neutrons rapides, soit à neutrons thermiques,
6) les réacteurs refroidis à l'eau supercritique (RESC ou SCWR) pouvant être soit à neutrons rapides, soit à neutrons thermiques.

A.2. Le contexte français

Dans ce contexte, les concepteurs et exploitants français (CEA, AREVA et EDF) se sont associés en constituant le « Projet RNR-Na » et se sont engagés dans des programmes de R&D, principalement sur la filière RNR-Na mais également sur la filière des réacteurs à neutrons rapides refroidis au gaz qui demande davantage d'innovations technologiques, notamment sur les matériaux et les combustibles. Le CEA mène actuellement des études de conception d'un prototype de réacteur de la filière RNR-Na dans le cadre du projet de prototype ASTRID.

Dans ce contexte, l'ASN souligne l'importance qu'elle accorde à la justification du point de vue de la sûreté et de la radioprotection du choix d'une filière par rapport aux autres retenues par le GIF. L'ASN considère en particulier que la filière qui serait retenue pour le développement d'une quatrième génération de réacteurs en France, dont le déploiement industriel se ferait au plus tôt au milieu du siècle, doit présenter un niveau de protection des intérêts mentionnés à l'article L.593-1 du code de l'environnement significativement supérieur à celui des réacteurs de génération III[2].

En outre, si l'article 3 de la loi citée en référence [2] indique que les études et recherches relatives à la séparation et à la transmutation des éléments radioactifs à vie longue doivent être conduites en relation avec celles relatives aux nouvelles générations de réacteurs nucléaires, l'ASN considère, dans son avis en référence [3], « *que les possibilités de séparation et de transmutation des éléments radioactifs à vie longue ne devraient pas constituer un critère déterminant pour le choix des technologies examinées dans le cadre de la quatrième génération. Les différentes technologies de réacteurs devraient être comparées notamment sous l'angle des perspectives de renforcement de la protection des intérêts mentionnés à l'article L. 593-1 du code de l'environnement par rapport aux réacteurs de troisième génération de type EPR.* »

*

L'ASN souhaite que, au regard des enjeux de sûreté nucléaire et de radioprotection, les éléments relatifs au panorama des différentes technologies examinées dans le cadre de la quatrième génération, référencés plus précisément ci-après, fassent l'objet d'un examen par le groupe d'experts pour les réacteurs nucléaires (GPR).

[2] répondant à des objectifs de sûreté du type de ceux retenus pour le réacteur EPR de Flamanville 3 en cours de construction

B. Cadre de la réunion du GPR

L'analyse se focalisera sur les réacteurs mais abordera également les considérations relatives aux cycles du combustible.

A titre d'information, des éléments sur les possibilités de transmutation des éléments radioactifs à vie longue dans les différents systèmes seront également présentés.

La présentation effectuée devant le GPR se basera sur :

- les documents cités en références [4] à [7] issus du projet RNR-Na,
- les rapports émis par le CEA en décembre 2012 (références [8] à [12]) au titre de la loi en référence [2],
- des documents issus de recherches bibliographiques menées par l'IRSN ou des documents produits par ou pour l'IRSN.

L'annexe n° 2 présente la liste des documents complémentaires demandés au projet RNR-Na.

B.1. Panorama des différents concepts de réacteurs

Pour ce qui concerne les concepts de réacteurs, je souhaite recueillir l'avis du GPR sur chacun des concepts de réacteurs, tels qu'ils pourraient être construits vers 2050, notamment sur les points suivants :

- les caractéristiques générales en matière de sûreté nucléaire et de radioprotection des différents concepts,
- la maturité des concepts et les besoins de R&D vis-à-vis de la sûreté nucléaire et de la radioprotection,
- les risques spécifiques associés,
- les principales séquences accidentelles,
- les éléments de retour d'expérience éventuellement disponibles,
- les points durs et les éventuels verrous technologiques qui seraient à lever avant d'envisager la construction d'un prototype ou d'un réacteur industriel.

Un classement des concepts n'est pas *a priori* recherché.

