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GAS-COOLED FAST REACTOR SYSTEM SAFETY ASSESSMENT

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CONTENT

LIST OF ACRONYMS AND ABBREVIATIONS

AOO	Anticipated operational occurrences
CEA	COMMISSARIAT À L'ENERGIE ATOMIQUE ET AUX ENERGIES ALTERNATIVES
DBA	Design Basis Accident
DEC	Design Extension Condition
DHR	Decay Heat Removal
DiD	Defence in Depth
ESFAS	Engineered Safety Features Actuation System
GFR	Gas-cooled Fast Reactor
GoFastR	European Gas Cooled Fast Reactor Collaborative Project
IAEA	International Atomic Energy Agency
LOCA	Loss of Coolant Accident
LOFA	Loss of Flow Accident
MHX	Main Heat Exchanger
MIV	Main isolation valve
MOX	Mixed Oxide
SA	Severe Accident
UJD	Nuclear Regulatory Authority of the Slovak Republic (Slovak title "Úrad Jadrového Dozoru")

1 GENERAL OVERVIEW OF THE PERFORMANCE GOALS

The Gas-cooled Fast Reactor (GFR) system features a high temperature helium cooled fast spectrum reactor that can be part of a closed fuel cycle. The GIF Technology Roadmap [1] identified the GFR as a technology that associates the advantages of fast spectrum systems (in terms of sustainability with improved resource utilization and waste minimization through reprocessing) with those of the high temperature (in terms of high thermal cycle efficiency, co-generation of hydrogen, and other industrial uses).

The GFR cooled by helium is proposed as a longer-term alternative to liquid-metal cooled fast reactors. This type of innovative nuclear system has several attractive features: The helium is a single phase, chemically inert, and transparent coolant. It does not dissociate or become activated, and while the coolant void coefficient is generally positive, it is a comparatively small effect and dominated by Doppler and other feedback mechanisms. The high core outlet temperature above 750°C (typically 800-850°C) is an added value of GFR technology.

The reference GIF concept for GFR is a 2400 MWth plant operating with a core outlet temperature of 850°C enabling an indirect combined gas-steam cycle to be driven via three intermediate heat exchangers. The high core outlet temperature places onerous demands on the capability of the fuel to operate continuously with the high power density necessary for good neutron economy in a fast reactor core. This represents the biggest challenge in the development of the GFR system. The second significant challenge for GFR is ensuring decay heat removal in all anticipated operational and fault conditions.

A necessary step in the development of a commercial GFR is the establishment of an experimental demonstration reactor for qualification of the refractory fuel elements and for a full-scale demonstration of the GFR-specific safety systems. The proposed demonstration reactor for the reference GIF GFR concept will be ALLEGRO: A low power reactor (50 - 150 MWth) with the ability to operate in different core configurations starting from a “conventional” core featuring steel-cladded MOX fuelled pins and followed with all-ceramic fuel elements in the latter stages of operation.

2 HISTORICAL REVIEW

US

In 1962, General Atomics designed a prototype reactor with a rated power of 300 MWe as well as a 1000 MWe industrial-scale reactor [7]. The detailed design and certification programme for the 300 MWe prototype reactor 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.

Europe

In Europe [10] a number of players in the nuclear field joined forces to establish the Gas Breeder Reactor Association for development of a gas-cooled fast reactor. This group proposed a first design (GBR-1) in 1970, a 1000 MWe reactor featuring helium coolant, pin-type fuel, conventional outlet temperature, and a secondary steam cycle. This design was followed by GBR-2 and -3 (1971), also 1000 MWe reactors but using coated particle fuel, slightly elevated outlet temperature, and helium coolant for GBR-2; CO2 coolant for GBR-3 [13] . The 3 designs finally evolved into the GBR-4 design, a 1200MWe reactor with helium cooling and pin-type fuel.

The GBR-2 design is interesting because it resurfaces in modern design proposals for GCFRs, for instance, in Japan [11]. The objective of the coated particle fuels was to increase the core outlet temperature to improve the thermodynamic efficiency of the secondary steam cycle. For both GBR-2 and -3 coated particles were only used for the driver fuel, the blankets employed traditional pin type fuel. This solution was chosen because, at the time of design, reprocessing of coated particle fuel was not proven. GBR-2 and -3 required several ceramic parts, most notably the structures at the outlet side. The fabrication difficulties related to large ceramic parts led to the development of GBR-4, which is a much more conventional design. In GBR-4, the outlet temperatures are decreased, enabling the use of stainless steel components throughout the core. The plant efficiency is lower, which is offset by a larger total output of the reactor: from 1000 MW_e to 1200 MW_e. A last reference to the GBR-4 design was found in [12], where the safety case for large GCFR cores is discussed.

