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This deliverable provides a publishable summary of the main aims, objectives and achievements of the GoFastR project.

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Executive Summary

GoFastR was a 3-year project aimed at the development of the Generation IV GFR system through its viability phase. 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 burnup. With regard to GFR-specific safety systems, unlike gas-cooled thermal reactors, GFR does not 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.

The activities of the GoFastR project were aligned with those of the Generation IV GFR system and the structure of GoFastR design, safety and fuel work packages were aligned between the Euratom and Generation IV projects. GoFastR also included generic safety studies to enable the Technical Service Organisations (TSOs) in the project to become familiar with GFR technology and make a positive contribution on licensing whilst maintaining distance from developing the design and safety case.

The design and safety work packages were aimed at demonstrating the viability of the basic designs of the reactor unit (including core and fuel subassemblies), safety systems, the balance of plant and the containment concept. The fuel (and other core materials) work package aimed at contributing towards the development of a robust fuel concept in terms of its structure, fissile compound and cladding material. Irradiation of test fuel specimens was beyond the scope of GoFastR, but necessary development work on a capsule design and preparation of the supporting safety documentation for irradiation of a test fuel specimen was undertaken.

GoFastR included liaison with other European FP7 projects and initiatives. The most important of these was the Sustainable Nuclear Energy Technology Platform (SNETP) - GFR features strongly in its roadmap and in the European Sustainable Nuclear Industry Initiative (ESNII), which is proposed as the vehicle via which fast reactor elements of the SNETP roadmap are to be implemented.

GoFastR also featured a dedicated education and training work package which aimed to coordinate training of masters and PhD students and stage formal courses on GFR technology.

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1 Introduction

This deliverable provides a summary of the main aims, objectives and achievements of the GoFastR project. It has been compiled using contributions provided by each of the Work Package (WP) leaders.

Fast reactors have the unique ability to be sustainable by not only being able to generate more fuel than they use, but also by being able to burn minor actinides to reduce the quantity and radiotoxicity of nuclear wastes. The latter feature includes the ability of fast reactors to burn not only the minor actinides that they themselves produce but also the minor actinides arising from legacy wastes and thermal reactors in the nuclear park. The Generation IV International Forum (GIF) has identified six systems which merit development to achieve the goals of sustainability, proliferation resistance, economics and improved safety. Of the original six systems, three were fast reactors, the gas-cooled fast reactor (GFR) being one of these. Two others have the potential to be designed as fast or thermal reactors and for one of these the fast reactor variant has now become the preferred option in Gen IV. Hence four, and in the future possibly five, of the six Gen IV designs to be developed will be fast reactor systems. This emphasis on fast reactors in Gen IV is a natural consequence of their superior sustainability credentials.

The sustainability goal has been developed further within Europe through the establishment of the Sustainable Nuclear Energy Technology Platform (SNETP). As well as setting-out a vision for the development of sustainable nuclear energy within Europe, the SNETP has devised a Strategic Research Agenda (SRA) that identifies the priorities for research through which this vision can be realised. In this context, sodium-cooled fast reactors have been identified as the near-term technology that would allow rapid deployment of fast reactors. The SRA also identifies that gas-cooled and the lead-cooled fast reactors could be deployed in the longer-term. Both of these technologies will be capable of operating at higher temperatures than the use of a liquid sodium coolant will allow. As such, high efficiency electricity generation and a wider range of non-electrical applications becomes possible, such as the generation of high quality process heat and efficient mass production of hydrogen. In addition, the harder neutron spectrum improves the transmutation capabilities allowing minor actinides to be destroyed more effectively.

The GoFastR project has focused its attention on the gas-cooled fast reactor (GFR) option with a view to developing the GFR as a more sustainable version of the very high temperature reactor (VHTR), also one of the six Generation IV systems. The design goals for GFR are ambitious, aiming for a core outlet temperature of around 850^{o}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 achieved by refraining from the use of specific plutonium breeding elements.

This project developed the work of two previous projects within the 5th and 6th Framework Programmes respectively (FP5 and FP6). The work within the FP5 GCFR project was aimed at reviewing, preserving and consolidating the GFR knowledge base that then remained within Europe. The FP6 GCFR STREP was much more ambitious and was involved in active development of the GFR system and provided Euratom's contribution to the Generation IV GFR system. The FP6 work followed three parallel tracks of development. First there was the design and safety assessment of the Gen IV GFR system, second there was the development and safety assessment of a small experimental plant called (at the time[1]) ETDR (Experimental Technology Demonstration Reactor). This plant is a necessary stepping-stone to a full-sized GFR and is required to develop the high-temperature fuel required by the latter. The third track of development was concerned with the development of suitable high temperature and fast neutron fuels and core materials.

The FP6 GCFR STREP was strongly aligned with the Gen IV GFR system. The GFR and ETDR design and safety studies aligned directly with the work packages controlled by the Gen IV GFR Design and Safety provisional project management board (now the Conceptual Design and Safety PMB). The fast neutron fuels studies within the STREP aligned directly with the Gen IV GFR Fuel and Core Materials PMB. This alignment continued all of the way down to individual sub-tasks. The STREP contributed strongly to the exploratory and pre-conceptual phases within the Gen IV system studies. GoFastR has continued to provide Euratom's contribution to the Gen IV GFR system, was strongly aligned with the work of the two project management boards, and has continued to develop the system through the "viability" phase which concluded at the end of 2012. In parallel, conceptual design and safety studies for ALLEGRO were developed so that it can serve as the most representative irradiation facility for qualification of GFR's high-temperature fuel.

There are two main challenges facing the GFR system. The first is the requirement for a fuel that can operate continuously at high-temperature and high power density whilst maintaining good dimensional stability, retaining fission products and having adequate margins to failure in postulated fault conditions. The second challenge is to be able to cope with loss of flow and loss of coolant faults. The low thermal mass of the coolant coupled with the low thermal mass of the core (when compared with a gas-cooled thermal reactor) means that the fuel temperature rises very rapidly following a loss of coolant flow through the core. As such, highly efficient and reliable shutdown and decay heat removal systems are required. Further, if these systems are passive, then the inherent safety characteristics of the system are improved. In pressurized conditions, with helium as a coolant, there is enough gas density for decay heat to be removed by natural convection. However, in depressurized conditions, following a loss of coolant accident, natural circulation is no longer effective and a blower-driven forced flow is required. The current reference concept encloses all of the reactor primary circuit within a further pressure boundary, the guard containment, to ensure that the primary circuit does not blow-down fully to atmospheric pressure. This concept reduces the blower power required to be within the capacity of a battery supply for a period of at least 24 hours, after which, natural convection will be capable of removing the remaining decay heat.

The objectives of the GoFastR project were to address the main challenges to the viability of the GFR system. Clearly, fuel development and primary system studies have featured heavily, but more detailed issues such as being able to design and predict the performance of an all-ceramic core and fuel handling operations with such a core are also significant challenges. Other areas of investigation have included the performance of the shutdown, decay heat removal and guard containment systems, which are all central to achieving a safety performance as good as, and hopefully better, than current 3rd generation systems. In addition, probabilistic safety assessments have been studied within the context of the Gen IV GFR system to indicate whether there are sufficient provisions within the system and to quantify the safety performance to allow comparison with existing systems and other Gen IV concepts. The project has also performed some preliminary severe accident studies to start to assess the progression of accidents leading to and beyond core melt. In this context, new techniques for modelling how ceramic cores will degrade in such accidents have also been investigated. Qualification of the most widely used thermal hydraulics, neutronics and

[1] Since the FP6 GCFR STREP, the ETDR has been renamed ALLEGRO

transient analysis tools, education and training and the investigation of generic safety issues are other important areas of work covered by the project. In addition, to the technical programme of work, GoFastR has devoted considerable effort to the task of forging, strengthening and maintaining strong links with other related projects and initiatives, including other Euratom projects and Generation IV. This report provides a brief summary of the work performed and the main achievements of the GoFastR project.

2 Work package overview

GoFastR was a 3-year project consisting of eleven work packages. A brief description of each of these work packages is presented below.

Work Package 1.1 – GFR conceptual design
This work package aimed at progressing the designs of the GFR core, primary systems and specific safety systems to the point where the viability of the system can be established. The core studies undertaken included neutronics, thermal hydraulics and core mechanics. The latter studies are particularly important to demonstrate that a practical ceramic core can be produced and will be sufficiently robust to withstand handling and operation in a commercial power reactor. GFR primary system studies developed the concepts for the power conversion system and for the primary pressure boundary together with exploration of alternative concepts for the pressure boundary.

Work Package 1.2 – ALLEGRO conceptual design
This work package aimed at developing the conceptual design of an experimental demonstration reactor. Emphasis was given to the design and analysis of three potential cores required within ALLEGRO – a conventional metal-clad MOX fuelled starting core, a transitional core in which a limited number of modified GFR-type ceramic fuel elements will be irradiated and, finally, a high temperature (all-ceramic) GFR demonstration core.

Work Package 1.3 – GFR safety studies
This work package developed the GFR safety approach and performed extensive transient analyses to assess the safety of the system in a wide range of different scenarios. Probabilistic safety analysis was undertaken for GFR to develop the reliability requirements for the safety systems. Development of severe accident analysis methods for predicting the progression of degraded core scenarios was also undertaken.

Work Package 1.4 – ALLEGRO safety studies
This work package was analogous to WP1.3 but specific to the ALLEGRO cores and systems.

Work Package 1.5 - Methods development and qualification
This work package aimed at developing and qualifying core neutronics, thermal hydraulics and transient analysis methods. Code to code and code to experiment comparisons were made using different codes and specifically commissioned thermal hydraulics and system performance experiments.

Work Package 2 – Fuel and other core materials
This work package addressed some of the key issues on fuel and other core materials needed for the deployment of ALLEGRO and its test assembly positions. In addition the necessary development work on an irradiation capsule design and preparation of the supporting safety documentation for irradiation of a test fuel specimen was undertaken

Work Package 3 – Links with other Euratom activities
This work package covered management activity dedicated to establishing links between the different consortia working on advanced reactor concepts within Euratom. This included links with other FP7 projects, with other Euratom initiatives such as the Sustainable Nuclear Energy Technology Platform (SNETP) and the European Sustainable Nuclear Industry Initiative (ESNII) and collaboration with the Central European ALLEGRO Consortium.

Work Package 4 – Euratom representation within Generation IV
This work package covered management activity associated with Euratom's contribution to the Generation IV GFR System Steering Committee (SSC) and Project Management Boards (PMB). This included attendance at SSC and PMB meetings as well as updating the system research plan and project plans and addressing actions arising from the committees.

Work Package 5 – Coordination
This work package covered the project management activity which includes hosting the project website, managing and maintaining the website, routine management, chairing progress meetings, preparing of management reports and production of the coordination deliverables.

