Gas-cooled Fast Reactor (GFR) Risk and Safety Assessment White Paper

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Abstract

This paper presents the application of the Integrated Safety Assessment Methodology (ISAM), developed by the Generation IV International Forum (GIF) Risk and Safety Working Group (RSWG), to the Gas-Cooled Fast Reactor (GFR). Developed in the framework of the Euratom project SARGEN_IV, under the supervision of the GFR System Steering Committee (SSC) and RSWG, the paper compiles information that has been generated within the project and has been collected in the GFR-related Euratom projects GCFR STREP and GoFastR. After presenting a short overview of ISAM, the design of GFR and its demonstrator ALLEGRO are reported in more detail and then the results of the ISAM application are summarized.

1. Short overview of the ISAM assessment methodology

The RSWG has developed a methodology, called the Integrated Safety Assessment Methodology (ISAM), for use throughout the Gen IV technology development cycle. The ISAM consists of five distinct analytical tools (Ref. 1) which are intended to support achievement of safety that is "built-in" rather than "added on" by influencing the direction of the concept and design development. The ISAM tools are the follows:

- Qualitative Safety Features Review (QSR)
- Phenomena Identification and Ranking Table (PIRT)
- Objective Provision Tree (OPT)
- Deterministic and Phenomenological Analyses (DPA)
- Probabilistic Safety Analysis (PSA)

Figure 1 shows the overall task flow of the ISAM and indicates which tools are intended for use in each phase of Generation IV system technology development.

Each of the analysis tools that is part of the ISAM is briefly described here:

- *Qualitative Safety Features Review (QSR)*

The Qualitative Safety Features Review (QSR) is a new tool that provides a systematic means of ensuring and documenting that the evolving Gen IV system concept of design incorporates the desirable safety-related attributes and characteristics that are identified and discussed in the RSWG's first report entitled, "Basis for the Safety Approach for Design and Assessment of Generation IV Nuclear Systems", as well as in other references (e.g. the INPRO Safety methodology). Although this element of the ISAM is offered as an optional step, it is believed that the QSR provides a useful means of shaping designers' approaches to their work to help ensure that safety truly is "built-in, not added-onto" since the early phases of the design of Gen IV systems. Using a structured template to guide the process, concept and design developers are prompted to consider, for their respective systems, how the attributes of "defence in depth", high safety reliability, minimization of sensitivity to human error, and other important safety characteristics might best be incorporated. The QSR also serves as a useful preparatory step for other elements of the ISAM by promoting a richer understanding of the developing design in terms of safety issues or vulnerabilities that will be analyzed in more depth in those other analytical steps.

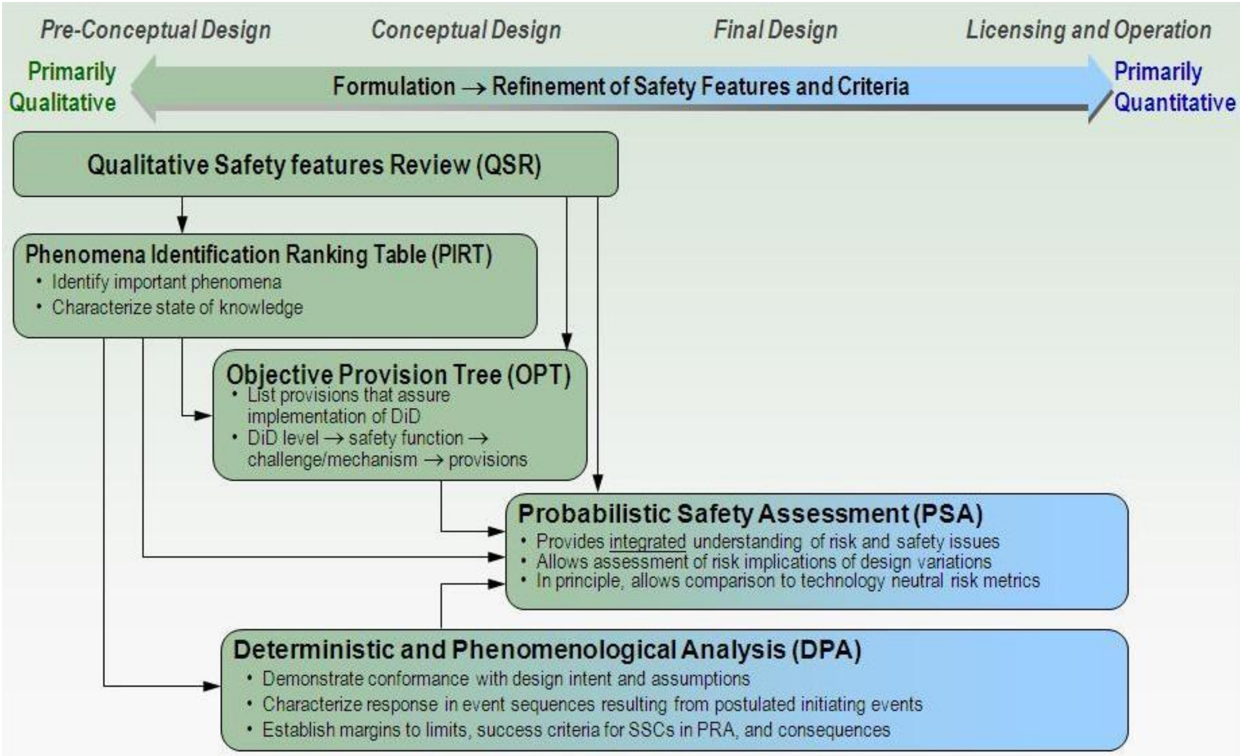

Figure 1: Proposed GIF Integrated Safety assessment Methodology (ISAM) Task Flow

- *Phenomena Identification and Ranking Table (PIRT)*

The Phenomena Identification and Ranking Table (PIRT) is a technique that has been widely applied in both nuclear and non-nuclear applications. As applied to Gen IV nuclear systems, the PIRT is used to identify a spectrum of safety-related phenomena or scenarios that could affect those systems, and to rank order those phenomena or scenarios on the basis of their importance (often related to their potential consequences), and the state of knowledge related to associated phenomena (i.e. sources and magnitudes of phenomenological uncertainties).

The method relies heavily on expert elicitation, but provides a discipline for identifying those issues that will undergo more rigorous analysis using the other tools that comprise the ISAM. As such, the PIRT forms an input to both the Objective Provision Tree (OPT) analyses, and the Probabilistic Safety Analysis (PSA). The PIRT is particularly helpful in defining the course of accident sequences, and defining safety system success criteria. The PIRT is essential in helping to identify areas in which additional research may be helpful to reduce uncertainties.

- *Objective Provision Tree (OPT)*

The Objective Provision Tree (OPT) is a relatively new analytical tool that is enjoying increasing use. The International Atomic Energy Agency (IAEA) has been a particularly influential developer and proponent of this analysis tool. The purpose of the OPT is to ensure and document the provision of essential “lines of protection” to ensure successful prevention, control or mitigation of phenomena that could potentially damage the nuclear system. There is a natural interface between the OPT and the PIRT in that the PIRT identifies phenomena and issues that could potentially be important to safety, and the OPT focuses on identifying design provisions intended to prevent, control, or mitigate the consequences of those phenomena.

- *Deterministic and Phenomenological Analyses (DPA)*

Classical deterministic and phenomenological analyses, including thermal-hydraulic analyses, computational fluid dynamics (CFD) analyses, reactor physics analyses, accident simulation, materials behaviour models, structural analysis models, and other similar analysis tools collectively constitute a vital part of the overall Gen IV ISAM. These traditional deterministic analyses will be used as needed to understand a wide range of safety issues that guide concept and design development, and will form inputs into the PSA. These analyses typically involve the use of familiar deterministic safety analysis codes. It is anticipated that DPA will be used from the late portion of the pre-conceptual design phase through ultimate licensing and regulation of the Generation IV system.

