SafeG

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D4.1 GFR refractory fuel qualification options

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Version control table

Version number	Date of issue	Author(s)	Brief description of changes made
1.0	16/05/2021	Z. Hózer, E. Slonszki,	First draft
2.0	15/02/2022	Z. Hózer, E. Slonszki, J. Klouzal	The French SFR wrapper tube material was changed for EM10(9%Cr-1%Mo) in the *Introduction* (request by Thierry Beck) The role of fuel supplier described in more details in the *Introduction* (request by János Gadó). Chapter *2.2. V4G4 ALLEGRO fuel design* was extended with information on the current status and design of the fuel for ALLEGRO (request by Jozef Sobolewski). Chapter 4.6 on *VTR* development added (request by Olivier Baudrand). Requirements on the reprocessability of materials in order to close the fuel cycle, and on the capabilities to withstand long term storage of spent fuel were added to TRL1 in appendices D and E. The need for development of reprocessing technology and long term storage solutions is added to TRL 2 in *Appendices D and E*, and the existing reprocessing technology for LWR UOX fuel is mentioned in TRL2 of *Appendix E* (request by Jan Husarcek). The requirement for modelling of the two fuel types in DEC conditions by severe accident codes, and high temperature measurements to investigate ceramic fuel behaviour were added to TRL 3 in *Appendices D and E* (request by Jan Husarcek). *Appendix E* was added in order to cover separately the TRL of UOX pellets in SiC_f/SiC cladding type fuel. The oxide fuel statements from *Appendix D* were removed.
3.0	12/05/2022	Z. Hózer, E. Slonszki,	Final version ready for review of MST and consortium
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CONTENT

1 INTRODUCTION

The main objective of the development of ALLEGRO gas-cooled fast reactor is the demonstration of the operability of a fast neutron reactor with gas coolant. Since the reactor will operate at high temperature and the core components will receive high fast neutron doses, the selection of appropriate fuel materials and the qualification of fuel is a key action in the design and development process and these conditions create great challenges for the development of GFR fuel.

The concept of the ALLEGRO as a European GFR demonstrator unit assumes that the "final" refractory (ceramic) fuel system design will finish its qualification process by the irradiation in several experimental positions in the starting ALLEGRO core(s). The starting ALLEGRO core will consist of already well qualified fuel assemblies. The obvious shortcoming of such idea is that there is no well qualified fuel system for GFR because no GFR has ever been operated (and, as stated above, the fuel system qualification is product specific). The closest option is the reference French SFR design – a bundle of thin steel (15-15Ti) clad rods with UO_2 / MOX fuel in a form of pellets spaced by a helically wound wire within a EM10(9%Cr-1%Mo) hexagonal wrapper tube. From the fuel behaviour point of view the main differences between SFR and GFR normal operating conditions are:

- Higher desired outlet temperatures in GFR, about 850 °C in GFR compared to SFR temperatures up about 620 °C. To overcome this difference, the outlet temperature of the steel clad ALLEGRO core would be reduced in order to fit within the 15-15Ti qualification range. Note that three main phenomena of 15-15Ti behaviour would have to be investigated beyond SFR range to ensure reliable operation at higher temperatures – swelling, creep and fuel – cladding chemical interaction.
- Higher system pressure, about 7.5 MPa in GFR compared to atmospheric pressure in SFR, leading to inward cladding creep in GFR as opposed to outward cladding creep in SFR. Nonetheless, the creep behaviour of 15-15Ti in the SFR temperature range is well known allowing to take this difference into account in the fuel pin design.
- Specifically in ALLEGRO the power density will be lower than in the reference SFRs leading to much lower fuel centerline temperatures and hence reduced fission product migration and release and fuel restructuring.
- Flow induced vibrations will be different in the GFR. The impact of this difference is unknown.
- Coolant-cladding interactions will differ (in case of GFR, the main issue are the impurities in the He coolant)
- The coolant volume in the core compared to the fuel volume is about 50% higher in GFR compared to SFR

Considering above, the idea to use the fuel system based on the reference SFR fuel seems well founded, especially considering its proven manufacturing process including the QA on a semi-industrial scale.

A limited post-irradiation examination of the fuel would still be necessary to confirm the expected behaviour (especially the chemical interactions with the impurities in the coolant). The core power would have to be adjusted to keep the cladding and wrapper temperatures in the SFR range, but several fuel assemblies could be operated at reduced He inlet flow and hence temperatures in order to widen the experience base. The coolant flow rate would also be adjusted in the experimental positions containing the novel refractory fuel assemblies in order to reach the desired temperatures.

However, the analysis of the postulated accidents in the ALLEGRO shows that due to low thermal inertia of the core and relatively low melting temperature of the 15-15Ti, the cladding temperatures are predicted to rapidly approach melting temperature in the DEC scenarios. Even in the DBA scenarios, the margins to clad melting (not to mention the more stringent design criteria) are uncomfortably small considering the novelty of the proposed GFR safety systems.

In case the design studies of the ALLEGRO show that the required levels of the safety are not achievable with the steel clad core, the alternative approach to the qualification of the refractory fuel would have to be taken. Even in case the ALLEGRO starts with the steel clad core, any irradiations of the refractory fuel performed before ALLGERO start in other reactors will greatly facilitate the process.

Today there is no available regulatory guidance to address fuel qualification for GFRs. The GFR fuel designs are outside of the large experience base available for traditional light water reactor (LWR) fuel. For these reasons the qualification of GFR fuel cannot be based directly on the requirements, criteria, experimental and NPP operational experience of widely used LWR fuel.

- The developers of GFR reactor have to prove the applicability of fuel for GFR conditions including both normal operations and accidents. Furthermore, the fuel fabrication technology must be also established at high technical level.
- The regulators are responsible for the protection of public health and safety. They have to specify general requirements and may identify nuclear fuel performance topics that should be investigated. It is generally recognized that fuel qualification is a part of the overall licensing of a nuclear facility. As such, the requirements on fuel qualification are provided by top-level requirements attributed to the nuclear facility [1].

The objective of nuclear fuel system qualification is the demonstration that a fuel product fabricated in accordance with a specification behaves as assumed or described in the applicable safety case, and with the reliability necessary for economic operation of the reactor plant [2]. According to a recent CNRA report fuel qualification for advanced reactor needs to include the followings:

- A defined test envelope based on expected conditions during operational states (normal operation and anticipated operational occurrences) and accident conditions (design basis accidents and design extension conditions)
- An irradiation testing program, which includes full scale integral testing, to identify fuel failure and degradation mechanisms
- Transient testing to assess fuel performance under transient and accident conditions
- A demonstration that the controls on the manufacturing process will deliver adequate levels of reliability.

In more specific terms, the fuel system qualification consists of demonstration that:

- The fuel system provides containment of radioactive substances in the normal operation at the level assumed by the reactor design. *Note that while zero leakage is the goal of the fuel system design, reactor design must conservatively assume some degree of release.*

- Fuel system behaviour is well characterized in all abnormal and accidental conditions. The limits, up to which it provides its fundamental safety functions (contains radioactive substances, allows effective heat transfer to coolant, allows for reactivity control) are well known and are used in the design and safety assessment of the reactor.

The fuel system qualification range is always limited, for example in terms of the temperature, power density, fuel burnup or accumulated fast neutron fluence. A natural process is to start with a qualification just allowing the operation up to design goals and then to extend it using the operational experience.

The fuel system qualification involves not only the investigation of the material properties or general design, but also of the manufacturing process and QA – i.e. the fuel system qualification is product specific, not technology specific.