Work Package 6 – Education and training
A specific education and training work package was included in GoFastR, with the goals of providing project placements for undergraduates, Master's and PhD students within the participants' organisations and the provision of formal training courses/workshops to attract a wider audience of students and researchers.

Work Package 7 – Generic safety studies
This work package was aimed at carrying out a set of licensing and generic safety studies arising from the use of technologies currently favoured for GFR. The technical objectives were to assess the acceptability of GFR of safety concepts in terms of current licensing requirements, to identify the most important transients for safety demonstration and to carry out independent safety studies related to topics relevant to the GFR concept.

3 Work Performed, Results and Achievements

3.1 WP 1.1 GFR Conceptual Design

3.1.1 GFR core studies

A preliminary design of a 2400 MWth self-sustaining core with carbide fuel pins and SiC cladding was developed early in the project with the purpose of enabling the various parties to work on the same core configuration as part of studies focusing on the fuel concept. In addition, a complete neutronic specification of the GFR core was produced.

A scoping study on GFR penetration in a nuclear park was performed using the ORION fuel cycle modelling code to analyse three fuel cycle scenarios:

(i) an all-PWR AP1000 and EPR reactor fleet,

(ii) as (i) but with the addition of 5 GFRs phased in gradually while the PWRs are being phased out followed by a period where these 5 GFRs are allowed to become self-sustaining. An additional 2 GFRs are then introduced fuelled by the remaining PWR-sourced Pu,

(iii) as (i) but with the addition of 7 GFRs phased in after the PWRs are closed down.

It was demonstrated that GFRs can be integrated into an existing modern PWR fleet, with the Pu for the initial GFR (U,Pu)C fuel charge coming from reprocessed PWR fuel. The results also show that GFRs could be used to lower the amount of minor actinides in a fuel cycle. The fuel manufacturing requirements for typical operating scenarios have been quantified and the decay heats and radiotoxicities of the spent fuel determined.

Core neutronics and thermal-hydraulics calculations were performed on the GFR2400 design by a number of partners. All derived parameters show good agreement between the studies of the GoFastR project, including k-effective, β_{eff}, neutron generation time and temperature profiles for the average and the hottest fuel pin.

A Computational Fluid Dynamics (CFD) investigation of the ceramic pin core sub-assembly was performed, as well as thermal-hydraulic analysis of one inner pin with six spacer grids and two support grids. This work generated detailed results, including pin wall temperatures within spacers and fluid axial velocity.

A study investigated GFR 2400 MWth core performance and uncertainties. The following neutronic parameters of the reactor concept were determined: the k-effective, Doppler coefficient, depressurization and core expansion reactivity effects, power distribution, fundamental and effective delayed neutron fractions, as well as the neutron generation time. Simulation of an open fuel cycle was also performed. Cross section uncertainties were propagated to most calculated quantities and the effects of geometrical uncertainties were investigated.

In general very good agreement was achieved for the calculated parameters using different methods, code systems and cross section libraries. Acceptable correlation was found in the uncertainties as well, although two parameters were shown to exceed their target uncertainty values, underlining the need for better nuclear data measurements.

A study on GFR core analysis and transmutation capabilities assessment was performed covering:

- Neutronics and burnup analysis of the GFR core containing additional plutonium and minor actinides (MA) originating from pressurized water reactors.
- Assessment of the transmutation capabilities and fuel utilisation of the GFR integrated in a symbiotic nuclear energy system of thermal and fast reactors
- Precise core characterisations and assessment of GFR MA burning capacities

A new approach was developed for the modelling of the reactor burnup, using a multidimensional regression method to determine one-group cross-sections as functions of the core composition based on the results of numerous core calculations with different isotopic compositions. With the help of the generated composition -dependent cross-section functions, a fast and flexible burnup calculation scheme was developed and verified, which can be easily integrated into fuel cycle simulations. The burnup model was applied for the analysis of fuel cycle scenarios including a mixed fleet of GFRs and conventional and MOX fuelled LWRs, considering several recycling strategies. It was found that the GFR is close to self-breeding and only about 3% of its Pu load needs to be fed from LWR spent fuel in equilibrium. In case of recycling of MA in the GFR, the MA content in the GFR core reaches equilibrium at about 2.36% of the total actinide content. If more MA is fed into the GFR an optimal value can be found where the GFR consumes both the Pu and the MA production of the LWR park. MOX recycling in LWRs deteriorates the fuel utilisation efficiency and MA transmutation capabilities since the GFR has better potential in Pu utilisation.

The assumption that fast reactors charged with some moderated Dedicated Assemblies (DAs) could supply the high neutron flux and high neutron capture cross-section required for the implementation of the transmutation process was also investigated. DAs are filled with MA and are inserted in the active core with the aim of optimising and improving the reactor burning capability.

Placing DAs in central and in peripheral zones allows an acceptable delayed neutron fraction. In the study, different peripheral configurations were analysed to maximise reduction in MA amounts. Furthermore two different reprocessing options were studied: the standard and the disposal options. In the former, MA are reprocessed immediately after the cooling pool, while in the latter a longer storage time is allowed for spontaneous radioactive decay (about 30 years is forecast) before reprocessing takes place. As expected, the disposal option is a preferable solution to reduce ^{244}Cm mass.

The analysis of results indicated that the case with 20 DAs each containing 6kg MA with long decay time was more suitable from the whole core burning capability point of view.

3.1.2 GFR primary systems and balance of plant

A study of candidate technologies for balance of plant (BOP) components was performed which summarised the status of power conversion systems for GFR at the start of the GoFastR project.

A critical review of primary system and BOP arrangement provided the following main conclusions:

- Components design in a 3 loop configuration with one component (blower, intermediate heat exchanger (IHX) and steam generator (SG)) per loop is a very challenging option.
- Considering 2 blowers, 2 plate IHX and 2 SG per loop is still a challenging design while representing a strong impact on general installation and, particularly GoFastR containment.

- The R&D challenge on component design should be compared to general impact on containment and overall complexity to support the best design compromise.

Various design studies investigating power conversion system and heat exchanger technology were performed. Key findings were as follows:

- Compared to any Brayton cycle equivalent, coupling the gas-cooled high temperature reactor to a traditional steam plant achieves high efficiency cycles whilst maximising the use of low risk, well understood technology.
- An indirect Combined Cycle Gas Turbine (CCGT) arrangement can achieve a high efficiency whilst remaining a comparatively low risk solution. With such an arrangement, a gas turbine (GT) is required to step down the high reactor outlet temperature to one which matches current and projected boiler designs and steam turbine temperature limits.
- The isolation of the GT within an intermediate loop provides a safer overall system, and one which facilitates a smaller primary containment boundary, whilst affording access to the GT for any maintenance. Using a mix of 80% nitrogen and 20% helium within this loop gives rise to turbomachinery close to current aerospace standards, but does however imply a large IHX.
- Maximum plant efficiencies are achieved in any CCGT arrangement by maximising the load on the more efficient Rankine steam cycle. A reheat loop and feed-heat systems, common within the UK AGR (Advanced Gas Reactor) fleet and modern coal-fired power stations, are proven ways to increase overall cycle efficiencies.
- The principal risk with the power conversion system remains the IHX, which provides a lifing challenge at such high temperatures. Current standards of helium circulator also present a risk with respect to their current low capacity and rating. Very little is known about the effects of helium embrittlement on materials within the primary and secondary loops, and this is likely to require further research.

A design study to understand the design challenges of any IHX for operation in a high temperature Gas-Cooled Fast Reactor concept was performed and concluded that a Printed Circuit Heat Exchanger (PCHE) is the most likely technology to provide a compact solution suitable to the GFR concept. It is however, noted that the low maturity of the technology presents risks in terms of achieving acceptable service life and recommends further design investigation.

An opportunity for improving the compactness of any IHX was identified and explored based on the internal plate design. The total module area required for GFR would require dividing up and packing into a modular format. Onward development of this would be required particularly with respect to stress analysis and associated welding technology. Finally, in recognition that attaining an adequate IHX service life is an economic issue, the lifetime costs of any IHX were explored and an example of how these could be minimised by design was presented.

Optimisation of alternative power conversion cycles was performed, which concentrated on the following issues:

- Obtaining the optimum power cycle for the gas-cooled fast reactor of the GoFastR project. On the basis of the cycle efficiencies attained, the cycle with the best results was the recompression with double High Temperature Regeneration.

- Selecting the optimum Supercritical Carbon Dioxide (S-CO2) cycle configuration which best suits the core inlet and outlet temperatures of GFR. Thermodynamic analysis and comparison of the simple Brayton, precompression, recompression, and partial-cooling cycles have been performed. Effects of turbine inlet temperature and pressure ratio on the cycles' performance have been investigated, with focus on general impacts on power plant design. The most promising S-CO2 power conversion cycle for GFR seems to be the partial-cooling cycle, since it features the highest intermediate heat exchanger temperature difference which is close to the GFR temperature difference. High specific net work of this cycle will result in low flow rate of CO2 minimizing frictional pressure losses. Furthermore, this cycle provides the highest efficiency at elevated temperatures. At lower temperatures recompression cycle may be preferable because it reaches slightly higher efficiency than partial-cooling cycle and has less components and simpler design; however its drawback is sensitivity to pressure ratio change. If the cycle complexity is an issue the precompression cycle may be a good compromise at high temperatures because of its high thermal efficiency, stability and wide IHX temperature difference. Hence the partial-cooling cycle, recompression and precompression cycle are recommended candidates for GFR and future work should focus on detailed analysis of them.
- Optimising the coupling of S-CO2 cycles to GFR. For current GFR parameters, a coupling of candidate S-CO2 power conversion cycles to GFR was assessed in terms of thermodynamic analysis. Due to wide temperature change across the reactor which does not fit directly to S-CO2 cycle, this task was split into three cases. First case aims to utilise whole GFR temperature range in single S-CO2 cycle. It was found that only precompression cycle is capable of reaching this goal, however cycle thermal efficiency is sacrificed in order to extend temperature difference across Intermediate Heat Exchanger (IHX) and maximum efficiency achieved was 41%. In second case the reactor heat is split into two S-CO2 cycles connected in a cascade – one uses high temperature heat and the other "low" temperature heat. Four most promising combinations of candidate cycles were investigated. They are: partial-cooling cycle + partial-cooling cycle (P-C+P-C), partial-cooling cycle + recompression cycle (P-C+Rec), precompression cycle + partial-cooling cycle (Pre+P-C) and precompression cycle + recompression cycle (Pre+Rec). Thermodynamic analysis has shown that the first cascade combination (P-C+P-C) excels among the others with average efficiency 50.5% (based on total produced power and input heat). The third case assumes that part of the high temperature heat is delivered to high temperature technology and therefore turbine inlet temperature is lowered. For this case only the secondary side inlet temperature of IHX is fixed and the best thermal efficiency for each cycle is sought by optimisation of pressure ratio. The partial-cooling cycle scores the highest efficiency 48% at turbine inlet temperature 600°C.