- *Probabilistic Safety Analysis (PSA)*

Probabilistic Safety Analysis (PSA) is a widely accepted, integrative method that is rigorous, disciplined, and systematic, and therefore it forms the principal basis of the ISAM. PSA can only be meaningfully applied to a design that has reached a sufficient level of maturity and detail. Thus, PSA addresses licensing and regulatory concerns and is performed, and iterated with a beginning in the late pre-conceptual design phase, and continuing through to the final design stages. In fact, as the concept of the “living PSA” (one that is frequently updated to reflect changes in design, system configuration, and operating procedures) is becoming increasingly accepted, the RSWG advocates the idea of applying PSA at the earliest practical point in the design process, and continuing to use it as a key decision tool throughout the life of the plant or system. Although the other elements of the ISAM have significant value as stand-alone analysis methods, their value is enhanced by the fact that they serve as useful tools in helping to prepare for and to shape the PSA once the design has matured to a point where the PSA can be successfully applied.

It is intended that each tool be used to answer specific kinds of safety-related questions in differing degrees of detail, and at different stages of design maturity. As indicated within the Ref. 1 it is envisioned that the ISAM and its tools will be used in three principal ways:

- A use throughout the concept development and design phases with insights derived from the ISAM serving to influence the course of the design evolution.
- A punctual implementation of selected elements of the methodology which are applied at various points throughout the design evolution to yield an objective understanding of risk contributors, safety margins, effectiveness of safety-related design provisions, sources and impacts of uncertainties, and other safety-related issues that are important to decision makers.
- An application in the late stages of design maturity to measure the level of safety and risk associated with a given design relative to safety objectives or licensing criteria.

[This section anticipates the following sections which organise the preparation of the inputs in order to facilitate the work for both: the RSWG which has to prepare the supports and on the other side the System Steering Committee or the Project Management Boards which will be asked to provide the needed inputs.]

2. Overview of Technology

The main GFR design specifications as derived from the general objectives of Generation IV systems are [4], [5]:

- Use of gas as a coolant as a mean of reaching high temperatures. The proposed coolant is helium, which has many beneficial properties: negligible activation, optical transparency (important during fuel shuffling and internal core structures inspection), no phase change (the potential of reactivity swings is reduced), small void effect, harder neutron spectrum, no parasitic capture (improving the breeding gain), and chemical compatibility with water and with structural materials;

- Potential for economic competitiveness by means of simplicity, compactness and efficiency. Due to the high temperature increase in the core there is the opportunity to reach a good thermal efficiency and also to be used for process heat;
- A robust safety demonstration, based on probabilistic safety assessment and defence in depth principles, and including severe accident management.

Specifications related to the core lay-out of the GFR include:

- Fast neutron spectrum core with a zero (self-breeding) or positive breeding gain, with no or very limited use of fertile blankets in order to:
  - Generate as much fissile material as it consumes, with an optimal use of uranium;
  - Have a fuel cycle fed with only depleted or natural uranium;
  - Achieve homogeneous recycling of all actinides, in order to have no separation of plutonium from other actinides (proliferation resistance).
- Core plutonium inventory not exceeding 10 tons/GWe, in order to have a realistic reactor fleet deployment (in a few decades) and high fuel burn-up.

GFR cores have relatively low thermal inertia; design features aimed at overcoming this apparent unfavourable feature include:

- A fuel element based on refractory materials and high thermal conductivity, with the ability to ensure radioactive material confinement up to very high temperatures.
- A primary circuit design based on upward core cooling and a moderate pressure drop for all the primary components and circuit involved in accident scenarios. One essential parameter for safety system performance is gas pressure. The primary helium is pressurized to 7 MPa under nominal conditions. A gas tight envelope enclosing the primary circuit has been added in order to limit the loss of pressure in case of loss of primary coolant. Maintaining high helium density allows the Decay Heat Removal system to rely on moderate pumping power and even on passive natural convection in some situations.

2.1. GFR conceptual design

A significant effort of work towards a consistent design of the reactor and its fuel has been carried out since 2001. Within the past years, the European GIF members supporting the gas-cooled fast reactor (GFR) (Euratom, France, Switzerland) have committed themselves to developing the conceptual design and safety of the concept, with their work coordinated within the EU FP7 project GoFastR.

GoFastR concentrates on developing the GFR as a more sustainable version of the very high temperature reactor (VHTR). The design goals for GFR are ambitious, aiming for a core outlet temperature of around 850°C, a compact core with a power density of about 100MWth/m^3, a low enough plutonium inventory to allow wide deployment, a self-sustaining core in terms of plutonium consumption, and a proliferation resistant core by not using specific plutonium breeding elements.

For GFR the main issues are centred around the development of a suitable fuel and achieving the necessary diversity and reliability of the safety systems. GFR requires a robust fuel that can operate continually at high temperature and high power density whilst achieving good fission product retention and economically viable burn-up. With regard to GFR-specific safety systems, unlike gas-cooled thermal reactors, GFR does not (and cannot) have a large solid moderator structure so there is little thermal inertia in the core structure. To limit the fuel temperatures, therefore, in fault conditions the safety systems have to supply a flow of coolant through the core with high reliability. The challenge in this instance is providing the reliability without compromising the economics of the system.

Key design choices for the GFR are [4]:

Fuel element

The reference concept is currently a refractory metal lined silicon carbide fibre reinforced silicon carbide clad pin. The pins contain a plenum in which fission gas can collect, and whilst manufacture is considered to be feasible, obtaining a satisfactory compromise between allowing enough pellet-to-clad gap, to avoid pellet-clad interaction, and a low enough fuel pellet temperature to reduce the swelling, is proving difficult. However, the basis of a design has been achieved and the quest to improve the burnup is an ongoing activity.

Core design and performance

The core layout has been chosen to be consistent with the maximum power derived from thermo-mechanical and thermal-hydraulic analyses, the requirements of the reactivity control system and the optimized power distribution.

Primary system

The overall view of the major components of the GFR primary system is shown in Figure 2. The cylindrical reactor pressure vessel is at the centre of the configuration.

Three Power Conversion Systems (PCS) with main heat exchangers and blowers, three Decay Heat Removal (DHR) loops and six gas reservoirs (accumulators, 540 m^3 each with discharge line flow area of 0.008 m^2) are connected to the vessel. The whole primary system is enclosed in a small pressure guard containment of free volume of 13 300 m^3 filled with helium at atmospheric pressure and 50°C under nominal conditions.

The reactor pressure vessel is a large metallic structure (inner diameter 7.3 m, overall height 20 m, weight about 1 000 tons, and thickness of 20 cm in the belt line region). The material selected, a martensitic 9Cr1Mo steel (industrial grade T91, containing 9% by mass chromium, and 1% by mass molybdenum) undergoes negligible creep at operating temperature (400°C). The reference material for the internals is either 9Cr1Mo or stainless steel, typically SS316LN. The global primary arrangement is based on three main loops (3 × 800 MWth), each fitted with one IHX–blower unit, enclosed in a single vessel.

This component limits the consequence of a concomitant first and second safety barriers rupture (the fuel clad and the primary system).

Decay heat removal system

Specific loops for decay heat removal (Figure 3) in case of emergency are directly connected to the primary circuit using a cross duct piping, in extension of the pressure vessel. They are equipped with heat exchangers to remove the heat to a secondary heat sink. The loops are equipped with blowers for forced convection, but they are designed to function on the basis of natural convection, independently of a power source. An essential part of the system is the spherical guard containment that is designed to ensure sufficient system back-up pressure by means of its tightness and of gas injection.