B.2. Aspects liés aux cycles du combustible

Pour ce qui concerne les aspects liés aux cycles du combustible, je souhaite recueillir l'avis du GPR sur :

- les différentes options de cycles du combustible envisageables par système (cycle ouvert, cycle fermé, cycles symbiotiques, conditions de déploiement,...), en particulier celles prises en compte dans les études de scénarios technico-économiques menées par le CEA dans le cadre de la loi en référence [2] (scénarios auxquels est notamment associé le déploiement d'un parc de réacteurs RNR-Na),
- l'état de la R&D sur les procédés associés aux opérations du cycle du combustible (état de maturité, verrous technologiques, enjeux de sûreté et de radioprotection déjà identifiés, ...),
- la gestion des matières et des déchets produits lors des opérations du cycle (inventaires/flux, caractéristiques, besoins d'entreposages, impact sur les installations du cycle et le stockage, ...).

B.3. Transmutation des éléments radioactifs à vie longue

Pour ce qui concerne les options de transmutation des éléments radioactifs à vie longue, le GPR prendra connaissance de l'avis cité en référence [13] relatif à l'évaluation, sur le plan de la sûreté, de la

radioprotection et de la gestion des matières et des déchets, de la faisabilité et de l'intérêt de la mise en œuvre, à l'échelle industrielle, de ces différentes options.

*
* *

En conclusion, l'ASN souhaite recueillir, pour la mi-2014, l'avis du GPR sur les systèmes nucléaires de 4ème génération sélectionnés par le GIF, principalement sous l'angle du renforcement de la protection des intérêts mentionnés à l'article L. 593-1 du code de l'environnement par rapport aux réacteurs de troisième génération de type EPR.

L'ASN vous demande de bien vouloir associer quelques membres du groupe permanent d'experts en charge des usines et de celui en charge des déchets aux travaux du GPR.

La présente saisine annule et remplace la précédente saisine en date du 12 octobre 2011 (référence [14]). En effet, cette dernière, qui portait sur l'examen des filières de quatrième génération et le retour d'expérience des réacteurs rapides à caloporteur sodium (RNR-Na), est remplacée d'une part par la saisine en date du 22 octobre 2012 (référence [15]) relative au retour d'expérience des réacteurs rapides à caloporteur sodium et au dossier d'orientation de sûreté du réacteur ASTRID (objet de la réunion du GPR du 27 juin 2013) et, d'autre part, par la présente saisine pour la partie relative à l'examen des filières de quatrième génération.

Le Directeur général de l'ASN,

Jean-Christophe NIEL

APPENDIX II: INFORMATION ON OVERALL FEEDBACK COEFFICIENTS

Three overall feedback coefficients are defined, denoted k, g and h, such that

$$\delta\rho = k.\delta Te + g.\delta\Delta Tc + h.\delta P$$

where

ρ: core reactivity (in pcm)

Te: coolant temperature at the core inlet (°C)

ΔTc: coolant temperature increase in the core (°C)

P: core thermal power (in MW or as a percentage of the nominal thermal power)

δX represents the variation of the variable X between two stable reactor states, with each state defined by the set (Te, ΔTc, P).

For example, k represents the change in reactivity caused by a 1°C increase of the coolant temperature at core inlet, with ΔTc and P kept constant.

For SFR, LFR and GFR fast reactors, the coefficients k, g and h are determined by taking account of the different types of temperature feedback affecting reactivity, which can be grouped into three categories:

a) the Doppler effect;
b) local effects linked to the way the concentrations per unit volume of the core components (coolant, fuel, cladding, etc.) change with the temperature;
c) general effects:

- the expansion of the diagrid, which aims to keep apart the assemblies it is supporting;
- the effect of differential expansion between the control rods, the core and the vessel, which causes relative displacement between the fissile zone and the absorber zone.

When the overall coefficient is negative, it is working against what caused it.

If the overall feedback coefficients k, g and h are all negative in normal operation and under incident conditions, the reactor can be operated in a stable way. During an unprotected transient from a stable state characterised by the set (Te1, ΔTc1, P1), the reactor will tend towards another stable state characterised by a different set, (Te2, ΔTc2, P2).

APPENDIX III: THE DIFFERENT TYPES OF NUCLEAR FUEL ENVISAGED FOR GEN IV REACTORS

Various different types of fuel can be envisaged for the Generation IV nuclear systems; some types can be used by several different systems.