The experimental program can be defined depending on the technological readiness of the fuel provided by the supplier. In principle, the ALLEGRO fuel as fuel of any reactor has to be purchased from the fuel supplier who is able to satisfy the technical and licensing needs of the reactor operator. Since ALLEGRO is a demonstrator of a completely new technology, it cannot be expected that any potential fuel supplier could deliver the appropriate fuel without further significant efforts. The development of ALLEGRO fuel should be shared between the fuel supplier and V4G4.

The purpose of the present report to review the GFR refractory fuel qualification options and the scientific, technical content of different steps of such a qualification process. **It has to be stressed that this report focuses on those parts of the qualification process which require irradiations. Other important aspects of GFR fuel qualification related to fuel fabrication, establishment of fuel safety criteria, development and applications of numerical models are only shortly mentioned. The key questions, to be answered by this document, are:**

- What irradiation experiments performed in other reactors before ALLEGRO start would **facilitate the qualification of refractory fuel in the experimental positions of ALLEGRO steel clad core**
- What irradiation experiments performed in other reactors before ALLEGRO start **would enable starting ALLEGRO with full refractory core**

2 FULLY REFRACTORY ALLEGRO CORE

The ALLEGRO could start the operation with fully refractory core, if several important steps of fuel qualification were successfully completed in other reactors. Following a historical review of GFR fuel designs, the possibilities of ALLEGRO operation without the first core have to be discussed.

2.1 EARLIER GFR FUEL DESIGNS

A gas-cooled fast reactor has never been built until today, but GFR projects were launched in different countries since the 1960s. Those GFR designs included probably the most diverse fuel types for nuclear reactors [4].

- The first GFR designs were based on the LMFBR experience and included conventional pin-type, stainless steel cladded fuel assemblies with oxide pellets and with roughened external cladding surface to enhance heat exchange.
- Coated particles were considered in some European designs with different geometrical arrangements (e,g. cylinders with perforated annuli or “stack of saucers” geometry). The proposed materials for structural elements were SiC and stainless steel.
- In the Soviet Union chromium dispersion fuel pins were proposed with small inclusions of U metal or UO_2 in a matrix of chromium for the gas-cooled fast reactors with corrosive, dissociating N_2O_4 coolant.
- In Japan coated particle fuel with nitride fuel kernels TiN sealing layers was considered. In one assembly type the coated particles were arranged in an annular bed. The other design featured large prismatic blocks filled with a mixture of coated particles and matrix material (TiN, SiC or ZrC). The material of the structural parts was SiC.

After 2000 new interests were expressed by several countries to develop gas-cooled fast reactor designs following the Generation IV International Forum (GIF) initiative. The imperative of the new development was to reach the SFR performance while maintaining the safety of HTR. Important step of these developments was the design of a small demonstration plant which has subsequently become known as ALLEGRO [4] and which was intended to develop and qualify the innovative refractory fuel system based on two successive core configurations. At first, the standard MOX core with metallic clad would be implemented at moderate temperature in order to irradiate some innovative refractory fuel at full scale. After this preliminary phase, a full refractory core, representative of the GFR, would be implemented.

The refractory fuel candidate concepts included several design versions [6]:

- Coated particle fuels with large kernels and thin coating layers.
- Dispersion fuels in which small particles of the fuel are dispersed in a ceramic matrix.
- Plate type fuel elements arranged within a basket. The basket structural reference material was SiC reinforced with SiC fibers (“honeycomb structure fuel”).
- Conventional pellet-cladding configurations, which would then require a refractory cladding such as SiC, and the means to join and seal the materials.

2.2 V4G4 ALLEGRO FUEL DESIGN

The most recent activities in the development of ALLEGRO are carried out by the V4G4 international consortium. According to the present design:

- The “first core” of ALLEGRO will be built with MOX or UOX fuel in 15-15Ti stainless steel (SS) cladding. These fuel types have been widely used in different sodium-cooled fast reactors, including NPP reactors. The manufacturing and operational experience and the available experimental data provide sufficient basis for application of such fuel, albeit at reduced temperatures compared to desired parameters.
- The refractory fuel for the” second core” of ALLEGRO reactor will be composed of UC or (U,Pu)C pellets in SiC_f/SiC cladding.

For a “starting refractory core” considered in this document, the selection of UC or even U,Pu C might be too ambitious because the operational experience with these material is much more modest in reactor conditions and the qualification process will last much longer compared to UOX or MOX fuel. Therefore the combination of SiC_f/SiC cladding tubes with UOX or MOX fuel is also considered.

Steady-state conditions of the reactor core were summarized in [112]. Parameters for the oxide fuel (MOX) with steel cladding are based on the ESNII+ core specification [113], SafeG deliverable D1.1 [114], and GoFastR deliverable D1.2-1 [115]. Oxide fuel with SiC cladding has not been investigated yet. Parameters for carbide fuel with SiC cladding are based on GoFastR deliverable D1.2-1 [115].

	Oxide fuel with steel cladding [113][114]	Carbide fuel with SiC cladding [115]
Fuel pellet average/maximum temperature	867/963 °C	990/1140 °C
Fuel cladding average/maximum temperature	447.5/562 °C	600/863 °C
Coolant core inlet temperature	260 °C	400 °C
Coolant core outlet temperature	535 °C	800 °C
Primary pressure	70 bar	70 bar

The main dimensions of the latest designs of oxide fuel in stainless steel cladding and carbide fuel in silicon carbide cladding are shown in Figure 1 - Figure 4.

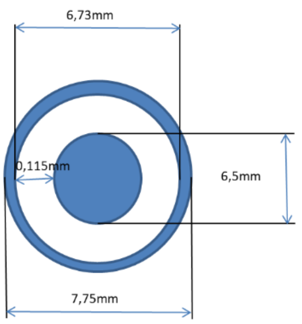

Figure 1: Cross section of UOX fuel with 15-15Ti stainless steel cladding [113]

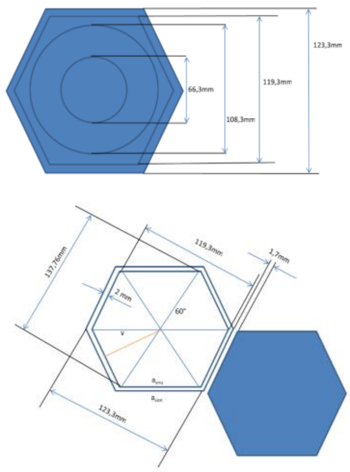

Figure 2: Wrapper dimensions of ALLEGRO fuel assembly with UOX pellets and 15-15Ti stainless steel cladding fuel[113]

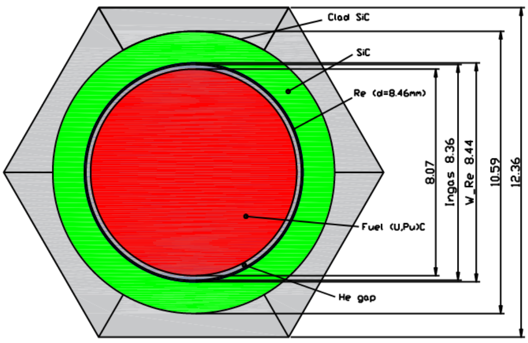

Figure 3: Cross section of (U,Pu)C fuel with SiC cladding[115]

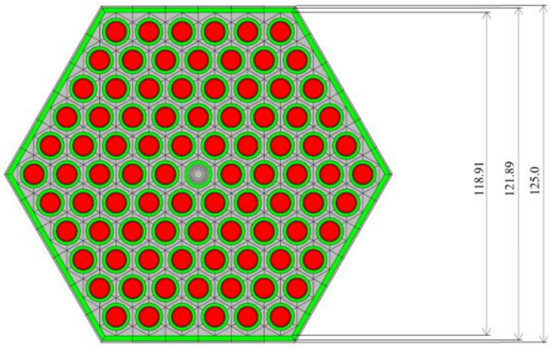

Figure 4: Wrapper dimensions of ALLEGRO fuel assembly with (U, Pu)C pellets and SiC cladding fuel[115]

In order to avoid generalities, this document will focus on the perspective refractory fuel in the following form (See appendices A and B for more information on material selection):

Fuel Assembly structural components	Cladding	Fuel material
SiC spacer elements	SiC_f/SiC composite tube (optionally with a metallic liner)	UO_2 or MOX pellets
SiC wrapper		UC pellets
other components		U,Pu C pellets

The main safety issue of the SFR fuel system in GFR conditions is the low melting point of the steel cladding. Therefore, its replacement with SiC_f/SiC cladding is vital from the safety point of view and is needed to reach the target outlet coolant temperatures of commercial GFR.