The DHR concept works as follows: As long as normal operating pressure can be maintained, the two high pressure DHR loops will remove the decay heat with moderate pumping power (300 kWe). In the absence of external energy, heat will be removed by natural convection. For cases where pressure cannot be maintained, convection will be ensured by 2 emergency low-pressure DHR loops. Again, coolant circulation may be ensured by electric blowers. In case of blower failure, heavy gas (nitrogen) will be injected into the guard containment. The natural convection achievable with nitrogen has been shown to provide the same cooling efficiency as helium in forced convection.

Fuel handling

The fuel handling system is based on a jointed arm system, with fuel element loading and unloading using a fuel storage drum via lock chambers, the vessel being closed. A dedicated forced convection device, located outside the reactor vessel, is designed to cool the spent fuel sub-assembly during its handling.

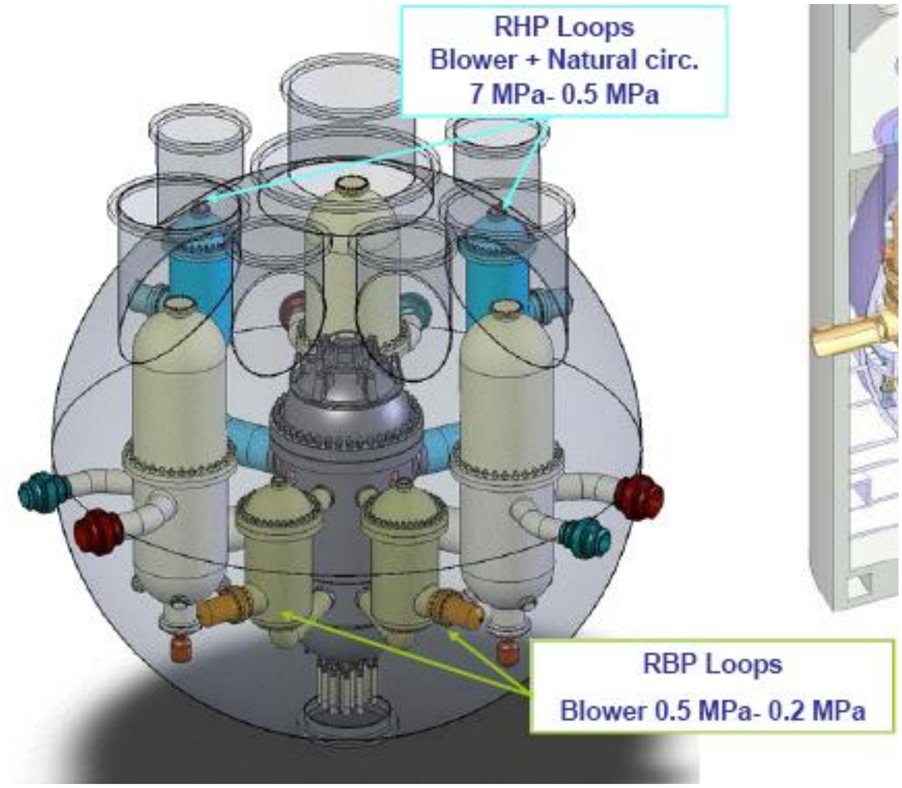

Figure 2: GFR Primary and DHR system inside the guard containment

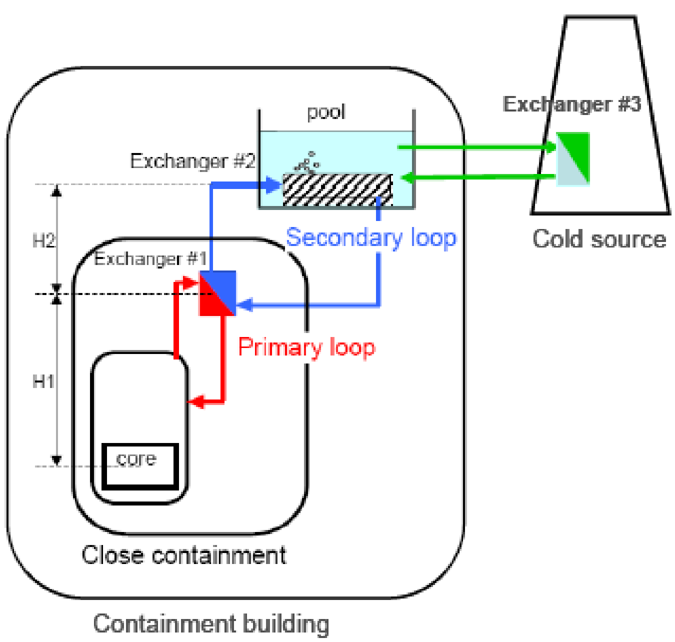

Figure 3: Dedicated DHR loops to be operated in forced or natural circulation

Power conversion system

The current choice is the indirect combined cycle with He-N2 mixture for the intermediate gas cycle. The produced heat will be converted into electricity in the indirect combined cycle with three gas turbines and one steam turbine. The cycle efficiency is approximately 48%, based on assumed component efficiencies and pressure drops. A schematic view of this power conversion system is shown below.

The main parameters of the GFR system are summarised in Table 1 in Section 2.3.

2.2. ALLEGRO demonstrator

A demonstrator reactor [4], [6] is an essential step to establish confidence in the innovative GFR technology. The proposed demonstrator, named ALLEGRO, would be the first ever gas cooled fast reactor to be constructed. The objectives of ALLEGRO are to demonstrate the viability and to qualify specific GFR technologies such as fuel, the fuel elements and specific safety systems, in particular, the decay heat removal function, together with demonstrating that these features can be integrated successfully into a representative system.

Core design

During its life, the ALLEGRO demonstrator will have different cores, with the initial scope of testing the technology at lower temperature.

A starting core or “MOX core” with existing or close to existing technology (MOX pin type sub-assembly with metallic cladding) will be used for the irradiation of a few experimental sub-assemblies to qualify one or several GFR sub-assemblies concepts (and in the first phase the experimental positions will be loaded by steel sub-assemblies). Due to its metallic cladding, the MOX core must be operated under moderate core outlet temperature (~530 °C), but locally, the GFR sub-assemblies outlet temperatures must be increased up to the GFR system values.

A second core or “refractory core” will be used to test the GFR reference technology for the qualification of the sub-assembly performance (temperature, burn-up).

Since both MOX core and refractory core are assumed to rely on the same core supporting structure and control rod drive mechanisms, both cores will have the same subassembly dimensions with the same control rods locations.

The experimental positions are placed near the middle of the core, where the neutron flux is quite flat and at the core boundary where the flux gradient is important. Therefore, various representative GFR irradiation conditions can be approached.

The reactivity control is handled by two independent Control Rod Assemblies groups: 6 Control and Shutdown Devices (CSD assemblies) and 4 Diverse Shutdown Devices (DSD assemblies). Since the core has a very small size, ALLEGRO has to face large reactivity variations along the cycle.

Primary and secondary circuits

In the ALLEGRO 75 MWth configuration, two identical and independent primary loops of 38 MW each remove the core power. Except for the exchange capacities of the main heat exchanger, these two main loops are identical to the main loop of the previous 50 MWth ALLEGRO concept, particularly the primary blowers (the changes in nominal capacities are managed by the variation in the speed of the blowers). The secondary loop includes pressurized water. The final heat sink is the atmosphere. An additional circuit reservation has been made to possibly test high temperature processes or components using part of the reactor power (10 MW). The main elements of the ALLEGRO thermo-hydraulic are displayed in Figure 4.

ALLEGRO decay heat removal

After SCRAM actuation, it is foreseen to use the normal heat removal path with two primary blowers driven by pony motors connected to diesel generators, providing

- 20% of nominal rotational speed at nominal pressure;
- 50% at backup pressure, and
- 100% at atmospheric pressure.

Secondary and ternary cooling circuits are in forced convection or, in case of power failure, in natural convection.

If the primary system does not suffice to remove the decay heat, it will be isolated and the dedicated DHR system, see the schematic of Figure 3, will be started. The decay heat removal system uses specific DHR loops directly connected to the primary vessel. The 3 x 100% DHR loop systems are designed to remove 3% of the nominal power.