UK

In the late 1970s, a UK program [10] was initiated into an “Existing Technology Gas Breeder Reactor” (ETGBR). This design focused on joining the experience gained in the UK on sodium systems (PFR, Dounreay) and the thermal CO_2-cooled AGR reactors. The fuel assemblies used stainless steel cladding with surface roughening, while the entire system was to be housed in a concrete vessel as used for the AGRs. ETGBR also used CO_2 coolant, and had a lower power density than LMFBRs, with the expected higher breeding gain to make up for the difference [14]. The ETGBR was not very different from other designs of the same era for GCFRs. However, the ETGBR idea lingered on for a long time well into the late 1990s. At that late stage, the ETGBR was rebranded as the Enhanced Gas-Cooled Reactor (EGCR). EGCR was proposed as an actinide burner, first within the European Fast Reactor (EFR) program, and later in the CAPRA/CADRA study [15]. By then, the reactor featured 3600MWth, CO_2 cooling, and nitride fuel in fuel pins.

Japan

In Japan [10] a fast reactor programme was initiated in the 1960s, including sodium and gas-cooled reactor concepts. Kawasaki Heavy Industries (KHI) investigated GCFR concepts cooled with steam, CO2, and helium [16]. The helium concept was based on LMFBR technology, but KHI opted for a very low core, to reduce the pumping power requirements. The flat core also increases breeding gain but requires a larger fissile fraction. Investigations into the GCFR concept seem to have continued without interruption in Japan, culminating in the late 1990s in a GCFR design proposal by JNC. This reactor also features a core with a low height/diameter ratio “pancake core”, and uses coated particle fuel. A nitride fuel compound is chosen for the kernels. Buffer layer and sealing layers are made of TiN. Two types of fuel assemblies are proposed. One fuel assembly resembles that of GBR-2: coated particles are arranged in an annular bed, with the helium flowing radially through the bed. The other design features large prismatic blocks filled with a mixture of coated particles and a matrix material (TiN, SiC or ZrC). Coolant channels run axially through the blocks. All structural parts are made in SiC. Thermal output is 2400MWth, with a power density of 100MW/m^3. The coolant is helium and a direct cycle energy conversion system is envisaged [11].

3 LEVEL OF ONGOING SAFETY- AND SECURITY- RELATED R&D

To achieve the highest level of safety in the design of a nuclear power plant, following measures consistent with national acceptance criteria and safety objectives [17] are required:

- To prevent accidents with harmful consequences resulting in a loss of control over the reactor core or over other sources of radiation, and to mitigate the consequences of any accidents that do occur;
- To ensure that for all accidents taken into account in the design of the installation, any radiological consequences would be below the relevant limits and would be kept as low as reasonably achievable;
- To ensure that the likelihood of occurrence of an accident with serious radiological consequences is extremely low and that the radiological consequences of such an accident would be mitigated to the fullest extent practicable.

The IAEA fundamental safety objectives, safety principles, and design safety requirements, including the application of the concept of Defence-in-Depth (DiD), form the basis of GFR design development. The safety principles can also be extended to address the security of facilities by application of measures that contribute to both safety and security, such as [18]:

- Appropriate provisions in the design and construction of nuclear installations and other facilities;
- Controls on access to nuclear installations and other facilities to prevent the loss of, and the unauthorized removal, possession, transfer and use of, radioactive material;
- Arrangements for mitigating the consequences of accidents and failures, which also facilitate measures for dealing with breaches in security that give rise to radiation risks;
- Measures for the security of the management of radioactive sources and radioactive material.

For GFR and other innovative reactors, there is an aim to maximise the use of inherent safety characteristics and passive systems/components. Passive systems are expected (but not guaranteed) to be highly reliable, with their successful operation determined by the phenomenological response of the plant to the initiating event. It is generally to be expected that the risk profile for Gen IV plants will be significantly different when compared to the existing NPPs and initiating events of very low probability may dominate the safety assessment.

4 ACHIEVEMENT OF FUNDAMENTAL SAFETY FUNCTION

A safety function is a specific purpose that must be accomplished for safety. In design of a nuclear power plant, following three fundamental safety functions are considered:

- Control of reactivity;
- Removal of heat from the reactor and from the stored fuel;
- Confinement of radioactive materials, shielding against radiation, limiting accidental radioactive releases.

4.1 Reactivity control

The basic requirement of any shutdown strategy is to safely shut down the reactor and maintain safe shutdown conditions during normal operational states and and after fault conditions. Although the use of innovative concepts for self-actuating, passive shutdown devices are also considered, more traditional methods for achieving shutdown must form the basis of the shutdown strategy for GFR concepts. Issues to be considered in the choice of shutdown devices include effectiveness, performance, reliability, and practicality (e.g. space requirements, ability to integrate into the overall design).

Two redundant trains of shutdown systems are foreseen, i.e. control system and shutdown device that consist of two independent sub-systems each connected to a dedicated I&C support system.

One of two different shutdown systems serves for quick rendering the reactor subcritical from operational states and in DBA conditions, and another one for rendering the reactor subcritical also for most reactive conditions.

It is under discussion whether a third shutdown system based on absolutely different physical principles will be applied. In the large commercial GFR design studies, for example, the use of a pneumatically operated Diversified Shutdown Device system is considered.

The fast spectrum design of the GFR offers an opportunity for enhanced reactivity feedback that, together with a refractory fuel, would offer promising prospects for surviving the anticipated transients without scram without severe core damage. The search for self-generating core (breeding ratio > 1, production of fissile material > consumption) will limit the core excess of reactivity. Nevertheless, specific R&D should be devoted to innovative and possibly passive shutdown systems. Design work will focus on features to limit the risk of re-criticality to a level such that it will could be considered as practically eliminated (prevention of severe situations achieved by robust safety design).