While the UC has better thermal properties than UO_2 or MOX, the effect of the reduced fuel temperature on the safety is, by itself, judged to be negligible for low linear heat rates. On the other hand, the swelling rate of UC is higher than that of UO_2. The creep properties of carbide fuel may also be more limiting in the long term pellet-cladding mechanical interaction. Therefore, it would seem natural to first pursue the technologically mature UO_2/MOX variants. However, there are some potential issues which might prevent this option and which has to be investigated:

- Long term chemical interactions of UO_2 / fission products / SiC based cladding at normal operating conditions, i.e. under neutron irradiation
- Short term chemical interactions of UO_2 / fission products / SiC based cladding at very high temperatures under accidental conditions (no neutron flux)
- Pellet-cladding mechanical interaction (PCMI) in power transients, which will be more pronounced with the UOX and MOX due to poor thermal conductivity and associated large thermal strains. Limitations caused by the PCMI could lead to strict burnup limits (operation only up to the hard fuel-cladding contact).

Up to date, the exact qualification process of the ALLEGRO / GFR refractory fuel system has not been put forward. The two extreme options are:

A. The fuel element had not been irradiated under representative conditions at all. Only limited irradiation data exists, providing sufficient confidence that the design is viable.

B. The fuel system had been fully investigated up to the fuel element (pin) irradiations under the representative conditions of normal and abnormal operation in different reactors before ALLEGRO starts. Subsequent testing of the irradiated fuel element under the accidental conditions had been performed and associated safety limits had been established. All fuel assembly structural materials had been irradiated under representative conditions, their properties are well characterized, modelling of in-pile fuel assembly performance is possible. Fuel assembly had been tested in an out of pile loop.

In this case of option A, the core has to be loaded with MOX/SS FAs and some dedicated experimental FAs loaded with few refractory fuel elements, if needed in a capsule design for safety considerations. The whole fuel system development programme would be gradually building on the irradiation experience from the PIE. Some experiments would have to be done in other reactors where on-line monitoring capabilities are available or ALLEGRO would have to be designed to accommodate in-pile measurements of fuel performance. After the irradiation, the pins would have to be taken for testing under accidental conditions. This option requires prolonged operation of ALLEGRO with steel clad cores, possibly over the whole lifetime of the reactor. If the first pins of the refractory design start to be irradiated in ALLEGRO, it would take at least 10 years (3 years irradiation, 2 years PIE + testing, 2 iterations) until the introduction of the first fully refractory FA to the ALLEGRO core. With this option, the fully qualified refractory GFR fuel would not be available in realistic timescales and another options to pursue the refractory fuel design and qualification must he sought.

With option B, the irradiation of the refractory fuel system in the experimental positions in ALLEGRO steel core starts with several refractory assemblies with a fixed, final design and serves only to:

- Confirm the design expectations on the FA dimensional changes under the irradiation
- Qualify the manufacturing and QA process of the fuel
- Possibly investigate several variants of design / manufacturing process on mass scale

In this case, the qualification irradiation of the refractory fuel in steel clad core could take only over two refuelling cycles:

- Irradiate all experimental FAs for the 1st cycle, evaluate the performance
- Unload some of experimental FAs for destructive PIE
- Perform PIE while irradiating rest of the experimental FAs, evaluate performance
- If both PIE and in pile results are satisfactory, perform transition to “refractory fuel core”
- Continue monitoring performance both in-pile and by PIE of the initial experimental FAs, which will be ahead of the rest of the core in terms of the burnup and fast neutron fluence.

Qualification needs for ALLEGRO core with refractory start-up core are schematically illustrated in Figure 5 as function of increasing technology readiness level (TRL). More detailed description of actions needed to reach the individual TRL levels are described in Appendix D – Detailed overview of GFR fuel TRL.

Note that if this state (B) can be achieved, **then starting with a full refractory fuel core is in fact feasible if**

- the safety limits of the refractory fuel system under the abnormal and accidental conditions had been established
- the safety analyses of the ALLEGRO with refractory fuel system had been performed and proven that the limits are met.

The crucial point is that, in order to derive the safety limits, no operational experience of the refractory FAs in ALLEGRO is necessary – the experiments to obtain the safety limits would be performed:

- out of pile on unirradiated material
- out of pile on material irradiated in a different reactor
- by an integral in pile testing is dedicated experimental device in another reactor.

Naturally, **there would be increased economic risk** – if the fuel proves unreliable and starts failing in ALLEGRO core under normal operation, the reactor would have to be shutdown **and the whole core reload worth of fuel would be lost.** In the worst case, the contamination of the primary circuit would exceed the design expectations leading to prolonged downtime and maintenance issues.

In the EU SafeG project it was proposed to review the possibility of ALLEGRO construction and operation without the starting SS clad core(s). On one hand such approach could accelerate the introduction of refractory fuel, but on the other hand it would need additional measurements in other material testing reactors.

Even if it is ultimately decided that as a last step of qualification the irradiation of the refractory fuel in the steel clad core is needed, testing of the refractory fuel in other reactors as much as possible before ALLEGRO is started will shorten its qualification process at least 10 years.