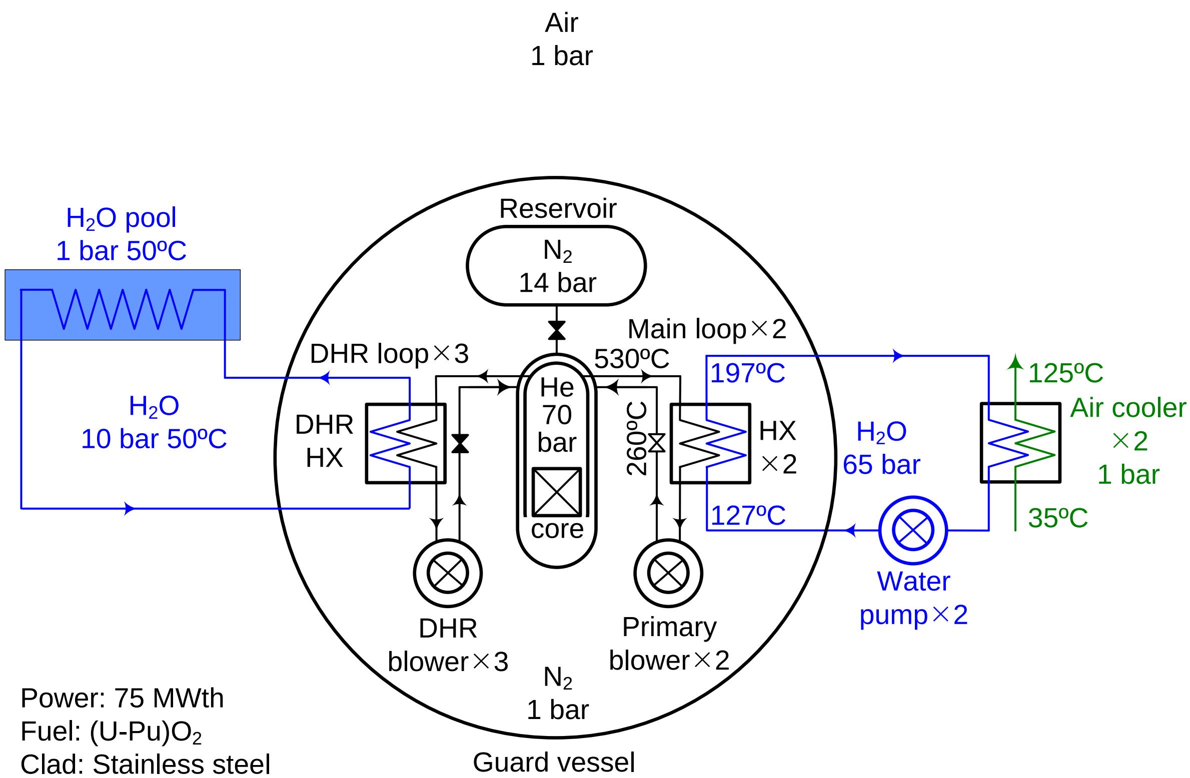

Figure 4: A simplified diagram of the ALLEGRO gas-cooled fast reactor

The decay heat removal strategy is as follows: DHR with natural circulation is proposed for the pressurised case (case of loss of flow combined with a blackout, where the DHR blowers – which would be launched at first intent - would not be available). For the depressurised case, it has been estimated that, due to a limited helium inventory and the free volume in the guard containment, the resulting backup pressure will be no higher than 3 bar. Therefore, forced convection is recommended with the use of blowers that could be fed by batteries if needed during the short term (24 to 48 hours) and then by the power grid or the diesel generator sets, which can be assumed to be recovered at that time. Since the blowing power decreases with the backup pressure, an alternative option – while keeping the same forced convection strategy – could be to pre-pressurise the guard containment in order to limit the pumping power requirements, especially when battery supply is involved.

Like for the main circuit, the DHR secondary loop is a pressurized water loop. The heat is removed from the secondary loop through a horizontal heat exchanger located in a water pool. The water pool is able to act as final heat sink during 24 to 48 hours, if necessary, but would be cooled by an external source after that time.

The intermediate heat exchanger of the decay heat removal loop was calculated according to the two following conservative situations:

- Loss of flow (LOF), with all forced convection unavailable and nominal pressure use (70 bars).
- Loss of coolant gas accident (LOCA), with different back-up pressure levels.

The reference DHR operating foresaw only the use of dedicated DHR loops to cool the core, possibly supported by nitrogen injection in order to enhance the core cooling in natural convection.

The DHR strategy relies on:

- natural convection to ensure a diversified functional redundancy for heat removal in case of high pressure scenarios;
- emergency power supply to be able to start and/or to maintain the DHR circulation for scenarios at low pressure.

For the short term, in case of small breaks (i.e. break which does not induce flow reversal in the core, less than 3 inches) nitrogen is injected in order to favour natural convection. For the long term, due to the close containment, pressure is set to the adequate pressure to ensure natural convection.

2.3. Summary of Generation IV GFR Tracks

Table 1 summarizes the key design parameters of the GFR design concepts identified in the previous sections.

Table 1. Key Design Parameters of Generation IV SFR Concepts

Design Parameters	GFR	ALLEGRO
Thermal power, MW	2 400	75
Primary coolant	He	He
Primary coolant pressure, MPa	7	7
Helium temperature at core inlet, °C	400	260
Average helium temperature at core outlet, °C	880	530
Primary flowrate, kg/s	940.1	28.2
Core pressure drop, bar	1.43	0.84
Fuel pin internal pressure at 20°C, bar	10.0	
Secondary coolant	Mixture of 20% He and 80% N_2	Water
Secondary coolant pressure, MPa	6.5	6.5
Secondary coolant flowrate in one loop, kg/s	902.0	125.4
Secondary coolant temperature at IHX inlet, °C	362.0	127.1
Secondary coolant temperature at IHX outlet, °C	820.0	197

3. Overview of the Safety Architecture's characteristics and performances
(*System's SSC/PMB in charge; RSWG in support*)

The safety studies to performed in GoFastR aimed at fulfilling Euratom's contribution to Gen IV GFR WP1.3 "GFR Safety Studies" of the GFR Conceptual Design and Safety Project Plan and included:

- Elaboration of the GFR safety approach
- Studies on risk minimisation
- Transient analysis
- Probabilistic safety studies
- Severe accident studies

The integration of safety issues in the early phase of the design of a 4th generation reactor is expected. For this purpose, probabilistic insights are increasingly employed in the safety demonstration in combination with the deterministic approach in the frame of a so-called risk informed approach.

The iterative process "Design/List of events/Safety analysis" is considered indispensable to achieve a comprehensive safety study.

3.1. Qualitative Safety Features Review (QSR)

The QSR provides a check list of issues which should be taken into account during application of the other safety techniques (OPT, PIRT, transient analysis and PSA). The RSWG ISAM report [1] contains a comprehensive list which is applicable to all Gen-IV designs.

In addition to the general check list, a GFR technology-specific (Class 4) set of recommendations could include the following issues:

- In case of failure of the primary pumps, consider the use of the nominal cooling path through Power Conversion System (PCS) for decay heat removal (DHR), in particular use of the pony motors.

- For the case when the cooling through PCS is not efficient any more, develop a simple logic for switching from the cooling path through PCS to cooling path through DHR loop, using if possible passively actuated devices (e.g. check valves). Analyse the dynamic behaviour of the system during this switching.

- Include in the analysis the erroneous opening of the DHR valves providing flow bypass.