The physical characteristics of the fuel and its chemical composition are important factors in safety analyses. In particular, may or may not lead to core melt depending on the nature of the fuel.

Reactions between some fuels and coolants are also possible.

This Appendix only provides information about the behaviour of the fuel in a reactor. Fuel cycle-related issues (fabrication and reprocessing) are discussed in Chapter 9).

1. FISSILE AND FERTILE MATERIALS IN FUEL

1.1 URANIUM

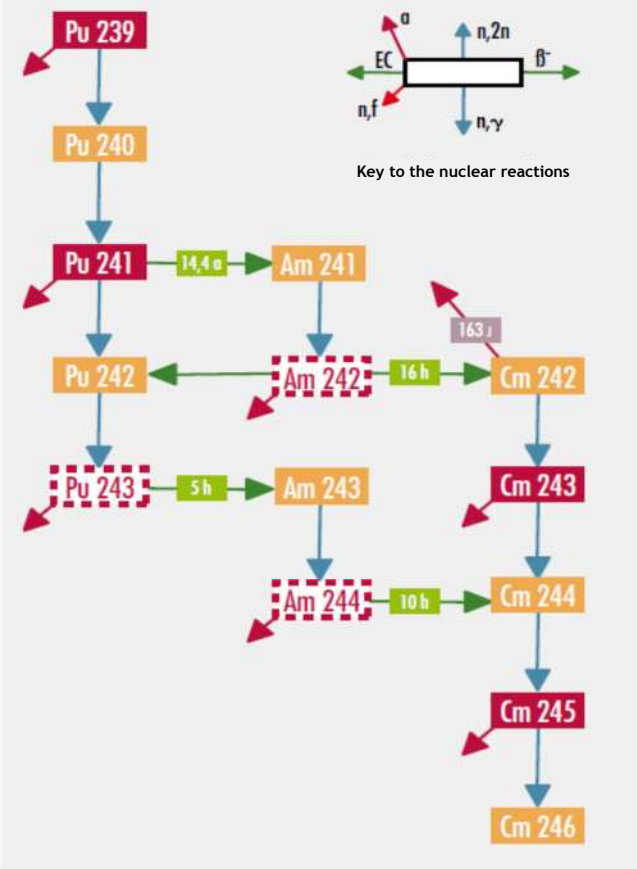
Natural uranium contains only 0.7% uranium-235 (^{235}U), which is the only natural isotope that is fissile regardless of neutron energy. Natural uranium has been used in GCRs (Generation I reactors built in France) and CANDU (CANadian Deuterium Uranium) reactors developed in Canada.

Through the effect of the neutron flux and radioactive decay, the isotope uranium-238 (^{238}U), which makes up 99.3% of natural uranium, is a fertile isotope[46] which converts to neptunium and then mostly to the isotope plutonium-239 (^{239}Pu). Plutonium-239 is fissile regardless of neutron energy. Neutron bombardment also produces other plutonium isotopes, some of which are fissile regardless of neutron energy and others only fissile in a fast spectrum. The different nuclear reactions cause the destruction of fissile material in the form of uranium-235 but also generate fissile material by producing plutonium-239.

$$^{238}\mathrm{U} + n \rightarrow {}^{239}\mathrm{U} \xrightarrow[23.5\ \mathrm{m}]{\beta^-} {}^{239}\mathrm{Np} \xrightarrow[2.355\ \mathrm{d}]{\beta^-} {}^{239}\mathrm{Pu}$$

Equation 1: Conversion of ^{238}U into ^{239}Pu

[46] ^{238}U is fissile in a fast neutron spectrum but with a low fission yield.



Equation 2: Radioactive decay of ^{239}Pu

Uranium enriched with uranium-235 is still used in some SFRs such as those built in the former Soviet Union (BOR 60, BN 350 and BN 600) and the Chinese CEFR reactor. This fuel is chosen when there is no fuel fabrication and reprocessing facility using plutonium, as discussed in the next Section.

While thermal reactors require less than 5% uranium-235 enrichment, fast reactors such as the SFR require enrichment greater than 20% because the effective fission cross-sections are smaller than with a thermal spectrum.