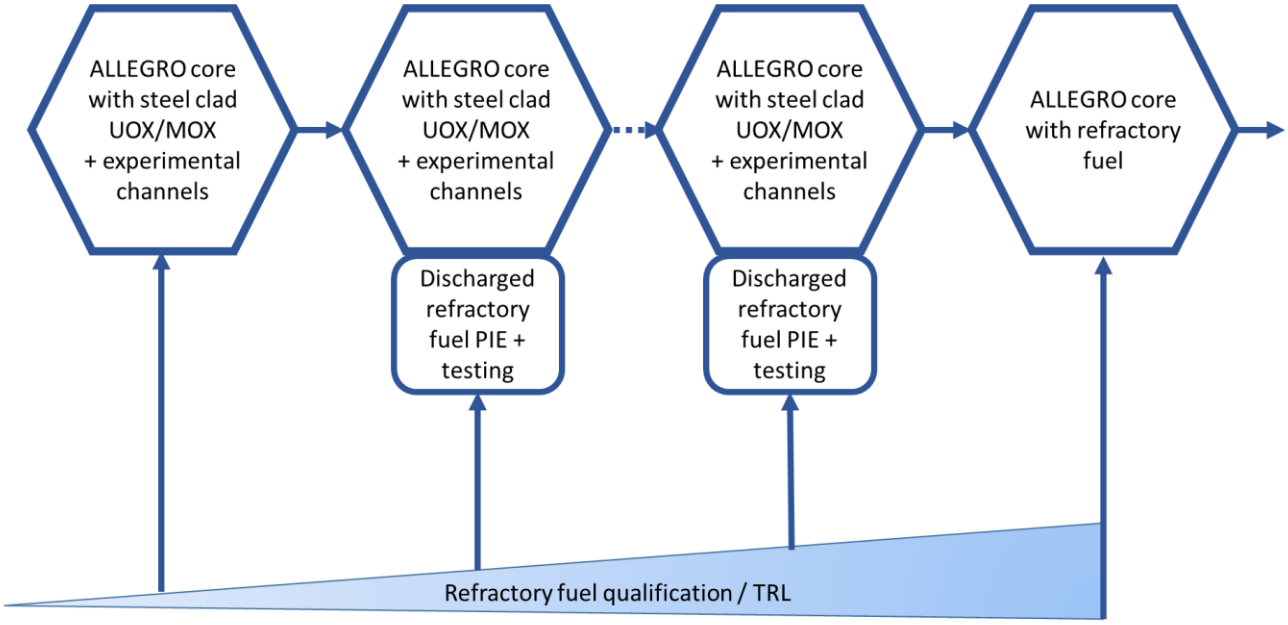

Figure 5: Qualification needs for ALLEGRO core with refractory start-up core

The following considerations should be taken into account during the design of ALLEGRO reactor start-up with refractory core:

- Stepwise start-up procedures have to be introduced with different power levels and core temperatures.
- The high power tests would be started in special core positions (e.g. with high local enrichment or special power control or coolant flow reduction).
- Possibility of wide range on-line instrumentation for reactor/fuel monitoring would be desirable
- The operational domain of the fuel will be limited (burnup, fluence, temperatures, power)
- Need for detailed post-test examination of fuel in hot cells facilities.

3 IRRADIATION EXPERIMENTS NEEDED FOR ALLEGRO REFRACTORY FUEL SYSTEM

The needs of the ALLEGRO refractory fuel system may be divided in three principal groups:

- Fuel behaviour with burnup (thermal properties, swelling, microstructural evolution, fission product behaviour…). This group of experiments is not needed for UO_2 (and to large extent for MOX) thanks to the maturity of the technology and availability of data covering the whole range needed for ALLEGRO. There is some information on carbide pellet behaviour in different reactors (see Appendix D, TRL 3), but additional measurements may be needed after the selection of pellet type and fabrication technology.
- SiC and SiC_f/SiC behaviour under irradiation. Here the data will be needed mainly for radiation growth, thermal properties of cladding, mechanical properties but also stability of joints – both cladding tube sealing and fuel assembly components joints. Despite the world-wide progress in the SiC development related to the LWR ATF programme, the number of the irradiation experiments and PIE needed for ALLEGRO / GFR is expected to be large due to specific GFR conditions and the variability of SiC – based materials. Fortunately, irradiation of the small non-fueled samples is less demanding and may be performed in numerous MTRs.
- Pin qualification experiments, both long-term steady state operation and operational transients. The main issue is the containment of the radioactive substances by the cladding – i.e. gross failure limits, leak-tightness of joints, release of the gaseous species through microcracks, diffusional release of Ag and other metals. The long term chemical interaction of the fuel with cladding will also have to be addressed, but here the experiments may be conducted on a smaller scale than a full sized pin. The secondary goal of the pin irradiation is the validation of the fuel performance codes.

While several material test reactors may host the experiments of the first two groups already, the pin qualification experiments will be problematic to perform under representative conditions because a He loop will be needed.

Full scale fuel assembly **irradiation** of the refractory fuel does not seem to be indispensable for the following reasons:

- Mechanical resistance, vibrational characteristics and thermal-hydraulic studies needs to be performed with unirradiated material only, **if it can be shown, that the irradiation effect on the material properties are well characterized and allow for the modelling of their impact.**
- The SiC, which is the material of choice, seems to saturate in terms of radiation damage relatively soons as opposed to steels.

However, if technical possibilities would allow (e.g. operation of He loop in a fast reactor) the in-pile testing of fuel assembly would be an important step in fuel qualification providing direct information on the potential interaction of assembly components.

Obviously, introduction of refractory fuel system without prior irradiation on an assembly level would require step-wise fuel qualification process with extensive PIE at various stages of irradiation and possibly more frequent core reloads than required by neutronics (most fuel assemblies would be kept in “safe” fluence levels with only few lead test assemblies allowed to go beyond).

4 IRRADIATION POSSIBILITIES

The irradiation by neutrons in a nuclear reactor can cause severe damage to the microstructure of the core materials and may result in the change of macroscopic dimensions, too. The effect of fast neutrons is much harsher compared to thermal neutrons. For material research the high fast flux is essential to test and qualify the materials.

The effect of neutron fluence on the material properties must be examined for each fuel component and the potential mechanisms leading to pin failure or large swelling (reduction of coolant channel) under and beyond normal operating conditions must be identified. Beside the irradiation of fuel components, the fuel rods and fuel assemblies have to be irradiated to check the potential structural changes and other effects.

Accelerator neutron sources or even ion accelerator techniques can be also used to simulate radiation damage in reactors, since high irradiation doses can be easily reached with ion beams. However, the effects caused by ions and neutrons are often different. Ion beams can be used in the screening of optimal material compositions for reactor components, but cannot be applied in the qualification process of reactor materials.

Any reactor used to qualify GFR will need to employ a dedicated test device designed to reach the desired test conditions (apart from HTR). The selection of the host reactor for the experiment depends on the experimental target. For example, for the fuel-cladding chemical interaction testing or fuel separate effect studies (densification, swelling), a high fast neutron flux is not strictly necessary. Such experiments might be performed in numerous reactors, provided that the temperature and power density conditions are made to be equivalent to GFR. It is a regular practice for Material Test Reactors (MTR) to use experimental devices emulating the conditions of the target reactor systems, but the development of new dedicated experimental device is a costly process. On the other hand, the experimental device, once developed may be adapted for different MTR provided its operating conditions are similar. For example, an in-pile creep rig, Melodie, was developed by CEA for Osiris reactor, now is being adapted for LVR-15 reactor and will be ultimately deployed in JHR.

For material studies, high dpa rate reactors will be needed.

A comprehensive study of experiments needed for ALLEGRO fuel element qualification based on future possibilities of JHR has been performed as an example and is listed Appendix C. As more information about other reactors is gathered, such study will be extended to them as well.

4.1 EXISTING REACTORS

Today there are no facilities available in Europe to provide the full qualification services for fast reactor fuel components and/or full assemblies [13].

- Outside of Europe the BOR-60 and BN-600 reactors in Russia, FBTR in India, can provide limited possibility to test nuclear fuel under fast neutron conditions. However, the access to these facilities is complicate.
- For high temperature testing in a gas-cooled thermal reactor the HTTR could be used in Japan but without fast neutron flux.

The irradiation possibilities for fuel testing was reviewed several times in the past [14][15][16][17] and the basic information on research reactors are available in the IAEA database [18]. The following short summary on existing reactors is based a review carried out within the EU ESNII Plus project [13].