Alternative design options for the GFR pressure vessel boundary were examined, together with an overview of the options for the use of prestressed concrete reactor vessels (PCRV) or prestressed cast iron reactor vessels (PCIV). International experience with these systems was reviewed and outlines of alternative pressure boundary design options were considered. The study concluded that the PCRV is still an open option, but needs further R&D as, for example, codes and standards need to be updated.

An indirect combined cycle gas turbine layout power conversion system is thought to be the most suitable to the Gas Fast Reactor environment in terms of safety and technology risk. Investigation of the recommended cycle concluded the following:

- Coupling a high temperature reactor to steam plant achieves high efficiency cycles whilst maximising the use of lower-risk, well understood technology.

- Limits on the temperature capability of steam plant technology mean that an intermediate loop with a gas turbine is required to “step down” high reactor outlet temperature to that which matches current steam cycle boiler and turbine material limits.
- These limits are being improved by research within the fossil fuel-based energy industry which can be used to produce a naturally higher efficiency cycle.
- The presence of the intermediate loop provides a safer overall system, minimizing the size of the nuclear island, and permitting the use of a nitrogen-helium blend as the working fluid. Although reducing the heat capacity of the flow, this gives rise to GT components within current aerospace technological understanding.
- The inclusion of a reheat and feedheat system provide a well established means, based on AGR and coal fired industry heritage, of increasing overall plant efficiency
- The efficiency of the indirect CCGT plant is maximised by maximising load on the inherently more efficient Rankine steam cycle.
- The principal technology risk with the GoFastR power conversion concept is the design of the Intermediate Heat Exchanger. The Primary loop presents challenging design conditions and a requirement for compactness. Further work is required to understand material technology and practical constraints in the design of the IHX.
- Further high risk items include the Helium Circulator and the Active Magnetic Bearings (AMBs) and associated cooling systems.
- It is currently not well understood how components in the primary and secondary loops will react over time in a helium environment owing to the effects of embrittlement. It is recommended that more research be conducted in this area in the future.

3.1.3 GFR system studies

Provisional conclusions of the safety analysis have confirmed the encouraging potential of the reactor system. Nevertheless, some situations require a quick switch from main to DHR loops and the failure of this primary circuit reconfiguration cannot be totally excluded, in the light of the Fukushima accident.

Significant improvements of the design robustness are possible using the principle of a primary circulator coupled to a secondary gas circuit turbo-machine. Even in case of primary circuit depressurization, the core cooling could be ensured for several hours, thus giving a large grace time to succeed the switch to DHR loops at low decay heat power. The reliability of the DHR loops could also be improved using the principle of an autonomous turbo-machine on one of them. This quasi passive system would complement the natural circulation in the low pressure range.

3.1.4 GFR specific safety systems

This task reviewed candidate technologies for the DHR, focussing on some promising principles and checking their viability. Overall, the DHR strategy adopted in GCFR during FP6 seems applicable.

The pre-design of the DHR system was completed, considering specific issues such as IHX compactness for integration in the guard containment, large DHR blowers operation range (from 70 to 10 bar or less), reliable valves/check-valves and helium natural circulation capabilities for some situations. The review assesses the components of the reference concept, with proposals and recommendations for design improvements.

The DHR strategy is mainly based on two DHR levels:

- The first level takes into account the forced convection regimes.
- The second level is necessary in case of failures of DHR gas blowers.

The overall DHR system may appear complex but this is mainly due to a minimisation objective (mainly cost minimisation) that induces the use of 2 (those capable of natural convection) over 3 DHR loops (designated X) and of one DHR loop Y for both DHR levels. Feasibilities were assessed and compared on the basis of several designs; some assistance is required in case of malfunction. This potential external assistance also allows verification of the passive shut-off device closure/opening. Comparison of technologies according to pre-defined requirements showed that split wafer disk check-valves and wafer swing check-valves are preferred and their feasibility in high diameters seems achievable. Beyond classical concerns due to material behaviour submitted to high temperature levels, the primary DHR heat exchangers seem the most critical component due to the need for large dimensions.

This study also aimed at proposing a functional preliminary design of the Decay Heat Removal Exchanger He/water called HX#1 of the loops X. The design proposed in this study enables definition of a loop X with three helical He/Water heat exchangers included in a vessel. This loop X is able to remove 72 MWth (Decay heat level equal to 3% of Nominal Power).

The DHR blowers seem feasible according to the requested electrical power. An issue for the gas blowers type 1 is the capability of the blowers to operate under a wide range of situations: pressurized (7 MPa) to low pressure (0.4 MPa) situations, with one, two or three operating loops. Design point of one blower is preferred at half of the minimum core cooling mass flow-rate: this is convenient for the 2- and 3-blowers operating cases, at constant rotation speed. For the one-blower case, the rotation speed has to be increased to recover the minimum core cooling mass flow-rate. This strategy has to be further confirmed by dedicated computations. This strategy allows significant reduction of the requested electrical power.

3.2 WP 1.2 ALLEGRO Conceptual Design

3.2.1 ALLEGRO core studies

An initial design at the start of GoFastR was produced for ALLEGRO 75MW cores (MOX + ceramic). A proposal for the design of experimental GFR sub-assemblies to be loaded in the MOX starting core of ALLEGRO was also produced. Once these sub-assemblies are tested successfully, it will be possible to proceed to a whole core with GFR sub-assembly technology.

In addition to this, more details were provided on core material isotopic compositions and calculation assumptions used for the neutronic characterisation of the MOX core, the MOX loaded with 6 experimental GFR S/As and the 100% GFR refractory pin core.

Studies on MOX and ceramic core designs were performed, covering neutronics, CFD studies of S/As and minor actinide burning. Detailed MCNP models of the three ALLEGRO cores have been developed and numerous calculations were performed to determine neutronics parameters, such as control rod worth. Further calculations have provided power distribution for CFD calculations and CFD analysis of the fuel assemblies has been performed.

Evaluations were also performed on the design of experimental S/As loaded in the MOX core (including minor actinides burning) and CFD calculations of the experimental S/As thermal barrier static helium zone.

The results indicate that the thermal shield with the original design geometry is very effective in protecting the wrapper from the high temperature in the assembly. The effectiveness of the thermal shield is mainly due to the poor conductivity of the static helium, which is used as a filler gas in the thermal shield. Increasing the width of the thermal shield practically does not affect the wrapper temperatures, while it increases the pressure drop in the bypass channel.

Inside the thermal shield natural circulation develops in the “static” helium due to buoyancy forces. This mainly affects the temperature distribution of the cold wall of the thermal barrier; leading to higher temperatures at the outlet, but the effect on wrapper temperature is negligible.

The U-shaped thermal barrier needs to be reinforced by baffles and various baffle designs were studied. Their main effect was twofold: on one hand, subdivision of natural circulation cells in the barrier and, on the other, periodic increase of the barrier’s cold wall temperature due to conduction through the baffles. It was found, in particular, that inclined baffles slightly reduce the convection and the conduction with respect to horizontal ones. The calculations show that the layout of baffles produced insignificant temperature change in the wrapper, while the number of baffles slightly increased it. These results indicate that it is preferable to reduce the number of baffles to the minimum needed for the mechanical strength of the thermal shield.

An alternative S/A concept was developed, based on the idea of an hexagonal tube made of SiC plates held together within a metallic skeleton made of collars at different levels connected together with tie rods. A high temperature resistant alloy would be needed for the tie rods but they could be cooled by a helium bypass if necessary. The collars could also have a function of contact pads between adjacent sub-assemblies.

Such a hexagonal tube could be used for the MOX feeding core and the experimental GFR sub-assembly, thus allowing a progressive transition from the MOX core to a full GFR technology core. More detailed studies need to be performed with realistic material properties.

A specific ALLEGRO MOX fuel pin performance analysis was performed using the TRAFIC fuel performance code. The analysis confirmed that the low rating and burnup of the MOX starting core had no significant impact on the safety and integrity of the limiting pin during normal operation and the transient conditions modelled (small and large break protected LOCA, protected LOFA and unprotected LOFA). The study confirmed that, from a fuel performance perspective, the ALLEGRO MOX core is safe to operate in normal operating conditions and in the transients modelled.

The concept of a third level shutdown system (TSD) was investigated. The small size of the ALLEGRO core and the requirement to host a range of experimental subassemblies during the initial period of operation places significant constraints on the design of the TSD. Two concepts were initially explored in some detail: A pebble-bed based DSD system, in which a diverse absorber release mechanism provides a TSD capability, and a lithium-6 based system, in which lithium-6 is driven into the core by a pressurised cover gas. Issues identified with these two TSD concepts motivated development of a second, more diverse, pebble-bed based TSD system, in which absorber beads are fed into the DSD assemblies via a modified hollow drive rod.

Criticality analyses on the MOX core and related radiation damage on the reactor was completed. The results from detailed core analyses were compared to results obtained with more approximate methods. This provides information about the importance of the reflectors included in the heads and feet of the sub-assemblies. Radiation transport analyses were carried out to evaluate the radiation damage to be expected in the lower diagrid. The radiation streaming through the gas ducts of the sub-assemblies appear to provide the main contribution to this damage.

At the end of the GoFastR project, the main viability points on ALLEGRO cores are the following:

- The global approach for the core, with a first MOX core including GFR precursor subassemblies was studied in detail. It appears that the 75 MWth MOX core gives satisfactory irradiation performance with interesting capacities (up to six in-core experimental subassemblies). These fast neutron flux capacities can also be valued for other applications.
- The GoFastR partners in the field of neutronics now have models of the ALLEGRO cores.
- A model of the MOX core has been developed, allowing estimation of the damages on steel structures of the sub-assemblies and on reactor structures. The model confirmed the low doses received on the diagrid.
- The CFD analyses performed on both MOX and ceramic cores confirmed that the modelling of heat conduction in the cladding material (including wire spacer) is important. Thermal exchanges were found within the uncertainties of usual correlations.
- One essential point is also the design of the experimental sub-assemblies which include a thermal barrier enabling to reach, locally in the MOX core, coolant and fuel temperatures of the GFR system. The thermal barrier has been modelled and improvements of the initial design have been proposed.
- The Minor Actinide burning capacity of ALLEGRO was also preliminarily assessed.
- An original sub-assembly concept was imagined which would allow a progressive transition from MOX to ceramic core, but this concept still needs big efforts to be consolidated.
- The ALLEGRO safety demonstration will likely need a third level shutdown system, and a critical review of the various systems which could be proposed in this field has been performed.

3.2.2 ALLEGRO system studies

A critical review of basic key components of ALLEGRO and their applicability to a GFR power reactor was performed.

The preliminary design for ALLEGRO prototype key components of primary and decay heat removal loops have been globally confirmed. An arrangement of main components within the guard vessel has been proposed for GFR, and these design principles can be retained and adapted for the ALLEGRO prototype.