- Develop a simple logic for connecting the gas reservoirs to the primary circuit in case when additional gas injection is needed for efficient DHR, using if possible passively actuated devices (e.g. check valves). Consider the use of the heavy gas (e.g. nitrogen) for injection. Consider the impact of the adiabatic expansion of the gas in the reservoir and corresponding gas cooling down; in particular optimise the injection point and the rate of the injection in order to avoid the high thermal stresses in the core structures and vessel caused by the cold gas injection. Analyse the dynamic behaviour of the system during gas injection.

- For the cases with loss of backup pressure (containment failure), consider a passive devices (e.g. Brayton cycle machine) for DHR under low-pressure conditions. Consider the problem of starting this device in a passive manner.

- In case of the transients in which the fuel heats up and expands, consider the probability of the fuel-clad gas gap closure and corresponding clad failure probability because of the high clad stresses under pellet-cladding mechanical interaction conditions.

- In case the water is used as a secondary coolant in the DHR system, consider the transients with water ingress in the core.

3.2. Phenomena Identification and Ranking Technique (PIRT)

While it is recognised in the GoFastR project [5] that PIRT is helpful in defining the course of accident sequences and defining safety system success criteria, a formal PIRT in the sense of systematically screening and ranking accident phenomena has not been considered yet.

Instead, the efforts on analysing and improving safety characteristics have been concentrated on critical issues, i.e. phenomena ranked to be of high priority, known from past work on the GFR system. Important work to quote in this respect is:

- A classification list of potential initiating events (PIEs) was produced in the FP5 GCFR project. It was mainly based on early design studies of GCFRs (in the sixties/seventies), and was adapted during the FP6 GCFR project **Error! Reference source not found.** using a simplified Master Logic Diagram (MLD) approach

  A MLD is a form of logic tree, a top-down method when developed from a top event (release of radioactive products to the environment) to more basic event (such as leak through a pipe). A MLD shows a "cause and effect" relation between two events, placed in different boxes connected by logic gates. It is an iterative construction coupling a top event to the successive causes, ending in elementary PIEs.

  The starting point for identifying the PIEs was to separate the GFR design into major plant items, covering all areas of the design. This method was considered appropriate as the GFR design evolves as it is simple to adapt the list of initiating events to include new systems or equipment as necessary. The next step was to identify the main challenges to the operation of each item, without forgetting the interactions between different items. This step was performed using the simplified logic diagram approach. The result was a list of PIEs which were assigned to different Design Basis/ Extension Categories (DBC/DEC) and which have been used as a basis for the ongoing design work and safety assessment.

- The GoFastR report on risk minimisation [6], where the following issues are considered: water ingress, rod ejection practical elimination, concrete/steel vessel, high pressure gas tanks burst, dynamic analysis of guard containment integrity under different depressurization scenarios and preventive DHR system architecture analysis to exclude core destruction. Exemplary qualitative conclusions from this work are that the positive reactivity effect of water ingress is weakened by the negative reactivity of refractory metals in the core, and that nitrogen as heavy gas needs to be injected if the DHR system is to achieve sufficient natural convection in case of a depressurisation of the core.

- The SARGEN-IV report [4] setting out the critical safety features of the GFR system, i.e. the reactivity control system and the decay heat removal system. The report discusses potential issues associated with these systems and further investigations to address these issues.

- The assessment of GFR safety concepts based on a technology neutral approach [7], where GFR features are compared to technology neutral principles and recommendations extracted from Technical Guidelines set up for Gen. III reactors.

The experience from earlier work and from the analyses carried out in the GoFastR project have clearly impacted the current design of the GFR and ALLEGRO and are a step towards a risk-informed design of the machines.

3.3. Objective provision Tree (OPT)

The safety objectives defined for GFR in [8] will be achieved through the application of the defence-in-depth strategy. The strategy uses as a starting point the method of OPTs.

The objective-provisions methodology can be used to systematically review the implementation of the DiD in the design. OPTs provide a logical framework for making an inventory of the DiD capabilities by screening the challenges and mechanisms and identifying the potential safety provisions to achieve the objectives for each level of defence.

OPT Reactivity/Shutdown Fundamental Safety Functions

As discussed in Section 3.2, a simplified MLD approach (similar to OPTs) was initially used for GFR during the FP6 GCFR project [9].

In addition, OPTs were used for a comparison of the DiD capabilities of the direct/indirect concepts, with regard to the Reactivity/Shutdown fundamental safety function. This work involved the creation of an inventory of the DiD capabilities:

- Produce a list of the mechanisms that challenge the successful achievement of the Reactivity/Shutdown FSF for each level of defence
- Identify the potential provisions provided for each level of defence to cope with these challenges – including inherent plant safety characteristics, safety margins, system design features and operational measures
- Assess the requirements for implementation of adequate provisions in design.

To assist in this assessment of the DiD capabilities, Objective-Provision Trees (OPTs) were developed to assess the implementation of DiD Levels 1 to 4 for the Shutdown safety function.

The challenges and mechanisms were identified in the OPTs, including consideration of the different challenges that can result from the same mechanism depending on the level of DiD and its specific objective. The study concluded that although there were a number of significant differences in the performance and equipment of the direct/indirect cycle concepts which need to be considered from a safety point of view, the majority of these differences will affect the DHR or containment safety functions rather than the reactivity/shutdown control safety function.

A list of potential provisions that could be provided for the reactivity/shutdown control safety function for either concept was developed. It was concluded that, regarding the reactivity/shutdown control safety function, there were adequate potential provisions available for each concept and hence Defence-in-Depth was achievable for both concepts.

OPT for degradation or disruption of the heat transfer path

As a practical example for the demonstration of OPT, the SARGEN-IV subtask 4.2 [10] has focussed on applying harmonised safety assessment methodologies to a representative set of initiating events for several Gen IV reactor concepts, identified in [11]. The OPT described below was developed, within this subtask, for the "degradation or disruption of the heat transfer path", which represents one challenge to the safety function "core heat removal" of the ALLEGRO design.

The DiD level 1 provisions for accident prevention can be summarised as conservative design choices (negative temperature coefficient, no substantial core by-pass flow), quality assurance of design and construction, regular inspection and maintenance, and the availability and application of operational procedures. (Figure 5)

The provisions proposed for DiD level 2 are defined as the control of deviations from normal operation and the detection of failures. Provisions to be shown to be implemented are a control and instrumentation system able to control the deviations from normal operation conditions (Figure 6).