HFR	The High Flux Reactor (HFR) is a 45 MWth versatile Materials Test Reactor (MTR) at Petten. It is owned by the European Commission, and operated by NRG. The HFR is a light water cooled and moderated aluminium tank-in-pool reactor pressurized to 3.4 bar. The HFR is a powerful multi-purpose research test reactor. Different fuel types are tested in the reactor and post-irradiation examination services are available. The HFR reactor will be replaced by the PALLAS reactor. Gas gap insulated capsules for material irradiation are routinely used HFR, LWR fuel irradiations are also performed in dedicated devices. No on-shelf device is available for GFR conditions.
BR2	The BR2 reactor is one of the major high flux MTR type reactors of the world, operated by SCK·CEN at the Mol site since 1961. It is a tank-in-pool type reactor. A very compact core provides high flux and power densities, while it leaves ample place for the connections of the experiments at the level of the reactor cover. The reactor is water cooled, and beryllium and water moderated. It is typically used at power levels of about 60 MW, but the cooling system is designed for operation up to 125 MW. A project of experimental irradiation, named IRRDEMO, was pre-design for testing plate type fuel for the GFR in the Belgium experimental reactor BR2. Currently, no on-shelf device is available in BR2 for GFR conditions, but there is significant experience with design and operation of various loops and capsules, including sodium loop in the past. Currently, this is the only European reactor available for larger scale fuel irradiations.
LVR-15	The LVR-15 reactor, located at the UJV Řež site and operated by Research Centre Řež Ltd. (CV Řež), is a tank type water cooled reactor moderated with beryllium reflector. The reactor was commissioned in 1957; since then it has undergone reconstruction twice, where the last reconstruction took place in 1989, when all reactor components and systems were replaced, including the vessel. The LVR-15 core is typically operated with 28–34 fuel elements with water, mixed or beryllium reflector. No fuel (fissile material) possible under current licence limitations. Limited dpa rate (up to 1 – 2 dpa/year in steel). Suitable for development and testing of irradiation devices for other reactors.
TRIGA-SSR	The Institute for Nuclear Research (INR) in Pitesti (Romania) operates since 1980 two research reactors: the 14 MWth TRIGA steady-state reactor and the TRIGA ACPR (pulsed) reactor. Both cores are placed in the same pool. The conversion of the steady-state core to low enrichment fuel has been finished in 2006, followed by a general refurbishment and upgrading of the facility that was completed in 2011. No detailed information on relevance to GFR available.
BRR	The Budapest Research Reactor (BRR) is located in Budapest, operated by the Centre for Energy Research (EK). The reactor was commissioned in 1959, since then it has undergone reconstruction twice, where the last, full-scale reactor

reconstruction was completed in 1992. The BRR is a light-water cooled and moderated tank-type reactor with beryllium reflector.

MARIA The National Center for Nuclear Research (NCBJ) in Poland is the operator of the multifunctional nuclear research reactor MARIA. The high flux research reactor MARIA is a pool type, water and beryllium moderated reactor, with graphite reflector and pressurized channels containing concentric tube assemblies of fuel elements.

This reactor is not apparently open for international research as an MTR, most probably not an option.

ATR The Advanced Test Reactor (ATR) is a research reactor at the Idaho National Laboratory, located near Idaho Falls. This reactor is primarily designed and used to test materials to be used in larger-scale and prototype reactors. It can operate at a maximum power of 250 MW and has a "Four Leaf Clover" that allows for a variety of testing locations. The reactor has 34 in-core irradiation channels and the unique design allows for different flux in various locations and specialized systems also allow for certain experiments to be run at their own temperature and pressure.

In recent years many ATF – focused irradiations were performed in ATR. Extremely flexible, but very focused on US needs.

HFIR The High Flux Isotope Reactor (HFIR) is a nuclear research reactor located at Oak Ridge National Laboratory (ORNL) in Oak Ridge, Tennessee, United States. Operating at 85 MW, HFIR is one of the highest flux reactor-based sources of neutrons for condensed matter research in the United States, and it provides one of the highest steady-state neutron fluxes of any research reactor in the world. The thermal and cold neutrons produced by HFIR are used to study physics, chemistry, materials science, engineering, and biology.

Extremely successful in high dpa irradiations of miniature sample of both construction material (ATF claddings inc. SiC) and fuel (inc. TRISO fuel) recently.

SM-3 SM-3 is a high-flux water-cooled vessel-type research reactor with a neutron trap that operates in the intermediate neutron spectrum. The core with dimensions of 420×420×350 mm has a central neutron trap and beryllium metal reflector 500 mm in height arranged in a steel vessel 1.46 m in diameter and 7.33 m in height. The core comprises 28 fuel assemblies. It is located in Dimitrovgad, Russia.

No detailed information on relevance to GFR available.

MIR The Russian loop-type research reactor MIR is designed mainly for testing fuel elements, fuel assemblies and other core components of different types of operating and promising nuclear power reactors. Tests and experiments simulate both standard (steady-state and transient) conditions and the majority of the design-basis accidents. Tests can be carried out in several (up to 10) channels at a time, the neutron flux density being 4-5 times different from the average one. The maximum in-core neutron flux is $5.0 \cdot 10^{14}$ n/cm^2·s. 8 loop facilities will be available in the MIR reactor (under construction).

Very much dedicated to support of Russian LWR fuel programme.

HANARO	The HANARO reactor is 30 MW_{th} research reactor operated by the Korean Atomic Energy Research Institute (KAERI). The reactor can be used for fuel and material tests, to support the manufacturing of fuel and reactor components for pressurized light water reactors (PWR) and CANDU power plants, as well as the production of radioisotopes, neutron activation analysis of nuclear materials, neutron radiography for the examination of spent fuel assemblies, and non-destructive examinations of both nuclear and non-nuclear materials.
JMTR	JMTR is a reactor for testing materials, operated by Japan Atomic Energy Agency (JAEA) in Oarai, that has been developed for the irradiation test of reactor fuels and materials in order to define their characteristics. I has been used mostly for irradiation test aimed at integrity evaluation of LWR materials and fuels, for the development of fusion reactor materials and also in order to elucidate the mechanisms of irradiation damage of materials.
HFETR	The 125 MW light water High-Flux Engineering Test Reactor (HFETR) is a light water moderated, light water cooled materials test reactor primarily intended for testing PWR fuel assemblies. It has been re-equipped with Low Enriched Uranium (LEU) driver fuel assemblies and has 7 in-core channels and 4 reflector irradiation channels. The reactor is located in Chengdu, China.
CARR	China Advanced Research Reactor (CARR) is an inverse neutron trap, tank-in-pool research reactor. Its fuel element is plate type, coolant and moderator is light water, and reflector is heavy water. It is a multi-purpose research reactor for fuel and structural materials irradiation testing that is also use for neutron scattering, radiotherapy and boron capture neutron therapy. CARR has 22 vertical channels (max flux $8{\cdot}10^{14}$ $n/cm^2{\cdot}s$), 9 horizontal channels, 1 loop (max flux 4E14 $n{\cdot}cm^{-2}s^{-1}$) and 4 in-core channels (max flux $1{\cdot}10^{15}$ $n/cm^2{\cdot}s$).
HTTR	The High Temperature engineering Test Reactor (HTTR) constructed is the first high-temperature gas-cooled reactor (HTGR) in Japan. The HTTR is a graphite-moderated and helium-gas-cooled reactor. The main objectives of the HTTR are to establish and develop HTGR technology and to demonstrate process heat application.
FBTR	Fast Breeder Test Reactor (FBTR) is primarily used for fast reactor fuels and structural materials testing, though it also has a steam circuit to generate power. The reactor design is based on the Rapsodie reactor that was built and operated in France, though it uses a novel driver fuel with Pu-U monocarbide pellets. There is a central driver core with radial and axial breeding blankets. There is one in-core irradiation channel and there are four vertical channels. The reactor is located in Kalpakkam, India.
BOR-60	The fast 60 MW_{th} sodium-cooled reactor BOR-60 is a low-power NPP prototype located in Dimitrovgrad. It is used to test fuel cycle, sodium coolant technologies and a wide range of design concepts for fast reactors. Being a powerful source of fast neutrons, this reactor is used to study the effect of neutron irradiation on various structural, fuel and absorbing materials. The maximal fast neutron flux density in the reactor is $3.7{\cdot}10^{15}$ $n/cm^2{\cdot}s$.