A design of purification circuit for the helium gas coolant of ALLEGRO has been proposed. This detailed assessment is based on high temperature reactors experience feedback and mainly on Chinese prototype HTR-10. The applicability to ALLEGRO has been developed and constitutes a sound basis for further design studies of this system

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The preliminary design of the different primary circuit components, the fuel handling, the secondary water circuit, air cooler, the guard containment, have provided a first global picture of the reactor. A review of the components has generated proposed improvements or alternate designs, particularly for the air cooler and the guard containment.

The preliminary safety analysis focussed on core cooling and shows a good resistance of the system to protected situations, which should be confirmed when the secondary water loops are considered. For unprotected situations, the conclusions are still partial, but the case of ULOF would be manageable. Beyond this, a probabilistic safety analysis in support of the design would allow identification of the weak points in the safety architecture and improvement of the design. Moreover, the Fukushima accident raised the particular concern of combining any initiating event with total loss of external and internal power sources. Although the present design is already well adapted to pressurized situations thanks to the natural circulation provisions, the situation is potentially difficult for depressurised situations. Like for the GFR, significant improvements of the design robustness are possible using the principle of a primary circulator coupled to a secondary gas circuit turbo-machine. The reliability of the DHR loops could also be improved using the principle of autonomous turbo-machine on one of them. This quasi passive system would complement the natural circulation in the low pressure range.

The development of ALLEGRO is necessarily connected to an R&D program on calculation tools validation and specific helium technologies. In particular, the fuel material development (high temperature resistant silicon carbide cladding) and the qualification of safety relevant tools and components appear as high priority needs. The main items and potential facilities able to achieve the R&D program were recalled mainly from the ADRIANA project outputs, but a first reflection on the helium purification system was also conducted.

A summary of the ALLEGRO mission, design and safety was produced, which describes the objectives of the Generation IV GFR system and presents the need for a demonstration reactor ALLEGRO as a necessary first to step to fulfilling these objectives along with a summary of the current status of the design and safety analysis. In addition to the review of the mission and the technology, the siting, licensability and environmental issues associated with establishing ALLEGRO as a research facility in Central Europe were explored.

3.3 WP 1.3 GFR Safety Studies

3.3.1 GFR safety approach and risk minimisation

The GFR Safety Approach was developed, covering the safety objectives and principles of safety classification, approaches to safety demonstration and assessment of hazards, safety issues specific for the GFR as well as safety requirements to the system as a whole and in particular to the shutdown system as well as to the guard containment.

GFR risk minimisation studies analysed a number of scenarios, including 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.

The use of water as a working medium in the secondary side of the decay heat removal loop could lead to an accident scenario in which steam and/or liquid water ingresses into the reactor core. According to the analysis performed, the use of refractory metals as materials for the cladding liners (to improve the leak tightness of the fuel claddings against diffusion of fission gases into the coolant) results in a significant neutronic penalty in normal operation conditions. However, the accidents in which spectrum thermalization is expected, e.g. steam or water ingress accidents leads – in spite of the increasing neutron production rate – to a net negative reactivity insertion, because of increasing absorption rate by the refractory metals. This observation gives a basis to study in the future the injection of the borated water into the primary system as one of the possible ways to effectively remove decay heat in the most disadvantaged GFR plant condition.

According to the analysis performed for large water/steam ingress into the primary cooling circuit of GFR, depending on the transient, a power excursion should be avoided due to the negative reactivity of the Control Rod Absorbers (at least 10 $) in case of protected transients, as long as no liquid water enters into the core. Based on the obtained results, the unprotected transients inducing large steam ingress into the primary cooling system should be considered as practically eliminated by design.

Analysis showed that each Control Rod is inserted within a specific sub-assembly called CSD (control and shutdown device) or DSD (diverse shutdown device). The sub assembly itself includes a static part (spike, hexagonal tube, reflector, shielding, central tubular pillar inside reflector and shielding, head) and a moving part composed of the absorber pin basket and the rod follower which moves through the central tubular pillar. The travel of the absorber pin basket is limited at the bottom by the central pillar and both are designed to be resistant to accidental shocks. So the rod ejection event is excluded by design for GFR, regardless of what happens to the Control Rod Drive Mechanism (CRDM).

Concrete vs. steel vessel option assessment showed that Pre-stressed Concrete Reactor Vessels (PCRV) have already been built and operated in gas cooled reactors in the past. There are certain potential advantages, as well as certain drawbacks of the PCRV. The weak points of PCRV are the penetrations (failure of joints) and risks related to the loss of concrete cooling (by loss of insulator or water cooling). So the PCRV option includes a large number of design issues that still need to be demonstrated to be solved for the GFR. Overall, from the safety point of view, no significant advantage of using PCRV is clearly visible considering past GFR projects experience.

On the other hand, assessment does show that an incident exceeding the design parameters, leading to a failure and bursting of the pressure vessel, has manifold and profound consequences for core geometry, as well as for surrounding structures like pipelines, control rod actuators etc. The event of a bursting pressure vessel is a safety-related and nearly non-controllable situation. Therefore, it is argued, a prior objective of plant design should be to physically prevent such incidents in any circumstances. This objective could only be reached reasonably by implementing a fully-integrated pre-stressed design concept. The safety-related amenities of a fully-integrated pre-stressed system were presented based on the THTR-300 and partly the HTR-500 experience. Safety features of a fully-integrated pre-stressed design concept structurally exclude the explosive bursting of the pressure vessel with a fast pressure release of the primary system within a few seconds. A result of this structural design is therefore an enormous time-related advantage in the case of a failure with pressure release in the primary system, which drastically improves the controllability of such an incident. Apart from the pressure vessel, other pressurized systems like helium reservoirs, hydraulic control rod compartments, etc. should be considered as they represent an additional source of safety endangering incidents due to the fact that these systems are partly operated under much higher pressures than these of the primary system.

Dynamic analysis of guard containment integrity under different depressurization scenarios showed that in depressurization scenarios of the GFR primary helium loop the released high pressure gas results in pressurization in the guard containment. The pressurization might endanger the integrity of the confinement. The dynamic transient in the gas volume of the confinement was studied, with an injected source term from the primary cooling loop, by using the CFD code GASFLOW. The system parameters like pressure, temperature, flow fields, gas species distributions were recorded and are presented as results of the performed simulations. Based on the obtained results, a maximum design pressure was defined for the confinement.

Results of the analysis of the preventive DHR system architecture to exclude core destruction showed that for decay heat removal purposes all three analysed decay heat removal strategies - using “AFW” systems, using DHR loops in forced convection, or using DHR loops in natural circulation – can be used for decay heat removal at the GFR-2400 plant, as long as certain operating conditions are fulfilled.

3.3.2 GFR transient analysis

After benchmarking the models developed for the GFR-2400 system, using various system codes, a list of initiating events was developed and extended transient analysis of various situations was conducted. Summarised below are the main sequences assessed:

- Protected loss of flow accident with three pony motors
- Protected loss of flow accident with one DHR loop in forced convection
- Protected loss of flow accident with one DHR loop in natural convection
- Protected loss of heat sink with three DHR loops in forced convection
- Protected loss of heat sink with two DHR loops in forced convection
- Protected loss of heat sink with one DHR loop in forced convection
- Protected loss of heat sink with three DHR loops in forced convection + one main loop remaining open
- Protected secondary circuit break with two pony motors
- Protected secondary circuit break with one pony motor
- Protected secondary circuit break with two pony motors + one main loop remaining open

- Protected large-break loss of coolant accident with three pony motors
- Protected large-break loss of coolant accident with two pony motors
- Protected large-break loss of coolant accident with two DHR loops in forced convection
- Protected large-break loss of coolant accident with two DHR loops in natural convection
- Protected small-break loss of coolant accident with three pony motors
- Protected small-break loss of coolant accident with two pony motors
- Unprotected internal break with three main motors
- Protected large-break loss of coolant accident + containment failure with three pony motors
- Protected large-break loss of coolant accident + containment failure with three DHR loops in forced convection
- Protected large-break loss of coolant accident + containment failure + blackout with one DHR loop in natural convection and nitrogen injection
- Unprotected loss of flow with three pony motors
- Unprotected transient overpower with 400 pcm insertion
- Protected blackout with one DHR loop in natural convection
- Protected large-break loss of coolant accident + blackout
- Protected small-break loss of coolant accident + blackout
- Unprotected large-break loss of coolant accident + blackout
- Unprotected large-break loss of coolant accident + containment failure
- Unprotected loss of heat sink with three main motors
- Unprotected overcooling with three main motors

The results of the conducted analysis confirmed that the current GFR system design relies on the active protective measures (SCRAM, valves, blowers). Decay heat removal under pressurized conditions (70 bar) can be performed under natural circulation conditions. Decay heat removal at back-up pressure (~7 bar) requires battery-driven blowers, unless back-up pressure is increased by means of heavy gas injection, in which case one can rely on natural convection. In case of depressurization to the reactor building pressure (2 bar) the natural circulation cannot guarantee the decay heat removal even in the case of the heavy gas injection. Forced convection using pony motors or DHR blowers should be provided in this case.

3.3.3 GFR probabilistic studies

A probabilistic framework for risk assessment was defined, that supports and complements the overall safety approach for GFR. It recommended that initial PSA models are developed early in the design process, using an iterative process to increase complexity and level of detail as the design evolves. Specific GFR/Gen IV issues were identified that require further consideration, such as methods for incorporating innovative design features in probabilistic models and improvement of the knowledge of accident phenomena and modelling within PSA. Recent proposed approaches for safety and/or PSA for Generation IV concepts were considered, including the GIF Integrated Safety Assessment Methodology (ISAM) and the Level 2 PSA best-practice guidelines from FP7 project ASAMPSA2.

The approach to be taken for reliability analysis as part of the PSA was also defined, particularly for the various passive and/or innovative safety systems. A method to manage the various factors affecting system performance was proposed, including a Component Failure Rate Database (CFRDB) building approach based on an analogy between components and their environment with regards to related information on similar components in existing databases. It was concluded that the available large experimental feedback seems not to be directly useful for the GFR application and existing CFRDB are inadequate for use with the innovative GFR design.

A preliminary reliability assessment of the Natural Circulation Decay Heat Removal was performed using the RMPS (reliability evaluation methodology for a passive system) methodology developed as an FP5 project.

3.3.4 GFR severe accident studies

The work within this task was divided into two parts:

- Identification of potential severe accident scenarios for GFR and preparation of models for the partners' computer codes, to enable the assessment of different aspects of severe accidents.
- Final preparation of models and assessment of different aspects of severe accidents.

One study focussed on the uncertainty quantification of accident scenarios using polynomial chaos (PC) techniques. These enable an efficient calculation of the uncertain output parameters (e.g. the maximum fuel or cladding temperature during a transient/accident) arising from the uncertain inputs (e.g. power peaking factors, delayed neutron data, uncertain thermal-hydraulics data, etc.).