4.1.1 Control system

A minimum of two diverse, independent, reliable and fast-acting systems will be provided to perform the shutdown function. The expected reliability of each shutdown system is of key importance and should, as a minimum, be such that unavailability of both systems will be limited to less than 10^{-6} / demand.

Availability of any support system necessary for actuation of the shutdown system will assure that the prescribed limits above will not be exceeded.

Each shutdown system will be required have sufficient redundancy so that the system will still meet the minimum requirements for negative reactivity in the case of failure of the most effective single absorber rod (SFC). Although one of the shutdown systems may also provide a reactivity control function, this should be in addition to the shutdown function and will not jeopardise the system’s capability to shut down the reactor.

The reactivity worth of each shutdown system will be sufficient for all operational states and postulated accident conditions, including postulated fuel handling errors to:

- achieve and maintain the reactor in a sub-critical configuration when required, and
- ensure that the design limits for fuel, cladding and structures are not exceeded.

The maximum reactivity worth of any single absorber rod should be limited such that accidental withdrawal of the rod would not result in exceeding any fuel, cladding or structures design limits.

Automatic activation of the shutdown systems will be by independent and diverse means, each with internal redundancy. Manual activation will also available from the main and emergency control rooms. To ensure high reliability, the automatic means of shutdown will ensure that no operator action is needed during the first 30 minutes following the fault detection.

The design of the absorber rods and drive mechanisms will aim to prohibit any accidental increase in post-shutdown reactivity due to unintentional movement or accidental ejection of rods. This could be achieved by, for example, gravity and/or hold-down latches.

The lack of reactor thermal inertia for specific faults involving loss of heat removal means that any shutdown system must have a fast response from receipt of shutdown signal to successful initiation and operation.

Diversity is required in the signal being monitored, in the signal processing equipment, in the electrical isolation system, in the rod insertion mechanisms, and in the rod design itself.

Generally, the initiating faults that have the potential to result in core damage must be protected by two physically different parameters connected to two diverse trip systems. The reactivity control is handled by two independent Control Rod Assemblies groups: Control and Shutdown Devices (CSD assemblies) and Diverse Shutdown Devices (DSD. The DSD rod design should, for example, provide larger clearance to enable their insertion into distorted core channels.

There should also be two diverse means of supporting the absorber rods and diversity in the provision of mechanisms for rod insertion, for example different release mechanisms for gravitational drop and motorised follow up to ensure insertion is achieved.

Each shutdown system must be able to shut down the reactor and maintain it in the cold shutdown state, and must be able to achieve this with a single failure.

Multiple sensors to detect respective different physical parameters must be provided to ensure redundancy.

Shutdown system can achieve the cold shutdown state with failure to insert a single rod (assumed the one with the highest worth).

There should be independence between the redundant trains of the shutdown systems provided, by means of spatial separation and protection by walls or barriers. This independence is required to protect control system against the adverse consequences of internal and external hazards.

Tertiary shutdown system should ensure core subcriticality in the event of the failure to insert both CSD and DSD absorber rods. It should be a passive system based on physical laws.

In accidental conditions, as for all fast neutrons systems, core compaction will increase the total core reactivity. This phenomena should be limited or avoided by design. Nevertheless, related to a SFR core, the quantitative effect should be lower because of the lower impact of the coolant on neutron absortions. GFR cores with SiC fuel structures, should have improved resistance to geometrical deformation compared to steel structures.

4.1.2 Risk of re-criticality

The risk of re-criticality is considered to be diminished by inherent characteristics of gas and mitigated by GFR design provisions and especially by its neutronic core design. Design work will focus on features to limit the risk for re-criticality so that it will have a very low frequency of occurrence and could be considered as practically eliminated from the design. A passively cooled core catcher maintains sub-criticality, when the core looses its geometry and core compaction takes place.

4.2 Decay heat removal

The basic requirement of any decay heat removal (DHR) strategy is to remove residual heat from the core after reactor shutdown, during operational states and after fault conditions. In a GFR, a number of different systems may perform a DHR function during different operating conditions. Their role and the requirement for their availability may depend on the status of the reactor.

The following general principles for the design and implementation of DHR systems for GFR are assumed:

- The expected reliability of each system providing the DHR function should be such that the frequency target for the total loss of DHR function should be less than 10^{-7} per year. This is consistent with EFR and is proposed for GFR.
- In order to ensure a reliability of DHR system, there should be adequate redundancy, diversity and independence arranged by provision of sufficient systems to provide the DHR function and within these systems themselves. There should be adequate segregation and separation of DHR systems and within these systems (e.g. between cooling loops within a system).
- The need for supporting systems, such as power supplies and ventilation systems, should be minimised and, when required, these systems should meet similar criteria for redundancy, diversity and independence.
- Activation of systems providing the DHR function should be ensured by independent, diverse automatic means (each with internal redundancy). To ensure high reliability, the automatic means of activation should ensure that no operator action is needed during the first 30 minutes following the fault detection, although operator action can also be considered as a diverse means of activation during this time.