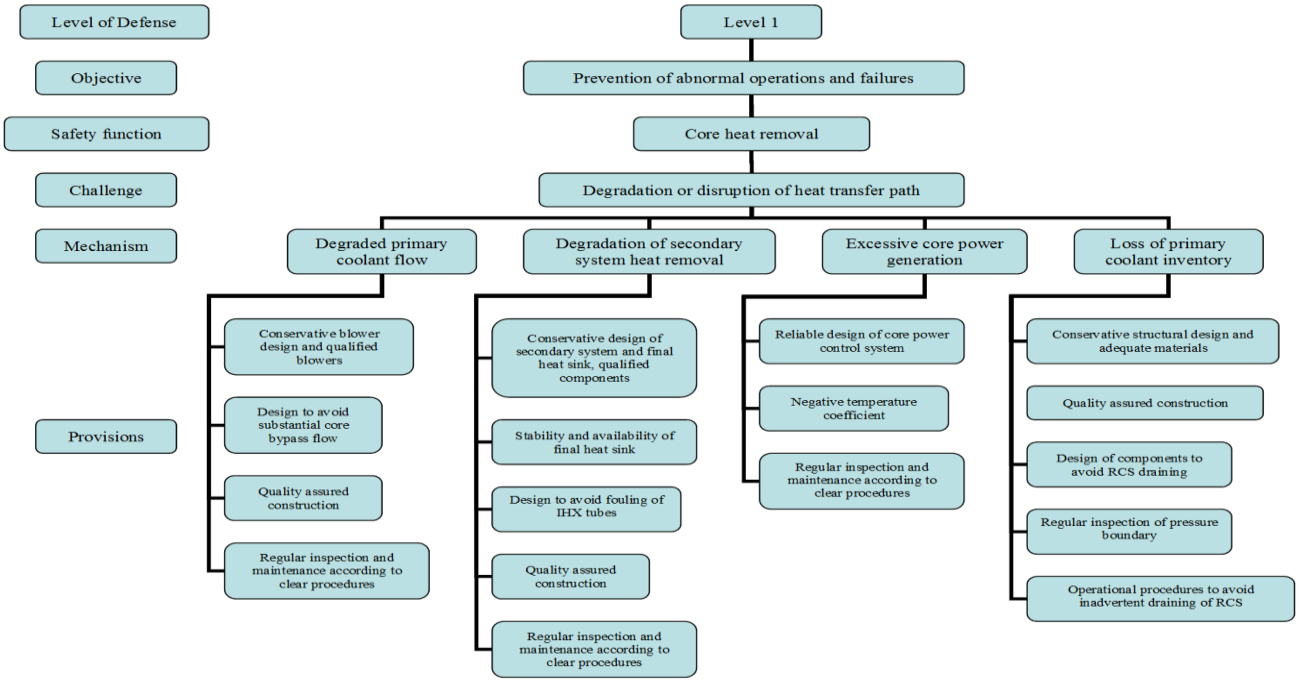

Figure 5: ALLEGRO OPT for Level 1 of DiD, heat removal from the core

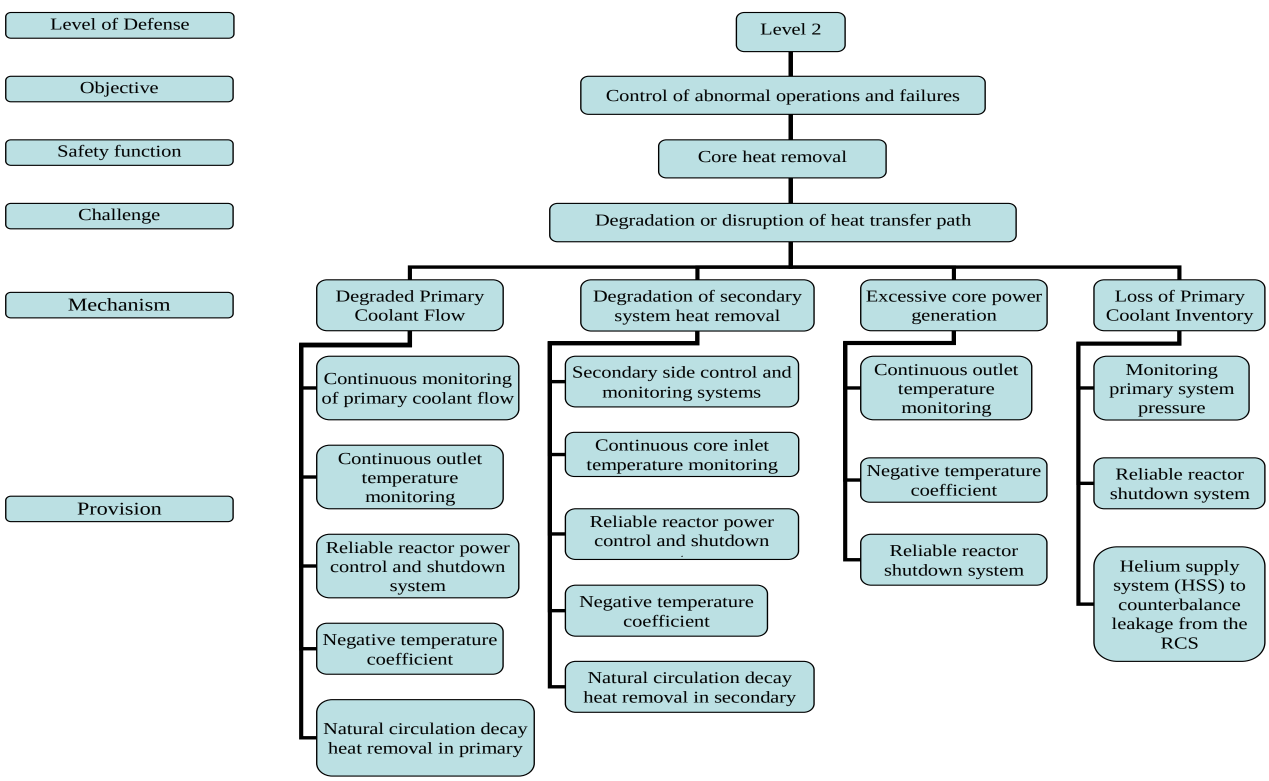

Figure 6: ALLEGRO OPT for Level 2 of DiD, heat removal from the core

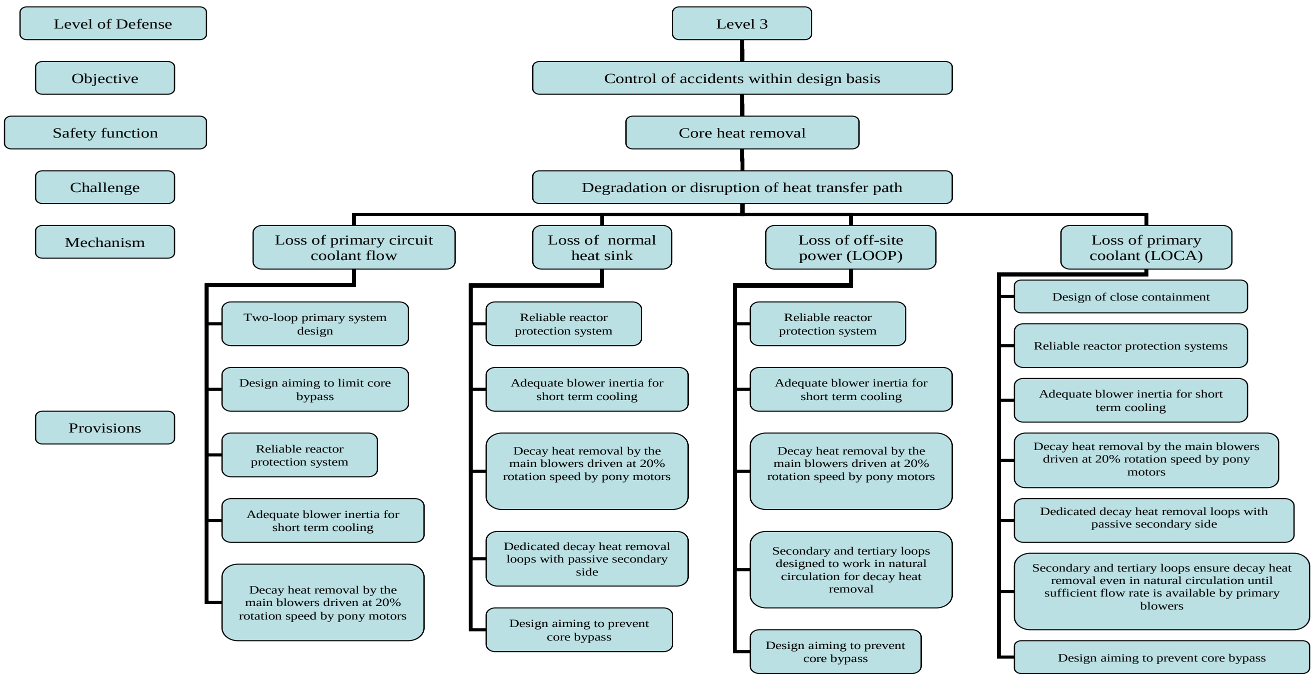

Figure 7: ALLEGRO OPT for Level 3 of DiD, heat removal from the core

The mechanisms for DiD level 3 challenging heat transfer from the core during a design basis accident are identified as PLOF, LOHS, LOOP and LOCA. These are, of course, standard accident scenarios for which the consequences are simulated under assumptions and boundary conditions posed by a certain initiating event. The provisions of DiD levels 1 and 2 are still essential for handling the event, but they are now complemented by independent (like pony motors securing DHR, powered by emergency diesel generators) and passive (natural convection DHR loops) safety systems (Figure 7).