Another study defined a new method of solving the compositional multi-phase flow equations. It was shown how bounded volume fractions and compositional phase fractions result and that the phase summed saturations and component summed (for each phase) unity constraints were naturally enforced. The overall aim was to enable severe accident modelling to be performed and, with the use of self-mesh adaptivity, to focus the numerical resolution on areas of the domain that most need it.

3.4 WP 1.4 ALLEGRO Safety Studies

3.4.1 ALLEGRO safety approach and risk minimisation

Preliminary safety analysis was performed on the ALLEGRO demonstrator equipped with two main loops. The reasons behind the design evolution towards two loops architecture were detailed in the frame of a risk informed analysis mixing deterministic elements to probabilistic simplified insights. Results regarding the assessment of this configuration were presented for several types of transients:

- Pressurised transients (loss of flow (LOFA), loss of off-site power (LOOP));
- Depressurisation transients (break on a main loop (LOCA)).

This work represented the first stage of study for the 75 MWth ALLEGRO reactor under accident conditions. In this configuration, the secondary water loop was modelled in a very simplified manner using boundary conditions. The analysis illustrated the benefit expected from the use of the main loops although these results have to be reinforced with calculations performed with a more complete input deck and on a broader set of situations. These results were used to improve the ALLEGRO design and the cooling strategy during the GoFastR project and beyond.

3.4.2 ALLEGRO transient analysis

Transient analysis was performed for ALLEGRO covering a number of important protected and unprotected accidents. The aim was to analyse the system safety behaviour at the current design stage, to examine relevance of the foreseen plant protection parameters and to identify the main threats to component integrity under transient conditions.

After benchmarking the codes against each other for several steady-state and transient cases, a number of important initiating events were analysed, including small and large break loss-of-coolant accidents (LOCA), loss of flow (LOF), loss of heat sink (LOHS) and reactivity insertion. The following conclusions can be derived:

- The analysis of the protected LOHS showed that the list of safety signals should be extended. A new safety signal was introduced to avoid boiling and the corresponding pressure increase in the secondary circuit. The pressure in the primary circuit was found to go below the secondary pressure in several protected cases. This phenomenon should be revised because of the possible water ingress from the secondary circuit. A possible way to avoid this scenario is the blow down of the secondary circuit. In this case new safety signals should be introduced. The different cases demonstrated that – with the addition of the new safety signal – the LOHS transient fulfils the design criteria, even taking into account the worst aggravating failures.
- A protected LOF event can be accommodated by the ALLEGRO reactor under pressurized primary system conditions without any safety concerns assuming battery power is available to operate the pony motors of the primary blowers.
- A protected LOOP event can be accommodated under pressurized primary system conditions without any safety concerns, assuming battery power is available to operate the pony motors of the primary blowers. Even in case of an aggravating event of failure of startup of one pony motor, according to the analysis the maximal cladding temperatures remains below the transient criteria.
- A protected small-break (3-inch) LOCA can be accommodated without serious problems provided that the backup containment pressure is 3.6 bar and that 3 DHRS loops are available for decay heat removal in forced convection, while a protected large-break (10-inch) accident coinciding with containment failure results in a severe accident consequence.
- During the unprotected loss of flow accident the fuel pins will retain their geometric integrity, but the clad of the peak power fuel pins will most likely lose their leak tightness releasing fission gases into the primary coolant early in the accident. The other safety concern is that high core outlet temperatures may challenge the mechanical integrity of the upper core structures.
- A protected station blackout can be accommodated under pressurized primary system conditions without any serious safety concerns if 3 DHRS loops are available for decay heat removal in natural convection.

3.4.3 ALLEGRO probabilistic studies

Safety and reliability assessment of the system designed for the ALLEGRO decay heat removal was performed, to identify potential safety shortfalls in the design. Failure probabilities were calculated on various system configurations and integrated with probabilities of occurrence of corresponding hardware components and natural circulation performance assessment.

The analysis suggested a need to improve measures against Common Cause Failures, in terms of an appropriate diversification among the redundant systems, to reduce the risk associated with the system malfunction.

3.4.4 ALLEGRO severe accident studies

A model was set up, including geometrical and material arrangement and thermal hydraulic and neutronic modelling and the associated input data development of ALLEGRO models using the SIMMER code. Effort was made to make the code applicable for steady-state and for transients near operating conditions to provide reasonably accurate initial conditions for severe accidents analyses.

An unprotected large break LOCA was chosen for the main severe accident transient analysis. SIMMER calculation results showed that the power excursion can take place, which leads to core melting and degradation and releases thermal energy up to 10 GJ within 10 sec. After the core melting, the fuel relocates at the lower part of core and the bottom of the vessel, leading to a configuration that is preliminarily evaluated as sub-critical. Therefore no further re-criticality is expected within the investigated time scale.

3.5 WP 1.5 Methods Development & Qualification

Qualification of the CFD codes used for evaluation of the subassembly and components design was performed by comparing the calculational predictions with measurements obtained in the experiments performed at the L-STAR air-cooled facility, allowing visualisation of the radial velocity profiles near the heat exchange surface.

Evaluation of the methodical uncertainties which are determined by the features of the gas-cooled fast reactor was also performed. Alongside this, the impacts of the uncertainties in the nuclear data of the integral parameters such as multiplication factor or reactivity coefficients were also analytically evaluated.

Finally, the system codes used for transient analysis of GFR and ALLEGRO system behaviour were validated. Use was made of the unique experimental information measured in steady-state and transient conditions at the helium loop located in Italy.

3.5.1 Core thermal hydraulics

Benchmark activities were performed based on gas cooling experiments performed with the L-STAR/SL facility at the Institute of Neutron Physics and Reactor Technology (KIT-INR). For L-STAR/SL a new heating rod was designed, analysed, and installed, which fulfils the geometry and power data requirements for ALLEGRO better. The rod allows investigating different surface structures starting with radial ribs.

Experimental results obtained with the air loop L-STAR were compared to results obtained by CFD calculations by several partners. The CFD calculations were made based on defined geometry and setpoints in parallel to the experiments ("blind" simulation). In general the agreement between experiment and calculations was found to be acceptable, especially taking into account that the calculation results were obtained “blindly” before the experimental results were released. Discrepancies between experiment and calculation are attributed mainly to meshing, in particular of the lower test section inlet, the choice of the turbulence model and the modelling of radiation.

3.5.2 Core neutronics

For the 2400MW GFR, uncertainties of the main neutronic parameters as a result of nuclear data uncertainties were evaluated. The parameters mainly studied were the multiplication factor k_{eff}, Doppler and helium depressurization effects, control/safety rod efficiency and reactivity loss in the course of the fuel cycle. The basis for the new analysis was the original GoFastR concept with carbide fuel pins and silicon-carbide ceramic cladding, which was developed and proposed in the first quarter of 2009. Clarifications of the original layout description were accounted for to prepare new inputs for this data uncertainty re-evaluation.

The main conclusions from the updated study confirmed that for the majority of the safety parameters considered, the calculated uncertainty is below the target uncertainty that has been defined for Generation IV reactors. However, the target uncertainties are not always reached. This implies that the present unbiased uncertainties on nuclear data should be studied further and reduced e.g. through adjustments based on available experimental data.

Similar studies were carried out for both the MOX-fuelled steel-clad ALLEGRO core and the carbide-fuelled ceramic pin ALLEGRO core. The results indicated that the uncertainties of the reactivity effects being considered were smaller than the targeted values for Generation IV systems, with the exception of helium depressurization. The fact that the uncertainty of the coolant void effect is found just a little larger than the target Generation IV value of 7% is of no concern for these gas-cooled reactors, given that the reactivity insertion induced by a sudden full depressurization of the primary system is quite small, at less than one β_{eff} (~400pcm).

Conversely, the calculated uncertainty in k_{eff} is clearly significant as far as the details of the final designs are concerned, indicating a need for further in-depth studies. In particular, the present unbiased uncertainties on nuclear data should be analysed and reduced e.g. through adjustments based on available experimental data. In addition, the use of a complete data set, such as the one currently being prepared in the framework of the Advanced Fuel Cycle Initiative would certainly be beneficial in this context.

It is also noted that the present studies of neutronic uncertainties did not account for the propagation of microscopic cross-section uncertainties to the burnup dependent heavy nuclide concentrations, which will certainly have an impact on the uncertainty of the reactivity swing over cycle. Further research will be required to address this aspect.

3.5.3 Transient benchmark

Initial work concentrated on preparation of the steady-state and transient benchmark specifications based on the experimental data obtained at the Brasimone HE-FUS3 helium rig. The data from seven steady-steady tests were analysed with various system and CFD codes.

A detailed description of the HE-FUS3 loop – a helium cooled facility located at Brasimone Research Centre in Italy – was issued to support the benchmark modelling. The experimental data made available and analysed for the benchmark consisted of:

- seven steady-states at different conditions, useful to assess the thermal-hydraulic models,
- loss of flow caused by compressor slow-down and loss of flow caused by bypass valve opening, useful to assess the transient modelling capabilities of the available codes.

All the codes participating in the exercise showed an acceptable capability to predict the behaviour of the loop both in steady-state and transient conditions. Nevertheless, the results of the benchmarking analysis allowed several recommendations to be made regarding the modelling of different aspects of the gas system dynamics. These recommendations were taken into consideration in the transient analysis studies for GFR and ALLEGRO.

3.6 WP 2 Fuel and Other Core Materials

3.6.1 Design studies for ALLEGRO test element

Benchmark studies were done on the ceramic fuel design using two different fuel performance modelling codes. The modelled fuel rod consisted of uranium-plutonium carbide fuel pellets in a SiCf/SiC (silicon carbide fibre-reinforced silicon carbide composite) cladding. The differences in the fuel performance results obtained by the two different computer codes pointed out the importance of the fuel modelling, especially concerning fission gas release (FGR) and fuel swelling.

Parametric tests suggest the high sensitivity of the simulated fuel swelling to the input data, as well as the dominating impact of the gap size on the simulated fuel centre temperatures. In order to assure cladding integrity for fuel pins with a ceramic-based cladding, pellet-clad mechanical interaction (PCMI) must be avoided entirely. Hence, fuel swelling can impact strongly on the fuel burnup limit and must therefore be very well understood and accurately modelled. Because of the strong impact on burnup limit, this would appear to be one of the key issues in the design of GFRs.

A benchmark against experimental data is strongly recommended for a better assessment of the models treating FGR and swelling as well as of the overall fuel pin behaviour which is essential for future studies concerning the design analysis of the ALLEGRO test element. The present studies did not consider the multi-layered nature of the cladding, which implies that more complete modelling will be necessary for a representative assessment of the actual design.

3.6.2 Preparation for an irradiation test of design concept in an MTR

A separate effect test on carbide fuel has been proposed to investigate the carbide fuel swelling. In this test, small fuel discs with the microstructure of the target GFR fuel (15-20% porosity, mostly open) would be irradiated at well-defined temperatures, to investigate the fuel swelling at these temperatures.