Core coolability and a lack of thermal inertia is a main issue for a helium-cooled fast reactor and this must be taken into account both in the design process and in the safety analysis.

The coolability of the core should be assured in any adverse situation combining pressurized and forced flow conditions or depressurized and natural flow conditions. The whole primary system is bounded in a metallic leak-tight vessel (guard vessel) to maintain the coolant pressure in the primary system. If a loss of coolant pressure is combined with an additional failure (e.g. loss of forced circulation due to total loss of power supply), the dedicated independent safety system increasing coolant density shall be actuated (e.g. nitrogen (N_2) injection system).

Within a facility, failure of a number of safety devices/components to perform their functions may occur as a result of a single specific cause. Consideration and prevention of such Common Cause Failures, by provision of DHR systems with adequate diversity and independence, are essential to ensure adequate reliability levels. Where redundant systems/components are at potential risk from common cause failures, one possible means of reducing the susceptibility of the design to such effects is to employ diverse provisions in separate redundant trains or systems. To achieve the required levels of reliability for the DHR function (frequency of total loss of DHR function < 10^{-7} per year), there should be an adequate diversity strategy for the GFR. The strategy should ensure that the GFR design will be able to maintain the DHR safety function when required, for all operational and accidental states. The general goal should be to achieve an appropriate level of diversity to enable the exclusion of common cause failures from design basis considerations. In particular, the issue of system/component unavailability during maintenance and repair should be considered.

The safety function of the decay heat removal (DHR) system shall be to transfer fission product decay heat and other residual heat from the reactor core. The rate of decay heat removal has to be arranged in such a way that both fuel design limits and the design basis limits of the reactor coolant pressure boundary specified in the design are not exceeded.

The DHR function is ensured by the circulation of helium gas contained in the primary circuit. Depending on the situation encountered, the connected circuits are normal primary circuits or dedicated DHR loops.

Under normal conditions after the reactor shutdown, the decay heat is removed through the normal primary and secondary circuits. Dedicated DHR loops are connected to the primary circuit operating in forced or natural circulation. These loops are put into operation after isolation of the main loops by (actuation of isolation devices) and residual heat is removed through dedicated DHR loops in forced of natural convection regime.

With a primary circuit pressurized at 70 bar, one of the typical accidents to consider is the primary depressurization. The low Helium density is not a favourable element for the natural circulation decay heat removal. Conversely, the Helium void reactivity effect is naturally very low (less than 1$) and this is a favourable element for unprotected situations.

On the other side, the GFR power density of about 100MW/m3 is significantly larger than the HTR values (4 to 6 MW/m3), the latter including advantageously graphite blocks (thermal inertia and conduction heat transfer capacity). The consequence for the GFR DHR strategy is:

- Need to develop a fuel element able to resist to high temperatures with a high thermal conductivity,
- Need to rely on gas circulation as the most appropriate heat removal mean, while privileging design options favouring natural of forced circulation core cooling: bottom to top core coolant circulation, top to bottom DHR heat exchangers circulation, the latter being located above the core, minimization of primary circuit pressure drops (at first in the core). The a minima objective was to be able to ensure natural circulation DHR for pressurized situations, allowing thus the safe management of total loss electrical power sources.

The key parameter to favour the gas circulation is the primary pressure: influencing the natural circulation potential and on the pumping power of the active systems (circulators). The latter is inversely proportional to the square of the pressure, and this is particularly sensitive in the low pressure range. So, if a break occurs in the primary system, the limitation of the primary pressure to 10 bar instead of 1 bar leads to a gain of a factor of 100 on the safeguard systems required pumping systems.

The DHR strategy is therefore based on a guard vessel enclosing – as close as possible – the primary circuit and allowing thus to limit the pressure loss in case of breach (notion of backup pressure). A middle pressure strategy was retained because it allowed the best trade-off between performances and the impact on the integration and reactor operation: it relies on a steel guard vessel initially not pressurised. The mechanical design optimization led to a backup pressure of about 10 bar (depending on break kinetic/sizes) with more than 4 bar after the first 24 hours. This pressure range allows to combine natural circulation core cooling (for break sizes of about 3 inches, thanks to heavy gas like nitrogen injection) and forced circulation core cooling (with a moderate pumping power : the power demand for all active systems supposed to be called simultaneously remains below 500 kWe for an order 3 redundancy of the DHR loops.

Different guard vessel concepts were considered and the spherical geometry (33 m diameter) was selected as the reference being the best adapted to the primary system architecture (Figure 4.1).