This reactor is open to commercial international R&D programs, however results form Russia might be somewhat difficult to accept by the EU regulators.

Nonetheless, the irradiation of SiC or SiC_f/SiC tubes in BOR-60 would be extremely useful for refractory fuel qualification. The samples irradiated in fast spectrum at high temperatures could be compared with the samples irradiated in thermal spectrum (BR-2, LVR-15…) to confirm the saturation of SiC radiation damage at low dpas. Unfuelled samples would be simple to transport to other hot labs for comprehensive PIE in line with ALLGERO/GFR needs.

BN-600 The Russian BN-600 fast breeder reactor – Beloyarsk unit 3 of 600 MW_e – has been supplying electricity to the grid since 1980 and is said to have the best operating and production record of all Russia's nuclear power units. It uses chiefly uranium oxide fuel, enriched to 17, 21 and 26%, with some MOX in recent years. It is a pool-type reactor, with heat exchanger for secondary coolant inside a pool of sodium around the reactor vessel and 3 steam generators outside the pool. The sodium coolant delivers 525-550 °C at little more than atmospheric pressure. Russia plans to reconfigure the BN-600 by replacing the fertile blanket around the core with steel reflector assemblies to burn the plutonium from its military stockpiles.

This reactor is closest to the global GFR conditions, but to our knowledge has not been used for the international R&D purposes.

SFR fuel development leveraged on irradiation of new fuel element designs or fuel assembly designs in existing SFR. Similar irradiation of GFR refractory fuel in SFR would provide data on the fuel behaviour in limited temperature range, but at large scale. Such process may be also feasible – to qualify refractory fuel up to SFR temperatures, use it to start ALLEGRO at lower parameters (equivalent to now envisaged steel clad cores, but with higher safety margins) and gradually extend the qualification up to higher temperatures in experimental positions with reduced coolant flow rate.

In the future the MBIR and the MYRRHA reactors could be used for in-pile testing of fuel related components in conditions close to the gas-cooled fast reactor conditions. The JHR reactor in France will be available for some irradiation and transient tests. The PALLAS reactor in the Netherlands will have also some capabilities for fuel testing. These future reactors will be shortly described in the following chapters.

4.2 MYRRHA

The Belgian Nuclear Research Centre (SCK·CEN) in Mol is working since several years on the design of a multi-purpose flexible irradiation facility to succeed the BR2 reactor, operated since 1962 as a multi-purpose materials testing reactor (MTR).

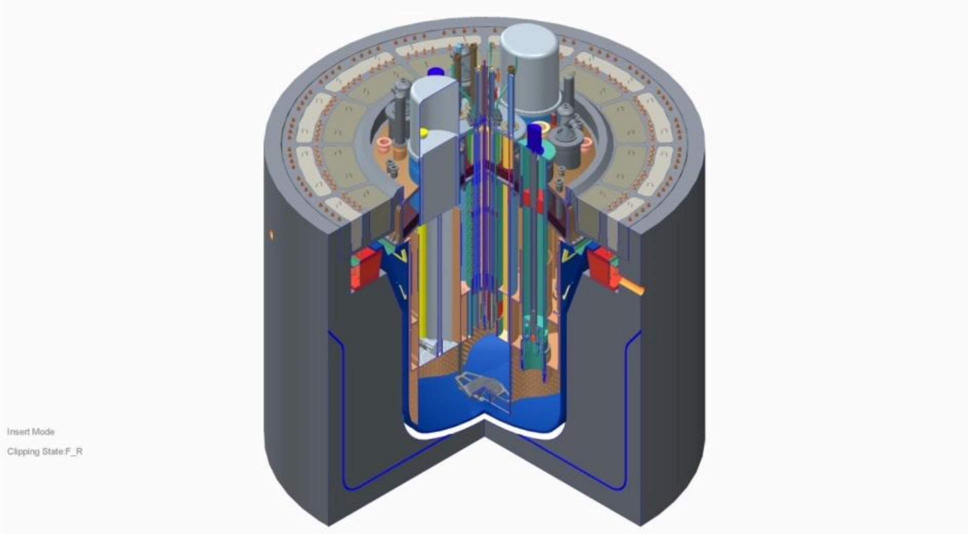

Figure 6: The most recent version of the MYRRHA reactor

MYRRHA is conceived as an accelerator driven system (ADS), able to operate in sub-critical and critical modes. It consists of a proton accelerator of 600 MeV, a spallation neutrons source and a nuclear core with MOX fuel, cooled by liquid lead-bismuth (Pb-Bi). MYRRHA will be a flexible fast spectrum irradiation facility for material developments for innovative fission and fusion reactors.

A phased implementation strategy spreading investment costs and mitigating the technical, cost and planning overrun risks was decided:

- A 100 MeV accelerator sub-programme has been launched in early 2016 as a first phase of the full programme allowing to have on site a research facility operational in 2024 allowing physics R&D through an ISOL-target and producing radioisotopes.
- The second phase will extend the accelerator to 600 MeV beam energy based on the design and prototyping already performed until 2024.
- The third phase is the construction of the reactor. The realisation of phases 2 and 3 can be conducted in parallel depending on consortium build-up and financial constraints at that time.

Due to the timeframe of MYRRHA construction it is extremely unlikely that this reactor would be able to participate in the ALLEGRO / GFR fuel qualification.

4.3 JHR

The Jules Horowitz Reactor (JHR) is a new MTR currently under construction at CEA Cadarache research centre in the south of France. It will represent a major research infrastructure for scientific studies dealing with material and fuel behaviour under irradiation (and is consequently identified for this purpose within various European road maps and forums; ESFRI, SNETP…). The reactor will also contribute to medical isotope production.

CVR / UJV are members of the JHR consortium with 2% share. Also, Euratom has another 6% share. The JHR consortium acknowledges the need to address not only the needs of the current EU reactor fleet but also of the perspective reactor systems including GFR.

It a compact 100MWt low pressure water cooled research reactor focusing on radioisotope production and LWR material testing. While hard neutron spectrum is obtainable in the core, the temperature rise from the reactor (50°C) or pool temperature to GFR conditions is challenging, but not impossible.

At nominal operation JHR is to operate with 10 cycles a year, representing about 260 Equivalent Full Power Days (EFPD).

JHR is designed to provide high neutron flux, to run highly instrumented experiments, to support advanced modelling giving prediction beyond experimental points, and to operate experimental devices giving environment conditions (pressure, temperature, flux, coolant chemistry, etc.) relevant for water power reactors (PWRs, BWRs, VVERs), but also in support of non-water reactors R&D (sodium cooled fast reactors).

JHR design accommodates improved on-line monitoring capabilities such as a fission product laboratory directly coupled to the experimental fuel sample under irradiation.