1.2 URANIUM-PLUTONIUM FUELS

Because plutonium has better fission characteristics than uranium-235 in a fast spectrum, a combination of uranium and plutonium has been used in most SFRs built so far (except for the ones listed in the previous Section).

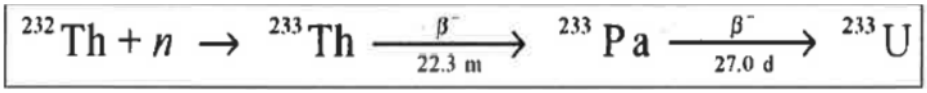
The plutonium is combined with depleted uranium from uranium enrichment plants, which contains around 99.8% uranium-238.

Depending on the relative quantities of uranium-238 and plutonium, the uranium-238 can be converted to plutonium in a fast spectrum, with more plutonium being produced than destroyed by the reaction. This characteristic, which is known as breeding, has been advocated since the start of the civil nuclear era. The possibility remains limited to fast-spectrum reactors.

1.3 URANIUM-THORIUM FUELS

The possibility of uranium becoming scarcer as a result of large numbers of reactors being built prompted designers to take an interest in thorium. Thorium resources are estimated to be three or four times those of uranium (according to reference 22).

Unlike uranium, natural thorium contains only one stable isotope with an atomic mass of 232 (^{232}Th), which is not fissile. However, thorium-232 in a neutron flux converts to uranium-233, which is not a natural element but is fissile.



Equation 3: Conversion of thorium-232 to protactinium-233 then to uranium-233

This combination (^{233}U,^{232}Th) has the peculiarity of being able to breed in a thermal spectrum as well as a fast spectrum, and breeding has been achieved in the experimental LWR at Shippingport in the United States. Preliminary studies have also shown that the breeding factor could be even greater with heavy water cooled reactors. Finally, another peculiarity of the (^{233}U,^{232}Th) combination is that it produces fewer minor actinides.

To start a reactor designed to use a (^{233}U,^{232}Th) fuel, the first cores have to use (^{235}U,^{232}Th) or (^{239}Pu,^{232}Th) pairings to create a sufficient quantity of uranium-233 to feed subsequent cores.

When irradiated, uranium-233 produces uranium-232, which has a half-life of 69 years in a decay chain that includes thallium-208 and polonium-232, both of which are high-energy γ emitters. This complicates the fabrication and reprocessing of the fuel, but makes it relatively non-proliferating.

(^{233}U,^{232}Th) fuel is currently the standard fuel for the MSFR molten salt reactor project described in Chapter 6.

2. PHYSICAL STATE OF THE FUEL

The nuclear fuel to be used in the reactors proposed by GIF can take different physical forms, described below.

2.1 PELLETS

In SFR, LFR, GFR and SCWR reactors, the fuel is made up of pellets containing the fissile and fertile materials. These materials are chemical compounds (see paragraph 3 of this Appendix). The pellets are stacked in metal or ceramic cladding, which is the first barrier for the radioactive products.

2.2 PARTICLES

For VHTRs, fuel is in the form of millimetre-sized spherical particles consisting of a central kernel containing the fissile and fertile materials. For the first HTRs, the kernel was surrounded by two layers of pyrolytic carbon, forming BISO (BIstructural ISOtropic) particles. Then a layer of silicon carbide (SiC) was added to make TRISO (TRistructural ISOtropic) particles.

This fuel is specific to VHTRs (see Chapter 3).

2.3 LIQUID FUEL

The MSR concept can use fuel of a totally different design from that chosen for other concepts. The fuel is in the form of a mixture of salts in a liquid state, e.g. lithium fluoride and fluorides of actinides (such as uranium and thorium).

In this configuration, the fuel is inseparable from the coolant and the nuclear power is generated directly within the coolant.

The peculiarities of this fuel and their impact on controlling the safety functions are discussed in the description of the MSR concept in Chapter 6.

3. CHEMICAL COMPOUNDS IN FUEL

3.1 OXIDE FUEL

Use of the different oxide fuels

Oxide fuels can use only uranium (UO_2) or a combination of uranium and plutonium in the compound (U,Pu)O2 (MOX or Mixed OXide fuel).