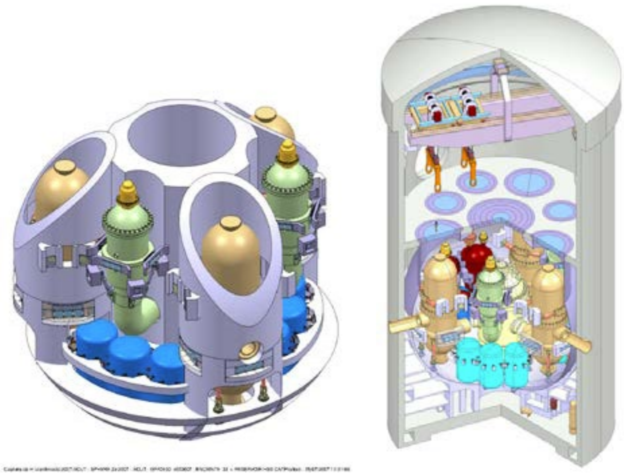

Figure 4-1 GFR and gard vessel

4.2.1 Thermal inertia and grace period

All pressurized accidental situations can be managed in a passive way, including the situation of the total loss of electrical power. The philosophy for the design of the safety functions in the GFR concept relies on passive systems as much as possible. The main remaining difficulty, inherent to the concept, is the absence of grace period between a primary circuit fast depressurization and the need for the active cooling of the core.

Decay heat removal can be through passive means during accidents with primary coolant pressure boundary is intact, and through active or passive means (depending on the pressure) when the primary coolant pressure boundary integrity is lost.

4.2.2 Diversification, active and passive systems

The reactor core can be cooled through active ways either under normal shutdown or emergency shutdown. A passive system is designed for the GFR based on natural heat transfer mechanisms, i.e., natural circulation of water or air.

In ALLEGRO design, there are 3 dedicated DHR loops (3 x 100 % DHR) for redundancy and diversification. Each DHR loop is able to realize the function alone. The heat exchanger of dedicated DHR loops (helium / water heat exchanger) is located above the core at sufficient elevation to allow also natural helium circulation if necessary. Each system has sufficient capacity to remove all decay heat on its own while meeting the fuel and primary coolant pressure boundary limit.

4.3 Confinement of radioactive materials

In the area of retention of radioactive substances, the ALLEGRO demonstrator has the following safety barriers against the release of radioactive substances into the environment: cladding, primary circuit, guard vessel and the containment.

The fuel design has a potential impact on the containment/confinement function. The GFR fuel will retain most of the fission products within the matrix. However, the failure of (and fission product release from) this matrix may be significantly different from the behaviour of conventional pin clad fuel concepts. Therefore, the question of the effectiveness of the matrix as a “barrier” must be examined. It is possible that a higher reliance may be required from the containment/confinement systems.

4.3.1 Materials

Concerning in-core structural materials (clad in case of pin-type fuel assemblies, reflector, control rod guides, others), the main challenge is to develop materials able to withstand fast neutron induced damages and high temperatures (up to 1600°C at least to be confirmed, in-core when considering accidental situations). Ceramic materials (monolithic, composite) will therefore be the reference option and as a back-up some composite cermet structures, refractory alloys as well as inter-metallic compounds will also be considered. The reflector structure requires in addition specific neutronic properties in order to reduce efficiently neutron leakage and to protect the surrounding vessels; due to the compromise of its expected properties, an inter-metallic compound of Zr and Si is the favorite candidate at this stage for this component.

These design provisions aiming at protect the structures and vessel materials will be tested progressively in the ALLEGRO reactor at lower temperatures and neutron fluxes related to the reference GFR reactor. In addition, such designed protections should include the reduction and control of neutron and gamma doses for radioprotection purposes for both the reactor and fuel handling and transfer systems.

An R&D programme on SiC and other materials will support such design provisions.

4.3.2 Safety barriers

The GFR has the following safety barriers against the release of radioactive substances into the environment: the fuel cladding, the boundary of the reactor coolant system, guard vessel and the containment. The safety concept includes protection of the barriers by averting damage to the plant and to the barriers themselves. It further includes mitigation measures to protect the public and the environment from harm in case these barriers are not fully effective [19].

4.3.3 Source term

From the point of view of the source term, following factors are considered:

- type of initiating event,
- extent of damage of the core (depending mainly on availability of safety and other systems and equipment),
- state of the containment (leaktightness),
- operator actions during severe accident course.

Type of initiating event is important especially from the point of view of the route of fission products escape from fuel rods and leak into containment and the environment. End points of the route (regarding fuel and containment or environment), length of the route and its characteristics are important parameters that set the fraction of fission products settled in the route. Velocity and resistance for flow of medium from the primary circuit in the stage of severe accident after fuel damage has relevant influence. It is possible in principle to predict that following groups of initiating events will have similar characteristics:

- Accidents initiated by a break of primary circuit (also accidents with loss of cooling and induced failure of the primary circuit) and leakage of helium into primary containment – accidents without primary containment bypass before containment melt-through by corium. In the case of such accidents, retention of fission products is significant in primary circuit and in the containment. Unmitigated release of fission products can be expected only in the final stage of unmitigated accidents – after damage to primary containment.
- Accidents with containment bypass. Bypass is formed either directly as an initiating event or it is induced especially in the first stages of accident. Containment bypass accident is a significant accident from the point of view of the source term, because although the route of fission product escape can be complicated in this case, the route is direct escape route into the environment. After failure of fuel cladding and opening of the route into the environment, there is no functioning barrier against spreading of fission products into the environment. Also if considering significant retention of radionuclides along the route, the final source term is large and more or less qualitatively similar in all variants of scenarios.