The capsules have been designed in such a way that small fuel discs would be placed in a heavy metal alloy which produces significant heat. In this manner, the temperatures can be kept steady with little influence of the decreasing power due to burnup. Two positions in the High Flux Reactor (HFR) at Petten have been considered, in which the burnup target will be met in approximately 12 and 17 HFR cycles respectively, with cycles being 28 days on average. The accelerated irradiation would necessitate a power density of around 2 to 4 times greater than that which would be anticipated in a GFR, depending on the irradiation position. More representative power densities, of 25 to 50 W/g, would lead to irradiation times of over 4 years before reaching the required burnup. Considering that accelerated burnup would lead to different fuel behaviour, it could be considered to update the irradiation design so as to achieve representative power density. This would lead to low-burnup but readily-relevant results after a year. Higher burnups would then be achieved in a follow-up irradiation project. This might be a sound choice considering the hazard of relying on results obtained in non-representative conditions.

The temperatures have been studied in detail for the high power position, and the design has been optimised to keep the average temperatures within ±50 °C for the entire duration of the irradiation. For the other position, with lower power density, only initial studies have been performed, but the required temperatures could also be reached in this position. However, further optimisation would be required, if this position were to be preferred.

No significant issues were identified regarding the feasibility of the irradiation test, in either the neutronic or thermomechanical assessments of the test design. Final approval however can only be given when the exact irradiation is planned and the actual core loading is considered. This was beyond the scope of the GoFastR project but it is hoped that the work will be progressed to the next stage in a future project.

3.6.3 Material alternatives: sourcing and assessment

An extensive literature search was performed to investigate alternative cladding materials for the ceramic-based fuel design. Advanced cladding materials will be required for the higher temperature cores of ALLEGRO, which are cutting edge for both nuclear and non-nuclear applications. An evaluation has been conducted for these materials of the mechanical, thermophysical and neutronic properties as well as to consider their technology readiness and availability.

This evaluation has concluded that a SiCf/SiC composite material is the most promising cladding candidate. Development of SiCf/SiC has proceeded rapidly in recent years through the use of advanced fibres, interphases and matrix forming techniques. This has yielded improvements in porosity and joining as well as the stability of the structure and mechanical properties under high temperature irradiation.

However, significant further optimisation of SiCf/SiC will be required to produce cladding that is acceptable for use in ALLEGRO fuel. Two of the most important development needs are retaining good thermal conductivity under irradiation and maintaining leak tightness of the cladding to the helium coolant and mobile fission products including at the joints. Erosion by the coolant also needs further investigation.

Vanadium alloyed with chromium and titanium appears to be the best available back-up choice for ALLEGRO cladding, but this would be at the expense of significantly limiting the operating temperature compared with SiCf/SiC.

Both SiCf/SiC and vanadium alloys would require supporting technology in order to operate in the ALLEGRO environment and the key requirements for supporting technology have been identified. For SiCf/SiC, the most promising would be a tantalum barrier to maintain leak tightness. For vanadium alloys, surface alloying with chromium or a multi-layered silicide coating would be required to inhibit oxidation. However, data for these technologies is limited and therefore significant further development and testing will be required. The optimisation of tantalum alloy composition for best performance under irradiation is of particular importance. A buffer layer of porous carbon is also a possibility in order to relieve potentially damaging stress concentrations in the event of PCMI.

3.7 WP 3 Links with other Euratom Activities

Throughout the course of the GoFastR project there have been interactions with other FP7 projects. These are summarised below:

- CP-ESFR – Exchange of physical property data on carbide fuels. A report on carbide fuel data was compiled within the FP6 GCFR STREP. A request was received from the Coordinator of CP-ESFR for this report to be made available to the participants of Sub-Project 2.2 within that project. The report was provided willingly with no request for anything in exchange. An attempt was made to try to coordinate education and training activities with CP-ESFR but both projects encountered problems either with the timing or with the lack of common ground on the subjects proposed. Hence this ambition did not come to fruition.
- LEADER – An attempt was made to coordinate education and training activities with the LEADER project, specifically on fast reactor physics.
- ADRIANA – Information was provided to ADRIANA with regard to the experimental needs of the GFR project and also to provide details of experimental facilities within the project that could be made available to other reactor projects.
- ARCHER – Collaboration took place in establishing a materials and components programme within ARCHER that would support simultaneously the development of the VHTR and GFR systems.
- SARGEN-IV – The Steering Committee of the GoFastR project agreed to supply a number of the restricted project deliverables to SARGEN-IV to allow the latter to assess the safety of GFR.
- ALLIANCE – Many discussions with the ALLIANCE project members took place and some ALLIANCE partners were accepted as associate members of the GoFastR Consortium. In December 2012 the GoFastR Steering Committee voted to donate the entire written output of GoFastR to the ALLIANCE project.
- SUSEN – Discussions were held in Rez with members of the SUSEN project to identify experimental needs to demonstrate the helium technology associated with various GFR primary systems.

In addition to the forging of links with other FP7 projects, there has been much activity by the Coordinator of GoFastR concerned with the establishment of the European Sustainable Nuclear Industrial Initiative (ESNII). Presentations on the progress with the GFR system were made at two meetings of the ESNII Task Force. The Coordinator was appointed to represent the GFR system on the ESNII Executive Board and has acted in this capacity at the regular meetings of this committee. Much work was undertaken in providing input into the ESNII Concept Paper and the ESNII implementation plan, and on providing input into the SET-Plan, particularly with regard to establishing key performance indicators (KPI). A sub-group of the ESNII Task Force which included the GoFastR Coordinator was established to formulate a consistent set of KPI in collaboration with SETIS and representatives from the European Commission's Directorate General for Energy. The aim of this sub-group was to generate KPI that were consistent across the various fast reactor systems and consistent across the different European Industrial Initiatives (EII), which were also providing input to the SET Plan. A joint presentation on the status of the Gen IV GFR system together with the status of the ALLEGRO project given by the ALLEGRO Consortium was presented at the 1st ESNII conference in June 2012.

The ALLEGRO Consortium was formed in the Spring of 2010 through the signing of a Memorandum of Understanding (MoU) between the research institutes in the Czech Republic, Hungary and Slovakia. The aim of the MoU was for ALLEGRO to be built as a research facility within one of the three countries with the other two providing support with the design, development and construction. GoFastR has supported this consortium in a number of ways, such as providing technical information, accepting ALLEGRO Consortium members into the GoFastR project as associates and providing specialists at two ALLEGRO safety meetings. The first of these meetings was an ALLEGRO Safety seminar held in Bratislava on 3 November 2011 at which GoFastR was represented by the Coordinator jointly with specialists from CEA. A further meeting was held at MTA-EK in Budapest on 28-29 March 2012 to work through the implications of the Fukushima accident on the safety case for ALLEGRO. The Coordinator together with two other specialists from GoFastR and four specialists that were joint with CEA, together with the European Commission's Project Officer for GoFastR attended this meeting to present the work on ALLEGRO transient analysis and measures introduced to deal with combinations of extreme external events.
Collaboration with the ALLEGRO Consortium has been excellent and it is hoped that this collaboration will be rebuilt in the frame of the Horizon 2020 programme.

3.8 WP 4 Euratom Representation in Generation IV

A number of activities were undertaken in GoFastR in conjunction with the JRC in representing Euratom in fulfilment of Euratom's obligations within the Generation IV International Forum (GIF) GFR System Arrangement. These activities involved representing Euratom at the GFR System Steering Committee (SSC), the Conceptual Design and Safety Project Management Board (CD&S PMB) and at the Fuel and Core Materials Provisional Project Management Board (FCM PPMB). Combined, they also contributed to the work of these boards through the joint production of system and project documents and publications, through the contribution of project deliverables. Furthermore, in 2010 the Coordinator of GoFastR was elected to be the SSC Chair on behalf of Euratom.

The GIF activities required to be undertaken by the Coordinator of the GoFastR project increased significantly following election as the SSC Chair. These activities included liaising with the GIF Experts Group (EG), the GIF Policy Group (PG), the cross-cut working groups, most notably the Risk and Safety Working Group (RSWG) and the Proliferation Resistance and Physical Protection Working Group (PRPPWG). The SSC Chair is responsible for accounting for the activities of the SSC and its PMBs through preparing contributions to the GIF annual reports, preparing the annual GIF cost statements and through preparing publications both external and internal for the triennial GIF symposia. Moreover, SSC Chairs take part in the interface meetings held once each year in Vienna between the GIF and the International Atomic Energy Agency's (IAEA) Innovative Reactors programme (INPRO).

3.9 WP 5 Coordination

This work package covered the project management activity associated with overall coordination of the GoFastR project. This included routine project and consortium management, such as organizing and chairing technical progress and steering committee meetings, preparing periodic and final management reports and producing any coordination deliverables identified in the Description of Work. It also included hosting, managing and maintaining the project website.

3.10 WP 6 Education and Training

The education and training activities have centred on providing project placements for undergraduates, Master's and PhD students within the participants' organisations and providing formal training courses/workshops.

The general objective of the Education and training work package was to prepare BSc, MSc, PhD students and post-docs for their scientific work in the project and to provide a good basis for their career in the nuclear field. In total 26 students contributed to the project. The supervision of students by senior researchers involved in the project will result in 6 BSc, 13 MSc and 10 PhD theses related to GFR technology. 10 out of these 29 degrees are to be earned in the following years, thereby ensuring also the continuation of GFR-related work at the partner institutes. Initially four universities in four different countries were directly involved in this work package and a fifth university joined later. Of course, in universities the supervision of student projects is part of the normal activity but WP6 also aimed to provide education and training to an audience wider than that of the universities within the consortium. As a result, 8 of the 26 students were affiliated to one of 5 non-university partners. It is concluded that the Education and Training work package of the GoFastR project achieved its goal to positively encourage the development of young scientists and engineers via the supervision of students.

In addition to providing placements for students, this work package also organized two training workshops, both of which were very well attended, with approximately 50 participants at each event. Both events received very positive feedback from the attendees.

The first event was held in Cadarache (France) in July 2011 and covered a wide range of topics including fast reactor physics, thermal-hydraulics, power conversion options, accident analysis, as well as materials and fuel-cycle issues associated with the technology of the GFR system. Essentially, all the key issues arising in the development of Gas-Cooled Fast Reactors were covered, by lecturers well known in their specific fields of expertise. The course gave a comprehensive insight into the specific features and advantages, as well as the challenges of the innovative Gas-Cooled Fast Reactor. The 3 days course also contained workshops on two specific computer codes widely used in the nuclear community.

The second event was held in Rez (Czech Republic) in December 2012 and was open to all participants and students from the GoFastR project as well as to participants of other FP7 projects currently running in the EU. The program of this second event was more focused on the specific work and achievements of the GoFastR project and covered all the technical work packages, giving a nice overview of the progress made during the three years of the project. The presentations, which were given by the work package leaders and other leading scientists in each work package, were very well received.