The former Soviet Union developed UO_2 on an industrial scale, but it is mainly MOX fuel that has been used and is being considered in pellet form for the SFR, LFR and SCWR systems.

Some characteristics of MOX fuel

Compared with the MOX fuel used in PWRs, MOX fuel for SFRs initially contains a larger quantity of plutonium (between 15% and 25%).

One characteristic of MOX fuel is that its melting temperature is high, but this decreases as the plutonium concentration increases (2850°C with 0% plutonium and 2428°C with 100% plutonium). Its thermal conductivity is low, producing steep temperature gradients in the fuel.

Using MOX fuel can, under certain conditions, lead to corrosion of the internal face of the cladding because of the migration of oxygen and some fission products around the pellets outside surface and their reaction with some of the compounds in the cladding. This phenomenon increases with the burn-up rate. For high burn-up rates, CEA recommends the use of ferro-martensitic steels, which should limit the fission product reactions with some of the compounds in the cladding and thus reduce this internal corrosion.

The temperature of the fuel in SFRs is high, which can lead to fission gas releases. The higher the burn-up rate, the bigger these releases. These gases will go into in the plenum of the fuel pins, which should be designed to cope with the end-of-life fission gas pressure so as to provide the first barrier for the fission products.

In the event of a core melt, MOX fuel in a solid or liquid state reacts chemically with the sodium (this point is discussed further in paragraph 2.2.5.1 in the body of this report).

Experience with irradiation

MOX fuel has been used mainly for the SFRs built in France, the United Kingdom and Japan. There is more operational experience with this type of fuel than with any other. CEA is planning to use it in the ASTRID SFR. Although it developed carbide fuel in the FBTR (Fast Breeder Test Reactor), India plans to operate the PFBR (Prototype Fast Breeder Reactor) currently under construction with a MOX fuel.

Significant operational experience has been gained of the behaviour of MOX fuel under irradiation: some 430,000 pins have been irradiated in French SFRs and in PFRs in the UK (according to reference 23) and 50,000 pins have been irradiated in Japan (according to reference 21) with burn-up rates of up to 150,000 MWd/tonne of oxide. India has also irradiated MOX fuel pins in the FBTR at up to 112,000 MWd/tonne. The linear power densities are around 400-450 W/cm.

Reference 21 states that burn-up rates of 200,000 MWd/tonne of heavy metal could be envisaged. It should be pointed out that the record burn-up rate achieved with Phenix was 150,000 MWd/tonne with the experimental BOITIX 9 assembly and that the limits for burn-up rate are intimately linked with damage to the fuel element cladding, expressed as the number of displacements per atom (dpa).

Fuels using thorium

The pairing of uranium and thorium has so far been studied and tested as an oxide compound in the form of pellets, particularly in heavy water reactors, and as TRISO particles in the HTR reactors built in Germany.

3.2 CARBIDE FUEL

Use with the different concepts

The use of carbide fuel in the form of pellets stacked inside the fuel pin cladding is being considered for SFR, LFR and GFR systems.

Some characteristics of carbide fuel

Carbide fuel, which is sometimes referred to with the abbreviation MC (Mixed uranium-plutonium Carbide) and corresponds to (Ux,Pu1-x)C, has been studied in a number of countries but has been used mainly in India in the FBTR reactor since its commissioning in 1985.

The characteristics of this type of fuel are:

- a higher density than MOX fuel, which limits the volume of plutonium and uranium in the core;
- a lower melting temperature than MOX fuel, with the temperature heavily dependent on the relative quantities of U and Pu (but typically 2500°C for UC and 1600°C for PuC);
- high thermal conductivity (higher by a factor of 5 to 10 than MOX fuel depending on the relative quantities of U and Pu).

Carbide fuel swells more under irradiation than oxide fuel and therefore an initial large pellet-to-clad gap is necessary.