Extent of core damage sets mass of fission products leaking from fuel assemblies – it means inventory of fission products that is flowing along the route from fuel into the environment. following core damage states are distinguished:

- Fuel cladding damage only – release of the gap and pin plenum inventory of fission products only (gap between cladding and fuel). In such a scenario, substantial part of fission products is localized in structure of fuel pellets and fission products from fuel are not released. Source term is limited only to fission products in the gap and the pin plenum.
- Damage of fuel rods with fuel melting – relocation in the reactor vessel and inside of the primary containment, without breaching of the primary containment (successful cooling before breach of the primary containment). In this case, gradual release of fission products from the ceramic structure of fuel takes place. Relatively wide spectrum of fractions of released fission products can be achieved if severe accident course is interrupted with stabilisation of corium in the reactor vessel and even

partially in the primary containment (based on reached extent of fuel damage). It is expected that coolable configuration of fuel can be reached only after restoration of heat removal from fuel in the case of only partially damaged fuel configuration – it means release of fission products form fuel structure will be limited.

- Damage of fuel rods with fuel melting and release of corium from primary containment together with reaction of corium with core catcher systems. In this case, source term is maximal with release of most of the inventory outside of controlled boundaries. Most of the fission products will be released from fuel. There is partial retention of fission products in internal spaces of the facility and almost all volatile fission products are released. Source term will be independent on scenario.

Leaktightness of containment in a substantial extent sets the size of source term as the leaktightness directly sets conditions for transport of fission products into the environment (only for scenarios without containment bypass). In the GFR during normal operation, practically perfect leaktightness of primary containment and not tight secondary containment are presently considered.

It is not expected, that physical criteria for containment damage will be modelled in computational analyses (during an analysis of containment response using an integral code, mechanical stress of structures is not calculated). Any potential leak or damage of containment must be based on additional criteria.

Taking into account the dynamics of severe accidents and source term, following variants can be distinguished:

- Isolated primary containment without any damage during the whole accident. In this case is release of fission products very low or limited for release through considered leak of attached systems.
- Formation of (relevant) leak of containment in the early stage of severe accident. Such a case characterises the source term for damaged containment in time before significant retention of fission products in the containment (with containment damaged during ongoing release of fission products from fuel and corium).
- Formation of primary containment leak in late stages of accident (for example due to reaching of containment failure pressure as the result of long-term containment pressurisation). Such a case represents opening of a route for fission product transport into the environment when retention in containment took place). In spite of low leak of primary containment (and also potentially long time period between fuel failure and primary containment failure), this case is similar to the previous case and separate group for scenarios with containment failure in later phases of severe accidents will not be inevitable.
- Variants in which failure of primary containment leaktightness is defined in the scenario (conditions of system operation in the scenario) form an individual group. Containment bypass as the result of initiating event or induced are included.

Operator actions influence the course of severe accident with various effects. Operator actions can significantly change the course of accident. From the point of view of source term, if effects from above mentioned factors are not taken into account, operator actions mitigating the consequences of severe accident are the only relevant actions.

4.3.4 Containment bypass

Regarding containment bypass, the GFR design aims to reduce the possibility of loss of primary helium using features such as a guard vessel and a combination of failures is required for containment bypass to occur. Nevertheless, the Helium Supply System for GFR is designed to maintain the primary coolant inventory and to extract coolant for purification. Therefore, certain failures of this system could provide a containment bypass route:

- a purification system break inside the containment building could provide a release path to the environment via the containment filtered ventilation system,
- a failure of a radwaste gaseous tank could provide a direct release path.

5 MANAGEMENT OF SEVERE ACCIDENTS

An R&D programme is defined in order to adequately identify the severe plant situations that will be managed by design for practical elimination. Several families of accident scenarios leading to severe plant conditions have been preliminarily identified. An approach was proposed to distinguish those families depending on the integrity of the safety barriers, the magnitude and the dynamics of the phenomena induced by the accidents, and the possible associated cliff-edge effects. A preliminary set of situations identified by means of this approach are categorized depending on:

- The dynamics, the linearity and the scale of the phenomena;
- The integrity of the barriers, the state of the close containment (guard vessel) and of the systems available;
- The core geometry and its coolability;
- The reactivity control (neutronic and chemical), the criticality control and the related power extraction capability;
- The factors governing the course of the accident and the possibility of controlling them in order to control the accident;
- The knowledge of the phenomena coming into play during the accident;
- The overall ability to control the accident or the possibility of demonstrating its practical elimination.

Moreover, the ability of the GFR to withstand severe plant conditions relies mainly on the behaviour of the core materials and core supporting structures at high temperature [20], including a chemically aggressive atmosphere due to nitrogen ingress, possible water ingress, and less likely air ingress. Tests must be carried out to assess the capability of the highly refractory GFR core materials to sustain the accidents associated with severe plant conditions. The behaviour of UPuC and SiC-SiCf at 2000°C under several atmospheres were studied. Oxidation on composites may occur due to dissociation of mixed carbide. As far as the air ingress situation is concerned, experimental studies, carried out between 1000°C and 1700°C, show two oxidation patterns: Passive oxidation with the formation of a protective SiO2 layer at low temperature and high oxygen partial pressure, and active oxidation with the formation of an unstable SiO layer at high temperature and low oxygen partial pressure.