The operation of the new JHR facility is planned for the beginning after 2030

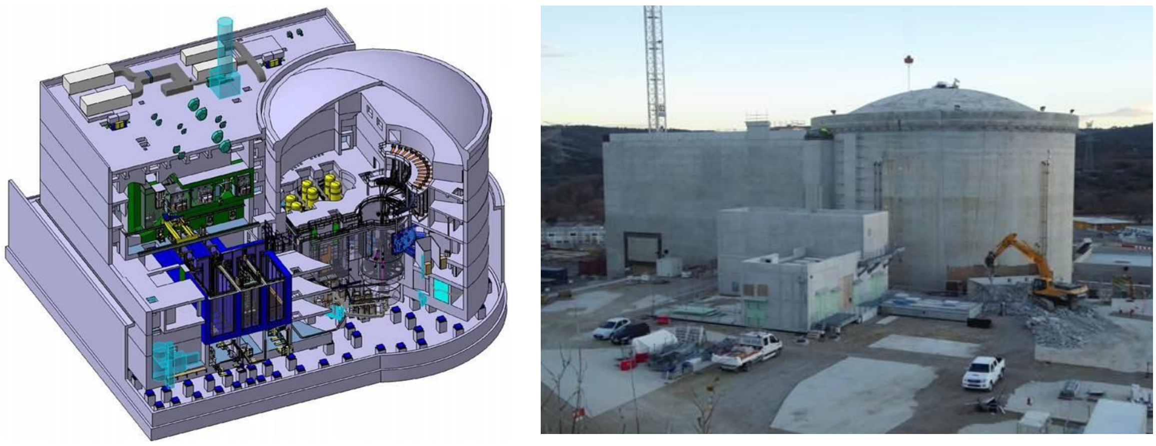

Figure 7: General overview of JHR and the building site as of end of 2015

The experimental devices available at the reactor startup are principally:

- MADISON – a PWR loop device in the JHR reflector for long term steady state irradiation. No release of activity into coolant during the experiments. The device may host up to 4 LWR rods with 2 sensors each.
- ADELINE – an PWR loop device in the JHR reflector for testing the fuel pins in the conditions where fuel rod failure is expected, typically ramp tests. A rod with up to 12.5 OD mm could be accommodated into a highly instrumented experimental position.
- MICA / OCCITANE – devices for non-fissile materials irradiated at controlled temperature within JHR core.

A boiling water capsule FUCA for the screening irradiation of fuel (fissile material) samples for scoping studies is under preparation. This irradiation will probably be without instrumentation or with limited number of sensors.

As a part of the preparation of the JHR experimental programme, the ranking of the interest in the possible experiments has been performed within the JHR consortium as a basis for the planning of the future expansion of the fleet of experimental devices. This list of experiments has been commented with respect to GFR needs in Appendix C (LWR specific experiments such as cladding corrosion and LOCA testing were omitted). The list is not exhaustive, but shows which experiments are under consideration in the JHR consortium.

The results of this exercise may be summarized:

- Stress free irradiation of cladding and FA structural material samples would be possible soon after the start JHR with irradiation temperatures up to 450°C with an epithermal neutron spectrum up to 12 dpa/year or with a fast spectrum obtained by application of shielding. In future, creep testing and temperatures up to 650°C and possibly 1000°C should be possible. In these devices, electrically heated NaK combined with gamma heating in the sample and gas gap insulation is used to achieve the target temperature.
- Fuel experimental capabilities of JHR are focused on LWR conditions. Of separate effect studies, limited non-oxide fuel irradiations in double wall capsules in MADISON or FUCA seems feasible without major effort, but none of the experimental devices had been designed with such fuels in mind so there may be licensing issues. Also, the size (and hence the total heat output) of the of the fuel samples would need to be carefully considered to obtain the representative temperatures while maintaining the safety.
- Integral pin studies would be limited at current fleet JHR devices, possible only with not fully representative spectrum (impacting cladding behaviour) and at basically LWR temperatures (impacting both cladding and fuel behaviour). Such experiments are not enough to fully qualify the fuel pin design for ALLEGRO. Design of dedicated GFR experimental device would be needed. This is not impossible task, but requires significant resources to be invested.

4.4 MBIR

At the Research Institute of Atomic Reactors (RIAR or NIIAR) in Dimitrovgrad the BOR-60 reactor will be replaced by a 150 MW_{th} multi-purpose fast neutron research reactor (MBIR), designed for a broad range of in-pile research activities and experiments [19][20][21][22]. The construction started in September 2015. The MBIR concept incorporates the ability to reach a high neutron flux ($\leq 5\cdot 10^{15}$ cm^{-2} s^{-1}) and to insert into the reactor up to three secondary-loop facilities using different coolant.

In the field of reactor materials science, promising types of fuel and structural and absorbing materials will be tested in MBIR and new and modified coolants and the means for monitoring them and controlling their quality will be studied. In nuclear and radiation safety, there are plans to validate new means of passive action and investigate fuel elements and fuel assemblies in transient, cyclic, and emergency regimes. Significant part of the research programme will be devoted to service-life tests and working out new technical solutions for fuel elements, fuel assemblies, absorbing elements, and other elements of the core as well as tests of new types of equipment, means of monitoring and performing diagnostics of the core, reactor, and coolant loops.

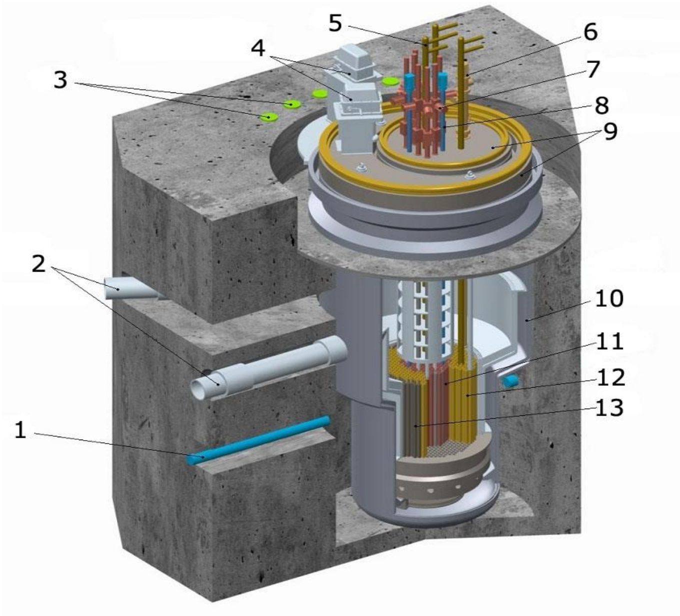

Figure 8: MBIR model [21]
(1 – horizontal experimental channel; 2 – primary pipelines; 3 – vertical experimental channels; 4 – rotary plug drives; 5 – large loop channel; 6 – reloading mechanism; 7 – CPS rod drives; 8 – experimental channel; 9 – rotary plugs; 10 – vessel with a safeguard shroud; 11 – fuel assembly; 12 – side reflector; 13 – in-reactor storage)

The in-vessel experimental facilities of the reactor consist of three cells for circuit channels making it possible to connect to the first loop facilities with different types of coolant, three cells for instrumented materials-science assemblies intended for testing fuel, structural, and absorbing materials, 14 cells for non-instrumented materials science assemblies and irradiation setups for isotope production.

Autonomous facilities of the type channel–loop with different coolant can be installed in cells with loop channels as well as instrumented materials-science assemblies. These facilities make it possible to maintain prescribed values of the thermodynamic parameters by means of natural or forced circulation of the coolant, organized within a channel and have outputs for measurement cables and service lines from the reactor.

Instrumented in-pile experimental devices allow:

- testing of structural and fuel materials in the set environment with measurement and regulation of irradiation temperature (320-1800 °C);
- in-pile investigation of material mechanical characteristics.

The behaviour of fuel in simulations of transient and nonstandard situations with different types of coolants – gas, light sodium, heavy metal lead, Pb+Bi, molten salt – will be studied in the MBIR. The sodium temperature at the reactor entry is set at 330 °C in order to test materials for pressurized water reactors.