The control of a BDBA under DiD level 4 relies on the expectation of a robust DHR system ensured by sound design, diversity and redundancy. For the case that the system heat sink remains unavailable, provisions typically linked to EOP have to be available (Figure 8).

3.4. Deterministic and Phenomenological Analyses (DPA)

DPA are used as needed to understand a wide range of safety issues that guide concept and design development. In many cases, analytical results show concept properties that remain valid even if the analyses were carried out on a previous or scaled design (as in GFR and ALLEGRO). At the present conceptual design stage of the GFR and of the ALLEGRO demonstrator, it is the main goal of analyses to show qualitative features of the concept and support the selected design options.

Numerous analyses have been carried out on safety issues of gas cooled fast reactors. This section will try to give an overview of analyses that support the current concepts and are addressing issues in three areas:

- Inherent safety features and basic design choices,
- DHR system architecture and
- Transient accident analyses.

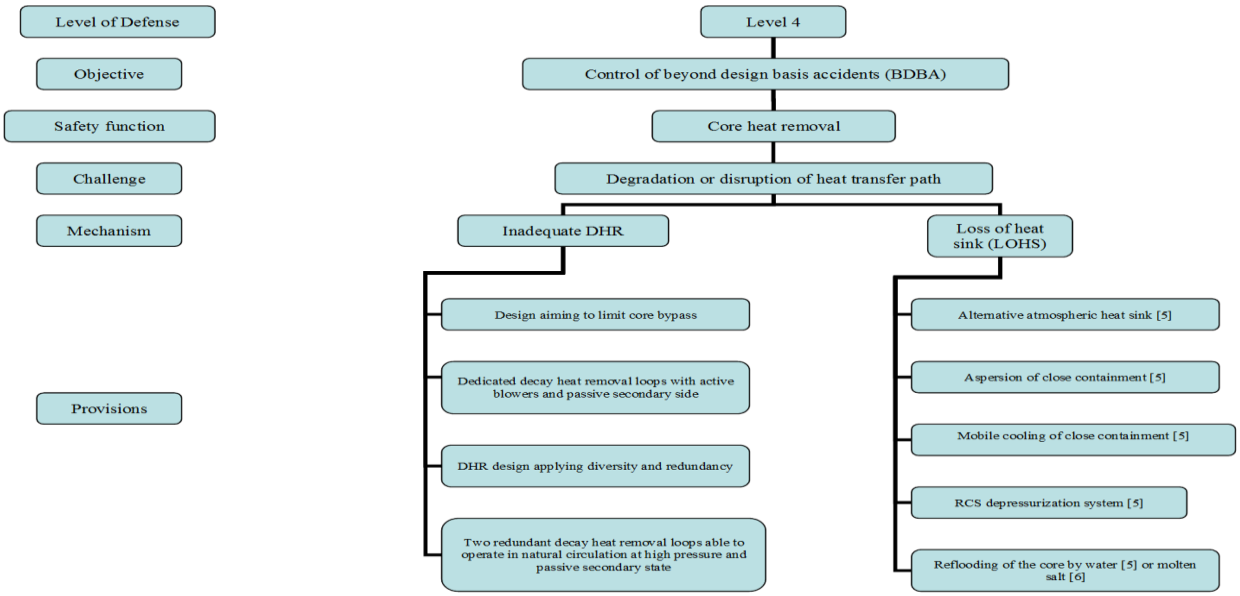

Figure 8: ALLEGRO OPT for Level 4 of DiD, heat removal from the core

3.5. Deterministic and Phenomenological Analyses (DPA)

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3.4.1 Inherent safety features and basic design choices

The current section gives examples of exploratory risk minimisation studies that were carried out in the first phase of GoFastR with the intention of selecting between design alternatives for the GFR safety systems, thus contributing to a risk informed design. The analyses are given in detail in [12].

Water ingress into the core

The use of refractory metals (W and Re) in GFR as materials for cladding liners (to assure cladding tightness against diffusion of fission gases into the coolant) results in a significant neutronic penalty. Analyses suggest a negative reactivity insertion during steam or water ingress because of the increasing absorption rate of the thermalized neutrons by W and Re – in spite of the increasing neutron production rate.

Control rod ejection practical elimination

A control rod design has been proposed that determines the limits for rod movement. The system is designed to resist accidental shocks, implying that the rod ejection can be excluded by design whatever happens to the Control Rod Drive Mechanism.

Dynamic analysis of guard containment integrity under different depressurization scenarios

A model of the guard containment and the main components of the primary system was built within the codes GASFLOW CFD and CATHARE CD. Size and position of the components are preliminary. Analyses assumed an adiabatic expansion into the guard containment through a break in the primary system.

Key results of the dynamic analyses are pressurisation histories for the guard compartment, with a conclusion that the design pressure of the confinement should not be lower than 10 bar.

3.4.2 Transient accident analysis

Within the GoFastR project, the programme of work has included transient analysis and severe accident analysis for both GFR and ALLEGRO.

The emphasis of the transient analysis has been on demonstrating the strategy of core melt exclusion in design basis conditions and increase of the grace time before core degradation in design extension conditions. The transient analysis performed in the FP6 GCFR project served as an important database and starting point for the analysis. The consistency and adequacy of the computer models of the GFR core and primary circuit developed by different GoFastR participants with different codes was first validated for steady-state and PLOF conditions.

Computer models of the GFR system were developed using the following system codes : TRACE/FRED, CATHARE, RELAP5 and RELAP3D. An existing CATHARE model developed by CEA was used as the starting point and modified to take into account the neutronic results for the new core design developed in GoFastR. The GFR system nodalisation diagram is presented in Figure 9.

The transients analysed by the project participants covered LOF, LOCA, TOP and LOHS, both protected and unprotected.

Report [6] has been dedicated to transient analyses of ALLEGRO, and comes to the following conclusions:

"This report presents the safety approach proposed for ALLEGRO and exploratory transient studies that aim at reducing the risk of core damages. To this end, a design including two main loops that can be used as a first line of defence in the core cooling strategy has been investigated. This configuration has been tested for several types of transients:

- pressurised transients belonging to the design basis domain (LOFA, LOOP);
- depressurised transients belonging to the design basis domain (LOCA on a primary loop);
- complex sequences of the design extension domain including unprotected transients.

Calculations have made it possible to analyse various heat removal strategies such as:

- utilisation of pony-motors;
- utilisation of a flywheel;
- utilisation of a non-failed cooling loop (optimisation of the main loops);
- gas injection via the accumulators.