In particular, the preliminary safety analysis suggests that the current ALLEGRO design allows the decay heat to be removed in any accidental situations (pressurized or not, even with a large leak, including an additional single failure or multiple failures), owing to diverse systems with moderate power supply requirement and a leak tight containment. In addition, the natural convection capabilities could be retained in most of the situations, including small break LOCAs (e.g. by utilizing DHR system capacity in combination with dedicated independent system increasing coolant inventory in the primary system).

All pressurized accidental situations can be managed in a passive way, including the situation of a total station blackout.

The guard vessel integrity in case of LOCA cumulated with SBO must be preserved by an adequate heat removal through the DHRS in natural convection to keep the guard vessel pressure at un acceptable level..

Additional R&D on severe accidents scenarios and phenomena need to be performed in the future.

5.1 Prevention

Considering the limited thermal inertia of the GFR core and of its coolant on the one hand, and the specific power of the core and the absence of high negative reactivity feedback, on the other, the prevention of severe accidents relies mainly on the refractory properties of the core materials and on the reliability of the shutdown system.

These additional considerations include:

- Systematic application of the principle of the Defence-in-Depth into the design at all levels;
- Expanded use of passive safety features to achieve the fundamental safety functions (control of reactivity & heat removal);
- Reduced plant complexity and improved maintainability;
- Optimized human-machine interfaces as well as extended use of information technologies.

In the preliminary steps of the study of the reference design basis accident (DBAs) the criteria retained for the assessment of the acceptability of the accidental situations are:

- category 3 situations (estimated frequency ranging between 10^{-2} and 10^{-4} by year):
  - clad temperature <1450 °C;
  - upper plenum gas temperature <1250 °C;
- category 4 situations, the more limiting criterion being considered among (frequency <10−4 by year):
  - fuel temperature <2000 °C;
  - clad temperature <1600 °C;
  - upper plenum gas temperature <1250 °C;
  - no degradation of the fluid channel able to prevent the core cooling.
- categories 3 and 4: a controlled state must be reached at the end of the sequence.

The criteria < 1250 °C for the upper plenum gas temperature prevents from the primary boundary loss of integrity including heat up resulting from the loss of secondary circuits.

5.2 Mitigation

The essential objectives of the accident management are:

- To monitor the plant status;
- To maintain core sub-criticality;

- To protect the integrity of the reactor vessel by ensuring heat removal from the core and preventing excessive loading conditions (both thermo-mechanical and chemical);
- To limit the release of radioactive material to the environment; and
- To regain and maintain control of the plant.

5.3 Situations to practically eliminate

The preliminary safety analysis of ALLEGRO design shows a good core coolant characteristics with resistance of the system to all protected situations. For unprotected situations, the conclusions are still partial, but the case of unprotected loss of flow would be manageable. The application of the Leak before Break concept should allow to practically eliminate very large break situations like guillotine rupture of the main primary loops, provided that an efficient leak detection is put in place in the frame of the “In-Service Inspection and Repair.” In addition, the introduction of a third emergency shutdown system allows reducing strongly the probability of unprotected situations.

Among the situations able to induce a severe accident, a practical elimination strategy is proposed regarding [20]:

- The total rupture of the reactor vessel; the provisions enabling this elimination rely mainly on prevention features as design margins, periodic control and inspection of the thermal barriers and on the vessel, limitation of the operation duration of the reactor and limitation of the loadings under operation including transients (limitation of the operation domain in order to keep reasonable loading during starting/shutdown transients for instance);
- A large radial core compaction; it is foreseen to reduce the gap between the assemblies in order to prevent an uncontrollable (namely, a reactivity insertion exceeding 1 $) power excursion in case of hypothetic closure of this gap; prevention provisions able to practically eliminate fuel handling faults able to lead to prompt criticality will also been foreseen but have not been yet studied in a detail way for the GFR;
- A control rod ejection will be a priori physically impossible by design on condition that the resistance of its stop in case of downwards ejection is demonstrated;
- The formation of a detonable mixture of hydrogen and/or carbon will be prevented by the implementation of recombiners and/or igniters (inerters or igniters ?) if the results of the calculation of realistic scenarios that should be performed with a severe accidents calculation code show the possible formation of such mixtures;
- A water ingress able to lead to a prompt criticality will be eliminated by dedicated provisions (isolation valve, early detection devices) if plausible, according to transient calculations still to be performed; the full power operation will be distinguished from the intermediate state, the shutdown state and the combination with a prior accident (induced break).

6 SAFETY OF THE FUEL CYCLE AND OTHER RISKS

6.1 Type of fuel

Pin concepts with a fuel (solid solution) in the form of pellets are not excluded. The use of pins with solid solution fuel, even if it refers to a different logic with respect to fission product retention, easily permits the required heavy atom content. The pin type solid solution fuel should be considered as a preferred back-up (for concept robustness). Furthermore, and with the same logic, it offers the possibility to consider the use of the oxide fuel. For the cladding, SiCf/SiC is the ambitious choice while metallic refractory or semi-refractory stainless steel could be envisioned as the back-up solution even if it limits the coolant temperature, and poses a compromise between neutronic properties and thermal durability.