Based on the feedback, it can safely be concluded that the workshops were a success. Participants gained valuable knowledge about Gas Cooled Fast Reactors, learned about past and present GFR programs, the latest developments and the current, most pressing issues. They were able to widen their insight into research areas outside their own fields of endeavour and showed great interest in the lectures. The vast majority were satisfied and considered the courses useful for their studies and work. The workshops also provided the attendees with good opportunities to have direct contact with the international experts presently involved in GFR research. It is the firm belief of the organizers that the GoFastR workshops were highly valuable events which fulfilled expectations and achieved their goals.

3.11 WP 7 Generic Safety Studies

The tasks performed in WP7 were addressing two main issues: safety approach and selected safety studies. The WP7 partners reviewed the safety approach issued by designers' WPs and selected safety studies were dedicated to specific safety cases of which the Technical Services Organizations (TSOs) wanted to outline the importance in the licensing of a GFR concept.

3.11.1 Safety approach

Details of technology neutral aspects were extracted from the Technical Guidelines (TG) written for the construction of the third generation of PWRs. It was shown that many high level safety objectives and requirements indicated in the TG for the PWR concept appear to be applicable to other concepts and might provide valuable guidance for the preliminary design and assessment of the GFR.

Another objective of this work was to review the GFR safety objectives and preliminary options in light of those technology neutral elements. The safety approach and preliminary design architecture proposed within the GFR viability report generally followed the most important recommendations of the TG dealing with the general safety objectives, the defence in depth principles and the consideration of external hazards. The safety approach includes also some of the recent recommendations issued within the safety assessment of the EPR™, and which complete the TG. On this point, the main feedback of the implementation of the TG in the frame of current development of nuclear plants (GEN III reactors), is to outline the issues of PSA use, systems diversity assessment and comprehensive integration of the external hazards in the safety demonstration. In the frame of the Fukushima accident analysis, the emphasis shall also be put on common mode failure assessment and response to "beyond design basis" rare and severe sequences.

A list of incidents, accidents and hypothetical sequences was drawn up to be used in the safety demonstration of the GFR, based on analogy with existing lists for a SFR (RNR-1500 project), the HTR-module project and the EPR™. This was reviewed against a similar list drawn up in WPs 1.3 and 1.4.

3.11.2 Fast depressurization of the primary circuit

A fast depressurization transient was performed to evaluate its potential consequences on the core mechanical stability. Three breaks were postulated with an equivalent hydraulic diameter of around 10 inches. One of the breaks corresponds to the control rod mechanism failure on the bottom of the main vessel.

Fast depressurization involves significant dynamic oscillations for about 2s, of amplitude smaller than 0.2 MPa. Dynamic strains deduced from pressure integration over the surface of the internal vessel are similar for all scenarios and very limited in value. These results were used as input data to perform a time-history computation of the dynamic response of the internals. The dynamic comportment of the vessel was also investigated based on eigen modes computation.

Globally, the stresses in core supporting structures and the vessel remained in the elastic domain showing sufficient safety margins. The rigidity of the predesigned diagrid, into which subassemblies are locked at the base of the core, showed a good capacity to withstand the dynamic loadings induced by the fast depressurization.

3.11.3 Fission product release from carbide fuel

Carbide nuclear fuels are an interesting alternative to oxide nuclear fuels, in particular for fast reactors. A study, focussed on UC, PuC and (U,Pu)C, investigated the phenomenology leading to fission product (FP) release in oxide fuel. The objective of the study performed was to investigate the potential similarities between the phenomena already identified for oxide fuel and those expected to pilot the FP release in carbides. A synthetic review of the available data in the literature on uranium carbides and FP was done. Then these data were compared to the same results on oxide fuels to highlight the qualitatively similar behaviour of oxide and carbide fuels.

A main aim of this work was to prove that oxide fuel mesoscale models can be adapted for carbide fuels; the question being how to achieve such an adaptation with reasonable experimental efforts. Exploratory work has been launched to demonstrate that this approach is indeed valid, based on electronic structure calculations, molecular dynamics simulation and/or thermodynamic equilibrium computations.

It appears that the evolution of the microstructure and the behaviour of FPs in carbides, at least from the phenomenological point of view, are rather similar to those observed for oxides.

It is likely that the models developed for oxide fuels can be largely used for the treatment of the carbides behaviour because they constitute the right physical basis to do so. The study pointed out the lack of some important data to adapt the mesoscale models developed for oxide fuels to carbides and proposed a roadmap to achieve this adaptation. Some of the missing data which are absent from the literature may be the object of further investigations with the tools available.

3.11.4 Fission products behaviour in the GFR primary circuit

It is important to the GFR development strategy to take benefit from the advanced maturity of VHTR concepts and the significant operating experience with experimental or demonstration HTRs/VHTRs.

Both the GFR and the VHTR concepts use helium as primary coolant and aim to generate high coolant temperatures; they have, therefore, similar materials requirements and component technologies. The purpose of this study was to examine the adaptability of primary-system FP-behaviour modelling developed for HTRs to the GFR primary circuit in accident conditions, through a review of available information on FP behaviour in HTR primary circuit and a comparison between materials options for both reactor concepts.

The study highlighted that while materials may be more-or-less the same, there exist major differences influencing the outcome of FP release into the primary system beginning with the FP source and the bulk-gas composition in severe accident conditions not to mention the considerable influence of large amounts of graphite in VHTRs. Methods used for determining FP distributions in HTRs in the past, essentially empirical/semi-empirical, are very probably inappropriate for Gen IV systems, either VHTR or GFR. In particular, sorption data for new alloys and graphite are lacking and there will be a long lead time before an adequate database is created for the materials being investigated.

The general-purpose methods and codes (1D and CFD) being used for thermal-hydraulic analyses of VHTR systems are suited without extra adaptations to GFR concepts. On the other hand, the detailed analysis of coupled core neutronics and core thermal hydraulics requires specific modelling that diverges for the two Gen IV technologies (and even between the pebble-bed and prismatic cores of VHTR designs). Lastly, more-integral codes, generally developed initially for LWR-accident analyses, have been applied in recent years to analysis of VHTR concepts. With the specific enhancements required for such VHTR analyses, no further requirements specific to GFR analyses would seem to be necessary. Concerning applications to industrial processes using the high-quality heat that GFR and VHTR concepts can provide, the safety analysis of such systems will be very substantially similar and very challenging. Again, nothing specific to GFR concepts would seem to be required beyond what is already required for VHTRs.

3.11.5 Investigation of the potential for local core damage

For a gas cooled fast reactor the comparable low material heat capacity for decay heat is a crucial point if assessing a potential overheating of fuel rods and subsequently local core damages. Additional coolant flow resistances would stress this point even more and were investigated.

The effects of small blockages of coolant flow between the fuel pins were investigated. The requested design parameters for the steady state condition (e.g. power density distribution, pressure drop and the axial and radial peaking factor) could be reproduced nearly exactly by the corresponding calculations. Values for the maximum fuel and cladding temperatures were calculated which slightly overestimated the design values. Variations of the explicitly considered 4 grid spacer layers showed no significant impact on the temperature distribution and the value or position of their maxima.

Based on the equilibrium condition, the most critical regions with respect to a local blockage and the correlated fuel and cladding temperature peaks were identified. Value and position of the maximum temperatures are dominated by the original hot spot regions of the reference case, which are determined by the coolant mass flow and the power density distribution. This distribution is only marginally influenced by the position of the grid spacer layers.

Additional investigations were also performed to determine the effect of partial blocked adjacent regions in comparison to fully blocked single regions. Blocked cross section sizes equivalent to the single blocked region case as well as larger blocked areas were considered.

The calculated maximum temperatures for the local blockage at steady state were put into perspective to the reference design values of the maximum fuel and cladding temperatures. Additional time-dependent calculations for single blocked region cases show that the fuel and cladding heat-up to their respective final maxima will be finished within 3 to 5 seconds, indicating that reliance on external measures is not an option and the material of fuel pin and cladding itself has to resist the peaking temperatures.

3.11.6 Review and assessment of the break size assumptions made on the primary pressure boundary

A review of the break size assumptions of the designers on the primary pressure boundary of the GFR was performed. A review of the available LOCA analyses from the GFR design and the safety categorisation of different LOCA scenarios proposed by the designers were also performed.

It was found that due to continued development of the GFR design since the reviewed analyses have been published, the conclusions of the LOCA analysis might not be valid any more. Therefore the fulfilment of acceptance criteria for LOCA cases in different event categories should be checked taking into account the up-to-date design.

3.11.7 Assessment of the measures to ensure a minimum system pressure for decay heat removal systems

GFRs require active systems for decay heat removal in the event of a loss of primary circuit pressure. The gas pressure in such situations is a key parameter in insuring that a feasible pumping power can provide the required coolant mass flow rate. To keep the required pressure in the case of a failure of the reactor pressure vessel, a guard vessel enclosing the primary vessel, the primary loops and a part of the decay heat removal system is envisaged. The purpose of this task was to assess the measures foreseen to insure that the minimum required pressure could be maintained. Key aspects to be considered in the further design evolution of a guard vessel for the ALLEGRO reactor were compiled.

4 Conclusion

GoFastR was a 3-year project aimed at the development of the Generation IV gas-cooled fast reactor (GFR) system through its viability phase. The design goals for GFR are ambitious, aiming for a core outlet temperature of around 850℃, a compact core with a power density of about 100MWth/m3, a low enough plutonium inventory to allow wide deployment, a self-sustaining core in terms of plutonium consumption, and a proliferation resistant core avoiding the use of specific plutonium breeding elements..

The GoFastR project developed the work of two previous projects within the 5th and 6th Framework Programmes respectively (FP5 and FP6). Like those of the FP6 GCFR project before it, the activities of the GoFastR project were strongly aligned with those of the Generation IV GFR development program and GoFastR has continued to provide Euratom’s contribution to the GFR system. To reflect this, the structure of the GoFastR design, safety and fuel work packages was aligned exactly with that of the equivalent parts of the Gen IV project.

In addition, GoFastR also included generic safety studies to enable the Technical Service Organisations (TSOs) in the project to become familiar with GFR technology and make a positive contribution on licensing whilst maintaining distance from developing the design and safety case. GoFastR has also featured a dedicated education and training work package which has coordinated supervision of masters and PhD students and continued development of young researchers, in addition to staging formal courses on GFR technology to attract a wider audience from across Europe.

For GFR the main technological 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 burnup. With regard to GFR-specific safety systems, unlike gas-cooled thermal reactors, GFR does not 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.

The objective of this project was to address the main challenges to the viability of the GFR system, particularly fuel development and primary system studies as well as performance and fuel handling operations for an all-ceramic core. In parallel, conceptual design and safety studies for a demonstration plant, ALLEGRO (formerly ETDR), were developed so that it can serve as the most representative irradiation facility for qualification of GFR’s high-temperature fuel. The results from the project are presented in 67 contractual deliverables, of which 18 have been identified as being Euratom’s contribution to the Generation IV GFR system.