When used in SFRs, carbide fuel does not react with the sodium, which theoretically means that the pellet-to-clad gap can be filled with sodium to increase heat transfer between the fuel and the cladding, and therefore the permissible linear power density (in reference 21, a limit of 1000 W/cm is mentioned for linear power density). But in practice, a gas-filled gap has been chosen because of the increased risk of cladding embrittlement due to carburisation (the carbon in the fuel spreading into the cladding) and the risks associated with the presence of a sodium bond during reprocessing operations. Consequently, the linear power density is limited to around 600 W/cm.

If this type of fuel is used in SFRs, the problem arises of the interaction between the molten fuel and the sodium: the high thermal conductivity of the fuel can cause larger amounts of energy to be transferred to the coolant than with oxide fuels, which could cause a reaction with an intense energy release.

Experience with irradiation

Operational experience with irradiation has been gained mainly with the Indian FBTR of which the original core, known as Mark I, consisted of assemblies containing 70% PuC and 30% UC, with a burn-up rate of 165,000 MWd/tonne and a linear power density of 400 W/cm. Moreover, this type of fuel has been reprocessed and fuel assemblies have been fabricated with the plutonium from the reprocessing and inserted into the reactor.

In the US, 700 fuel pins with a helium or gas bond were irradiated in EBR II and FFTF at burn-up rates of between 100,000 and 200,000 MWd/tonne.

In Phenix, the two irradiations of carbide fuel, SAUTERNES and NIMPHE 2 were intended for SFRs, while the FUTURIX-MI and FUTURIX-Concept irradiations, based on a fuel design with honeycomb plates now abandoned by CEA, were intended for GFRs.

3.3 NITRIDE FUEL

Use with the different concepts

The use of nitride fuel in the form of pellets stacked inside the fuel pin cladding is being considered for LFR and also SFR systems.

Some characteristics of nitride fuel

Nitride fuel is sometimes referred to with the abbreviation MN (Mixed uranium-plutonium Nitride) and corresponds to (Ux,Pu1-x)N).

This fuel combines two interesting physical properties: a high melting temperature similar to that of MOX fuel and high thermal conductivity comparable with that of MC. These characteristics make reactors using this type of fuel very tolerant of unprotected transients such as inadvertent control tod withdrawal, loss of flow or loss of heat sink, because they reduce the risk of the fuel melting. This fuel is being promoted by the Russian designers of the LFR.

Through a similar phenomenon to carburisation, nitride fuel can cause the embrittlement of the cladding by nitriding, restricting the potential in the case of SFRs for using a sodium bond.

The main problem with MN is the large neutron capture cross-section of nitrogen-14 (14N), which has major consequences for the neutron balance. Moreover, neutron capture by nitrogen-14 also produces carbon-14 (14C). The solution envisaged in Russia is the use of nitrogen highly enriched with nitrogen-15 and the recovery of the carbon-14 during the reprocessing operations.

Experience with irradiation

Experimental irradiations have been carried out on MN, particularly at the Russian BR-10 and BOR-60 reactors, though with linear power densities of up to 1500 W/cm, and also at FFTF and EBR II in the US. Both helium bond and sodium bond concepts have been irradiated.

Various irradiations have taken place in Phenix:

- NIMPHE 1, 1 bis and for SFRs;
- FUTURIX-MI, FUTURIX-Concept and FUTURIX-FTA nitride for GFRs but using fuel in honeycomb plates, a configuration described for carbide fuels that has now been abandoned by CEA.

3.4 METAL FUEL

Use with the different concepts

Metal fuel has mainly been studied in the US since the SFR concept was first developed, because of its high proportion of heavy atoms (plutonium and uranium). India and Korea also plan to use this type of fuel in their future SFRs.

Some characteristics of metal fuel

Irradiation causes metal fuel to swell significantly, so a large pellet-to-clad gap is necessary, and causes the formation with the cladding of eutectics having a low melting point. Adding zirconium can limit the formation of these eutectics, so the fuel is generally a ternary mixture of uranium, plutonium and zirconium.

The melting temperature is relatively low (around 900 to 1000°C), while thermal conductivity is comparable with that of MC and MN.

Experience with irradiation

Several irradiations have taken place in EBR-II and FFTF with burn-up rates of up to 20%. This type of fuel has also been tested during tests of loss of flow in the core without rod drop, carried out in the EBR-II reactor.