6.2 Management of waste (quantity, quality)

The gas management systems (primary and secondary circuits, guard vessel, containment) remove the concerned gases and radioactive substances from the primary and intermediate coolants. This complicates the design, service, and maintenance of this equipment, especially the maintenance of filters in the space-limited environment of the ALLEGRO design. The following main subjects associated with waste management will continue to be assessed:

- Activity-related issues in the design & maintenance of gas management systems (space & shielding, etc.).
- Issues related to regeneration & replacement of the primary circuit and close containment (guard vessel)filters.
- Waste management in the entire reactor.

The waste generated at the plant will consist of several streams:

- Spent fuel, including strategy for management of failed elements.
- Solutions from washing the fuel (if it is decided to do so) and other things.
- Filters from atmosphere management systems
- Various wastes from helium purification system. This may include liquid and gaseous waste.
- Other waste (clothes, gloves, broken components and tools, etc.)

6.3 Radiation protection

The containment structure is an essential design feature for the protection of plant personnel and public from undue exposure to direct radiation from radioactivity sources within the structures, systems and components located inside containment structure, both during the normal operation and under accident conditions.

As already mentioned in §4.3.1 (Materials), design provisions aiming at protect the structures and vessel materials will be tested progressively in the ALLEGRO reactor at lower temperatures and neutron fluxes related to the reference GFR reactor. In addition, such designed protections should include the reduction and control of neutron and gamma doses for radioprotection purposes for both the reactor and fuel handling and transfer systems.

6.4 Other risks

For the case of a depressurization event, the GFR will have to be designed to overcome the challenges of using a high-pressure gas coolant with poor heat transfer characteristics. The safety related consequence of the high coolant pressure is the potential for loss-of-coolant accidents (LOCAs). In addition to the poor heat removal properties of the coolant at low pressure, the GFR safety case is different compared to the VHTR. The safety-related architecture will rely on passive as well as active systems. The efficiency, simplicity, robustness, reliability, and economy will be the essential criteria for evaluating the selected safety design options. Early GFR concepts faced difficulties in managing LOCAs safely, as they were designed with a high power density (and low temperature margins) to achieve high breeding performances and short doubling times. Design parameters for the new GFRs aim at achieving a better balance to ensure very high performance with respect to both sustainability and safety objectives. More generally, as the Generation IV goals require excellence in safety and reliability, a very low likelihood and also degree of reactor core damage is aimed to eliminate the need for off-site emergency response. A good balance in the safety approach between active and passive means (to be promoted as far as possible) is a crucial issue and options in that area may remain open until final choices are taken for the GFR prototype.

Experimental studies carried out for air ingress, between 1000°C and 1700°C, show two oxidation features:

- passive oxidation with the formation of a protective SiO_2 layer at a low temperature and high oxygen partial pressure,
- active oxidation with the formation of an unstable SiO layer at a high temperature and a low oxygen partial pressure.

The passive oxidation regime would not lead to a loss of clad mechanical properties. However, the active regime will threaten clad integrity and must be demonstrated to be reached only for a limited duration and in a limited region of the core. Transient calculations allow assessment of the flow rate and the oxygen partial pressure crossing the core in a realistic air ingress accident. For the ALLEGRO design, the sensitivity study of the 10”-10” LOCA case with a 30000 m^3 containment building and only 1 RHP loop available is considered as a bounding case in terms of overheating and oxygen partial pressure. Based on these results, the active oxidation regime is only reached before 13,000 s when the oxygen concentration available for oxidation is very low.

Thermo-dynamical calculations have shown that the cladding (SiC/SiC) can react with nitrogen in cases with nitrogen ingress or injection in the primary circuit. This reaction can take place only if the cladding temperature is below a transition temperature ranging between 1200°C (at a low nitrogen partial pressure) and 1600°C (at a higher nitrogen pressure). However, in the transient investigated, the higher the nitrogen pressure, the lower is the temperature and the more the reaction should be limited by its kinetics. Moreover, considering that the enthalpy of the nitriding reaction is of the same order of magnitude as that of the Zircaloy oxidation by steam, the heat released by the nitriding in case of reaction runaway would heat the system above its transition temperature, thus not allowing a stable nitriding reaction.

7 SUMMARY OF PROGRESS NEEDED

GFR- Research requirements appropriate for intervention at European level include:

- Simultaneous improvement of the robustness and simplification of the decay heat emergency removal systems,
- Development of sandwich clad fuel concept including pin encapsulation and irradiation of assembled pins/rodlets;
- Studies related to the severe accident behaviour of an all-ceramic core: Core degradation mechanisms and radionuclide transport/retention in a gaseous environment;
- High temperature material qualification and component design and qualification;
- Experience feedback and current research relating to the HTR and VHTR concepts may yield numerous solutions of benefit to the GFR. This applies principally for
    1) development of structural materials suitable for high-temperature operation;
    2) thermal insulation technology;
    3) helium valve technology (in particular fast acting isolation valves);
    4) helium blowers;
    5) intermediate heat exchange and steam generator technology (in particular experience feedback from the VHTR);
    6) helium purification technologies;
- Development of high power blowing machines.

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