A brief summary of some of the main achievements and conclusions is given below, categorized according to the area of study:

- **Design** - The project aimed to progress the designs of the GFR core, primary systems and specific safety systems to the point where the viability of the system can be established. The studies undertaken included neutronics, thermal hydraulics and core mechanics, to demonstrate that a practical ceramic core can be produced and that it will be sufficiently robust to withstand handling and operation in a commercial power reactor. GFR primary system studies developed the concepts for the power conversion system and the primary pressure boundary, as well as exploring alternative concepts for the pressure boundary.

Conceptual design studies for the experimental demonstration reactor, ALLEGRO, were also performed, with emphasis on the design and analysis of the different cores required for different stages of the technology demonstration. The global approach for the core, with a first MOX core including GFR precursor subassemblies was studied in detail. It appears that the 75 MWth MOX core gives satisfactory irradiation performance with interesting irradiation capacities (up to six in-core experimental subassemblies). These fast neutron flux capacities can also be valued for other applications.

Numerous other studies in various fields, such as neutronics, fuel pin behaviour and damage to steel structures were also performed, the latter analysis confirming that the ALLEGRO diagrid is subjected to relatively low doses. CFD analyses performed on both steel-clad and ceramic-clad cores confirmed that the modelling of heat conduction in the cladding material (including the wire spacer) is important, with thermal exchanges being found within the uncertainties of usual correlations. One essential point is also the design of the experimental sub-assemblies, which include a thermal barrier that enables them to reach, in the steel-clad MOX core, local coolant and fuel temperatures that are representative of the GFR system. The thermal barrier has been modelled in detail and improvements to its initial design have been proposed. The minor actinide burning capacity of ALLEGRO was also preliminarily assessed and a novel sub-assembly concept was devised which would allow a progressive transition from the steel-clad MOX core to the ceramic core.

- **Safety** - An overall safety approach and probabilistic methodology were developed for GFR, as well as reliability and severe accident analysis to develop the requirements for the safety systems. Risk minimisation studies were also performed, covering issues such as water ingress, rod ejection, concrete/steel vessel, and dynamic analysis of guard containment integrity under different depressurisation scenarios to assess potential risks and provide further design information. Results showed that it would be possible to eliminate a number of risks by design.

  Extended transient analysis was performed for GFR and ALLEGRO covering a number of accident scenarios to analyse system behaviour and the relevance of the current plant protection parameters as well as to identify threats to component integrity. Around 50 initiating events were considered, covering a wide range of representative GFR full power operating conditions, corresponding to about thirty characteristic accidental situations with potential aggravating failures.

  The results of the conducted analysis confirmed that the current GFR system design relies on active protective measures (SCRAM, valves, blowers). Decay heat removal under pressurized conditions (70 bar) can be performed under natural circulation conditions. Decay heat removal at back-up pressure (~7 bar) requires battery-driven blowers, unless the back-up pressure is increased by heavy gas injection, in which case one can rely on natural convection. In case of depressurization to the reactor building (2 bar) the natural circulation cannot guarantee the decay heat removal even with heavy gas injection. Forced convection using pony motors or DHR blowers must be provided in this situation.

  The GFR safety analysis has confirmed the encouraging potential of the reactor system. Nevertheless, some situations require a quick switch from main cooling loops to DHR loops and the failure of this primary circuit reconfiguration cannot be totally excluded, in the light of the Fukushima accident. Significant improvements of the design robustness are possible using the principle of a primary circulator coupled to a secondary gas circuit turbo-machine. Even in the case of a primary circuit depressurization, the core cooling could be ensured for several hours, thus giving a large grace time to accomplish the switch to DHR loops at low decay heat power. The reliability of the DHR loops could also be improved using the principle of an autonomous turbo-machine on one of them. This quasi passive system would complement the natural circulation in the low pressure range.

The ALLEGRO safety analysis showed that the current safety provisions in the ALLEGRO design were adequate in most cases and recommendations for further improvements were provided. For example, the ALLEGRO safety demonstration will most likely require provision of a third level shutdown system and a critical review of various systems which could be proposed in this field has been performed. The ALLEGRO safety studies performed within GoFastR will be continued by the central Europe partners involved in the ALLIANCE project devoted to ALLEGRO.

In addition to the specific safety analyses, a set of licensing and generic safety studies were performed based on the technologies currently favoured for GFR. The objectives were to assess the acceptability of GFR safety concepts in terms of current licensing requirements, to identify the most important transients for safety demonstration and to carry out independent safety studies relevant to the GFR concept.

- **Methods development** - A number of benchmarking and neutronics studies were performed to develop core neutronics, thermal hydraulics and transient analysis methods for gas-cooled fast reactors.

  Experimental results for unheated and heated experiments in the L-STAR facility were compared to results obtained using computational fluid dynamics (CFD) calculations performed by several partners. In general the agreement between experiment and calculations was found to be acceptable, especially taking into account that the calculation results were obtained “blindly” before the experimental results were released. Discrepancies between experiment and calculation are attributed mainly to meshing, in particular of the lower test section inlet, the choice of the turbulence model and the modelling of radiation.

  A benchmark exercise using experimental data from the HE-FUS3 helium loop facility focused on the ability of codes used for gas-cooled system design and safety analysis to simulate seven steady-states and two partial loss-of-flow tests. All the codes participating in the exercise showed an acceptable capability to predict the behaviour of the loop both in steady-state and transient conditions. Based on the results obtained it is concluded that the system codes participating in the exercise are suitable for the thermal-hydraulic and transient analysis of gas systems when the system is subjected to limited variations, not too far from the reference operating conditions. A more general evaluation will require a wider test matrix, including transient conditions for larger variations of the gas physical properties. The study provided several recommendations for the GFR and ALLEGRO transient analysis regarding the modelling of different aspects of the gas system dynamics.

  Uncertainties of neutronic parameters due to a priori nuclear data uncertainties were calculated for ALLEGRO (MOX and ceramic pin cores), as well as for the large GFR2400. The parameters mainly studied were the multiplication factor k_{eff}, Doppler and helium depressurization effects, control/safety rod efficiency and reactivity loss in the course of the fuel cycle. Despite the different physical behaviours, the overall uncertainties appear similar for all three cores analyzed, although further work with improved covariance data is required to substantiate this conclusion. The calculated uncertainties of the reactivity effects considered were smaller than the target values for Generation IV systems, with the exception of helium depressurization. However the latter observation is not of great concern for these gas-cooled reactors because the reactivity insertion induced by a sudden full depressurization of the primary system is less than one β_{eff}. On the other hand, the uncertainty on k_{eff} of ~1.5% is several times the recommended value and is clearly significant for the details of the final designs. The high uncertainty in k_{eff} was found to be mainly caused by uncertainties in the ^{238}U inelastic scattering and ^{241}Pu fission cross-sections. These findings indicate a need for further in-depth studies. In particular, the present unbiased uncertainties on nuclear data need to be reduced.

- **Core materials** - Studies were performed to address some key issues on fuel and other materials needed for the deployment of ALLEGRO and its test assembly positions.

  Benchmark studies were done on the ALLEGRO ceramic fuel design and parametric tests suggest high sensitivity of the simulated fuel swelling to the input data, as well as the dominating impact of the gap size on fuel temperatures.

  Potential materials have been identified for the ceramic-based ALLEGRO fuel design, considering technology readiness and availability. Testing of potential cladding material has been conducted and in parallel an extensive literature review of materials alternatives was undertaken. It is concluded that continuous silicon carbide (SiC) fibres in a SiC matrix (SiC-SiCf) is the most promising cladding material. Development of SiC-SiCf is proceeding rapidly through the use of advanced fibres, interphases and matrix forming techniques. This has given improvements in porosity and joining as well as stability of structure and mechanical properties under high temperature irradiation.

  SiC-SiCf cladding will also require a metallic diffusion barrier to the helium coolant and mobile fission products. Tantalum has excellent resistance to corrosion and high temperatures but it incurs a significant neutronic penalty. An alternative to tantalum is niobium but niobium is less resistant to corrosion than tantalum.

  Significant further optimisation of SiC-SiCf is likely to be required to produce an acceptable cladding. Two of the most important development requirements are joining technology and maintaining good thermal conductivity under irradiation. Further data is needed on the performance of tantalum and on the neutronic consequences of its use.

  Vanadium alloyed with chromium and titanium has been identified as a lower temperature option for GFR cladding. There have been significant improvements in the resistance of vanadium alloys to high temperature irradiation recently but further development is required to optimise and qualify a suitable alloy composition and production route. The main concerns are irradiation-enhanced creep, oxidation by impurities in the helium coolant and embrittlement by irradiation below 400°C. Treatment of the surface may be required in order to give improved corrosion resistance.

  A design for a proposed experiment to investigate carbide fuel swelling under irradiation was also developed and studies were performed to provide initial data required for such an irradiation test. The experiment would involve small prototype fuel discs held in a specially designed capsule being irradiated at well-defined temperatures in a Materials Test Reactor. It is expected that a future project will progress this work to the next stage.

- **Education and Training** - In addition to the normal work of the universities in educating and training young researchers, a specific education and training work package on GFR was devised to attract a wider audience of students and researchers. This work package centred on providing project placements for undergraduates, Master's and PhD students within the participants' organisations and providing formal training courses/workshops. In total, 26 students contributed to the project, leading to 29 degree-level qualifications based wholly or partly on this work. Two training course/workshops were held, covering a wide range of topics concerned with gas-cooled fast reactor technology and the work of the GoFastR project. Both events were very well attended and based on the feedback, it is concluded that they were a success. Participants gained valuable knowledge about gas-cooled fast reactors, learned about past and present GFR programs, the latest developments and the current, most pressing issues.

- **Other activities** - Links were established between GoFastR and several other consortia working on advanced reactor concepts within Europe, including a number of other Euratom FP7 projects. Of particular note is GoFastR's cooperation with the FP7 ALLIANCE project, which will continue the development of the ALLEGRO demonstrator through its preparatory phase. All of the output from GoFastR has been made available to this project to ensure continuity and to maximise the benefits of the GoFastR project to the FP7 programme. In addition, cooperation with the ALLEGRO consortium, which was formed in 2010 with the aim of realising construction of the ALLEGRO demonstration reactor, has been particularly strong. Involvement in other Euratom initiatives such as the Sustainable Nuclear Energy Technology Platform (SNETP) and the European Sustainable Nuclear Industry Initiative (ESNII) have also been important aspects of the GoFastR project. In addition, a number of activities were undertaken in GoFastR to represent Euratom in fulfilment of Euratom's obligations within the Generation IV International Forum (GIF) GFR System Arrangement. This has included representation on the GFR system steering committee, project management boards and various related working groups, interfacing with the International Atomic Energy Agency's (IAEA) International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO), and providing technical deliverables to Generation IV.