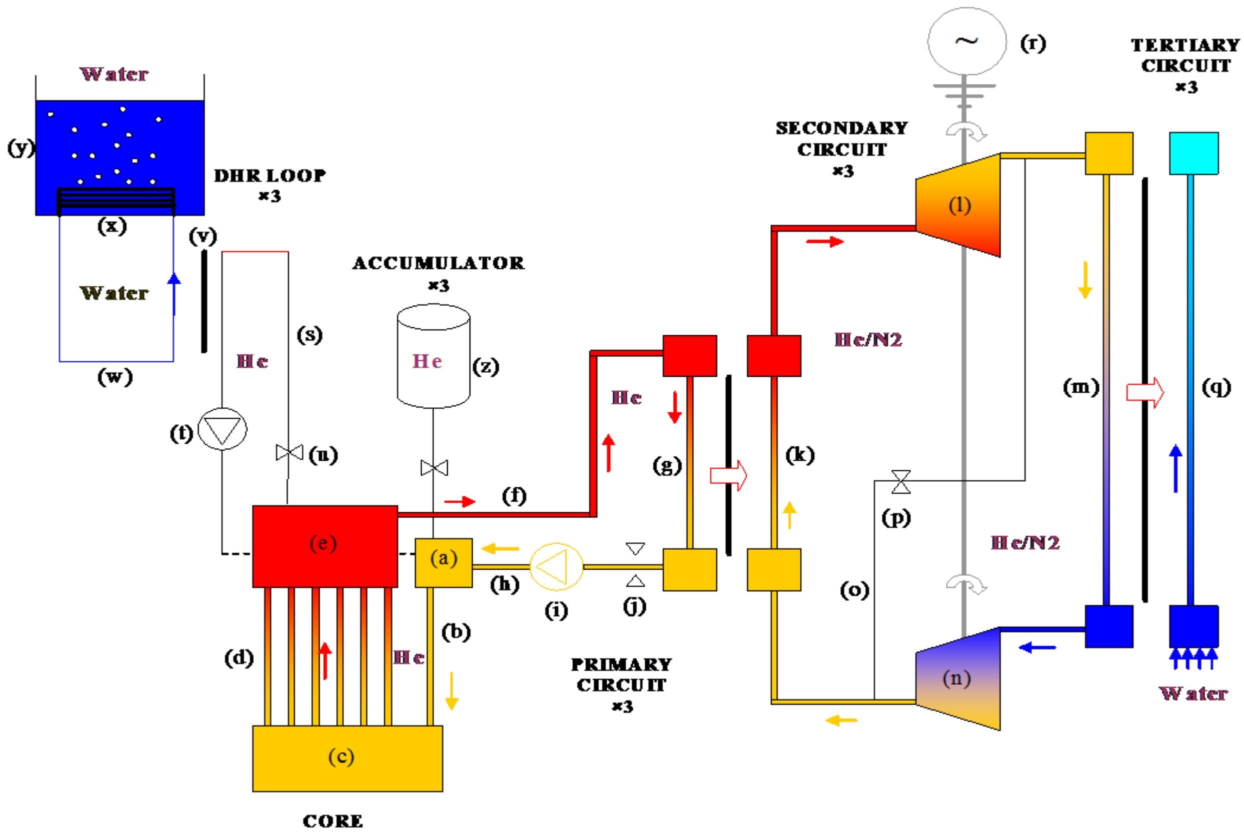

Figure 9: GFR system nodalisation diagram

This represents the first stage of the study for the 75 MWth ALLEGRO reactor under accident conditions. In this configuration, the secondary water loop is modelled in a very simplified manner using boundary conditions. The conclusions of this study generally confirm the results obtained in past studies, in addition to the new interest in optimising the main loops during unprotected transients in order to reduce the risk of core degradation. In particular, the pressurized transients can be managed, even in case of multiple failures, without exceeding the corresponding acceptance criteria.

The next step will now involve designing and modelling the secondary water loop in a more realistic manner. This will make it possible to calculate some transients more realistically (transients for which the behaviour of the secondary water loop may have an impact: shutdown of secondary pumps, function of thermal inertia associated with the water volume), and to broaden the scope of transients taken into account. Finally, the rules to be applied regarding the deterministic studies of accidents should be more conform to the future studies and a PSA supporting the design process would permit to focus our efforts on main vulnerable points of the safety architecture."

3.6. Probabilistic Safety Assessment (PSA)

As proposed in the ISAM, it is recommended to develop PSA models early in the design process, and will be an iterative process with an increasing complexity and level of detail as the design evolves. Thus the PSA can be used as a key decision tool throughout the life of the plant, starting in the design phase, and continuing through to the final design stages where it can help to address licensing and regulatory concerns.

The GFR probabilistic safety approach has been developed and is defined in [13]. The report defines a probabilistic framework for GFR risk assessment that supports the overall GFR safety approach. The application of Levels 1, 2 and 3 PSA for GFR are discussed, as is the scope and purpose of PSA during the different stages of reactor development and operation.

In [13], international best practice and approaches for current reactor designs and their applicability to innovative reactor designs are considered. It is concluded that the existing PSA approaches in use for LWRs are broadly applicable for GFR. However, specific GFR/Gen IV issues require further consideration, for example methods for incorporating innovative design features in probabilistic models and improvement of the knowledge of accident phenomena and modelling within PSA.

A preliminary PSA study is necessary to inform the design and ensure that it represents a balanced risk. Depending on the current status of the design, the transient/accident analysis and the availability of data the early PSA may initially be restricted to a qualitative Level 1 assessment covering a limited number of safety systems, to highlight over-reliance on specific systems, areas of weakness or accident sequences of particular concern.

During the early stages of PSA application, it may be necessary to use generic data (e.g. component failure parameters) as existing data may not be directly applicable to innovative components. The GFR Reliability Analysis Methodology has been developed and proposes a Component Failure Rate Data Base (CFRDB) building methodology [14]. This method is based on the degree of analogy of the components of an existing database compared to the components for the design in progress.

A significant PSA study has already been performed by CEA as part of the safety assessment of the preliminary design of the 2400 MWth GFR. The aim of the PSA study was to help identify design improvements in order to fulfil deterministic criteria as well as to reach a risk level comparable to the generation III reactors.

Considering the results obtained with a preliminary Level 1 PSA model, it emerged that an increased reliability of the DHR function in high pressure conditions (not corresponding to a LOCA) was suitable to reduce the overall core damage frequency. On the other hand, some small break LOCA situations were not adequately mitigated according to the line of protection deterministic method. Both issues have been solved by design improvements.

In addition, this final Level 1 PSA model, characterized by success criteria based on transient calculations performed with the CATHARE2 code and performed in a perimeter extended to all representative internal initiating events at full operating power, permitted to propose design evolutions that did not increase significantly the CDF. In the same time, those evolutions enabled the DHR system to increase its redundancy level as required in the deterministic approach. Finally, a modified design was reached implying a more extended coverage of various accidental situations by means of a progressive DHR operating strategy.

During the GoFastR project, a set of sample studies were to be performed to confirm the appropriateness of the PSA methodology and identify potential safety shortfalls in the design. These studies will pay particular attention to passive and innovative safety systems and include the following steps:

- identify key safety systems and events
- define success states
- perform reliability analysis for key systems
- identify any requirements for additional information
- develop basic fault/event trees

The results of any deterministic analysis available, including the transient analysis performed during the FP6 GCFR project and the transient and severe accident analysis intended for the FP7 GoFastR projects, will be used to inform the PSA development.

4. Current System Development Status

The GoFastR project that was the FP7 Euratom contribution to the Gen IV gas cooled fast reactors was concluded in February 2013. It was mainly concerned with the technical and safety challenges posed by the GFR system. Important design proposals in particular related to the DHR system were made [15].

After GoFastR, it is envisaged to focus attention on a demonstrator, ALLEGRO (< 100 MWth) without electricity generation as a necessary step towards an electricity generating prototype. An ALLEGRO consortium has been established that has the goal of preparing the construction of this demonstrator. To this end, a roadmap of construction has been prepared, with the main chapters General design, Safety principles, Licensing, R&D, Governance and IPR issues.

The ESNII roadmap foresees that ALLEGRO design will start in 2014, and will be followed from 2018 by construction. Parallel to this, supporting infrastructure and research facilities will help address relevant R&D needs.

5. Conclusions: System's Issues, Concerns and Benefits

The conclusions drawn at the end of the GoFastR project on system issues are the following:

- Innovative SiC fuel cladding solutions were found
- A first design confirming the encouraging potential of the reactor system has been proposed
- Design improvements are nevertheless recommended and interesting tracks have been identified (core & system design, DHR system)
- The GFR requires important R&D efforts to confirm its potential (fuel & core materials, specific Helium technology; molten fuel and clad behaviour)
- ALLEGRO prototype studies are the first step of the further development and are drawing the R&D priorities.

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