The yearly damaging dose in side-screen cells will be 11–15 dpa, up to 33 dpa in cells in the central part of the core and reaching 40 dpa in the cells with installed capacity utilization factor.

In 2017 a memorandum of understanding between State Atomic Energy Corporation ROSATOM and V4G4 Centre of Excellence was signed with the intention to test GFR fuel in MBIR.

The MBIR reactor is supposed to be commisioned in 2028 [23], however the progress seemed to slow down in last years.

4.5 PALLAS

PALLAS is a new research reactor which will replace HFR in Petten. HFR. This will be a state-of-the-art reactor equipped to meet the growing world demand for both nuclear knowledge and services and the production of essential medical isotopes. PALLAS will make accelerated developments of new fuel designs and improvements of existing concepts.

The major requirements for PALLAS, derived from the expected utilization, are:

- peak fast neutron flux at least one and a half time the value of the HFR,
- peak thermal neutron flux two to three times the HFR value,
- compact, flexible core with a replaceable beryllium reflector concept to economize on the use of fissile and reflector material.

The PALLAS reactor will be a tank-in-pool type for simple reliable handling of experiments and isotope production. The reactor power interval will be flexible within the boundaries of 30 MW to 80 MW maximum to optimize fuel utilization in line with demands for irradiation services.

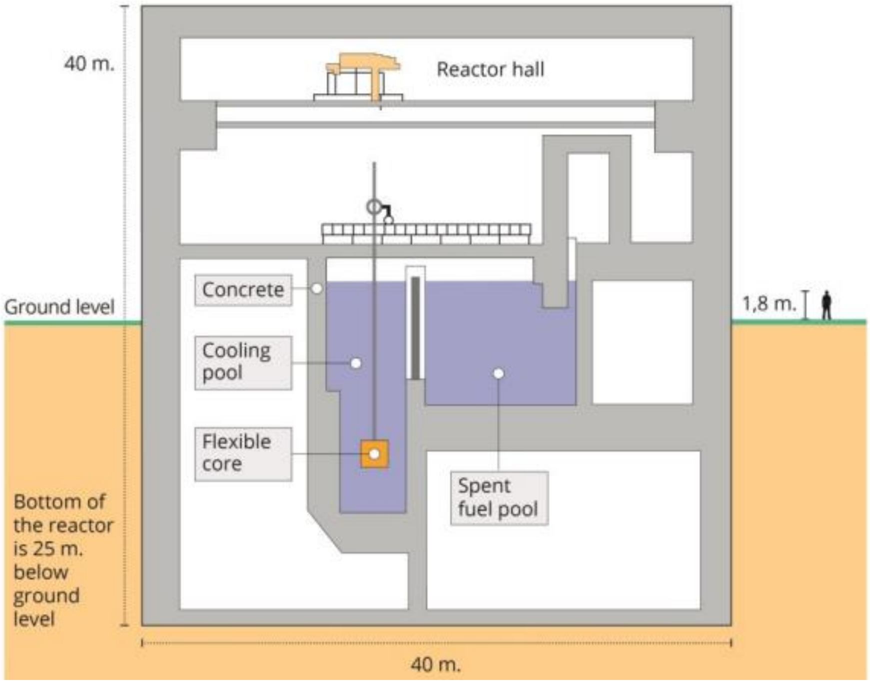

Figure 9: Scheme of the PALLAS reactor [24]

The research reactor might provide rigs and loops for the science and engineering of materials in a neutron radiation environment inc. the Gen IV, however its focus is on the radioisotope production.

It is expected that PALLAS will not be fully operational before 2026 (with a full transition from HFR not sooner than 2030).

4.6 VTR

The Versatile Test Reactor, or VTR is new research reactor, that will be capable of performing irradiation testing at much higher neutron energy fluxes than what is currently available today [25].

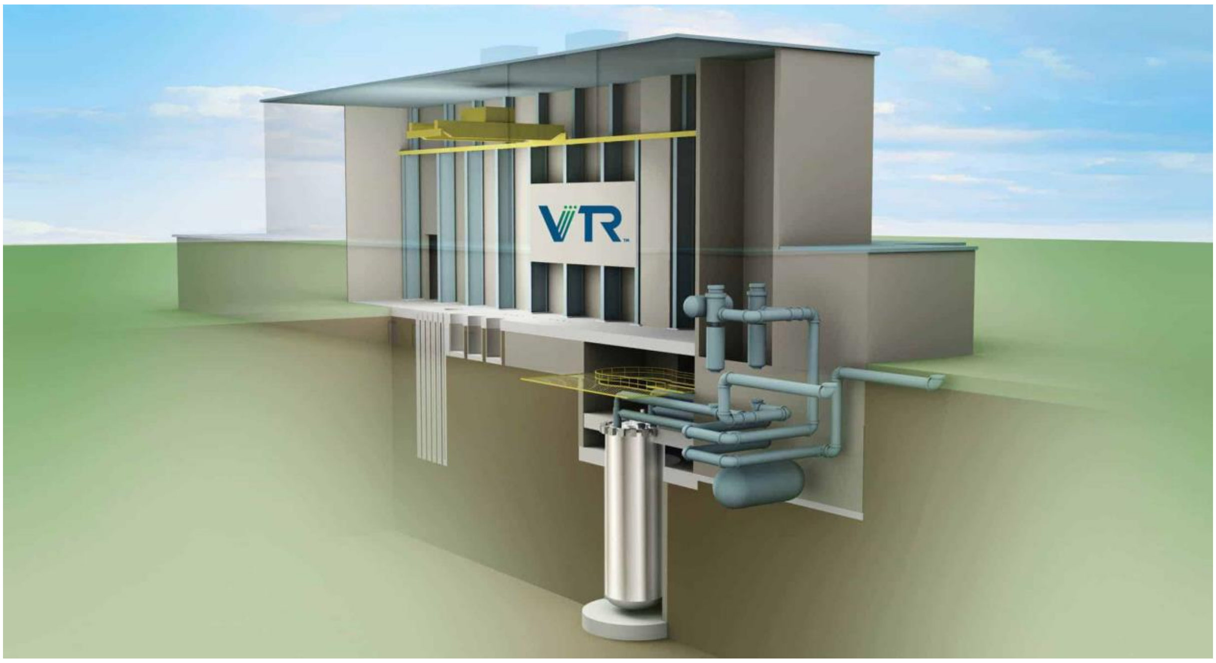

Figure 10: Visual design of the VTR reactor

This capability will help accelerate the testing of advanced nuclear fuels, materials, instrumentation, and sensors. It will allow to modernize its essential nuclear energy research and development infrastructure and conduct crucial advanced technology and materials testing.

The VTR will be based on GE Hitachi Nuclear Energy's (GEH) PRISM pool-type sodium-cooled small modular reactor design with metallic (uranium-plutonium-zirconium alloy) fuel [26].

VTR, which can be adapted for several types of experiments, is designed to support university researchers as well as industrial designers and developers.

The versatility of VTR's design can produce results for gas-cooled, lead and lead-bismuth, sodium and molten salt reactors. These technologies use different fuels and coolants than today's light-water reactors.

There will be four test-vehicle types, or methods, of inserting experiments into VTR: normal, extended-length, rabbit, and dismountable test assembly.

VTR could be completed as early as 2026 at the site of one of DOE's national laboratories.

5 GFR FUEL QUALIFICATION PROCEDURE

The qualification process for nuclear fuel is not a standard procedure today. It depends on the reactor and fuel type, and on the actual requirements of the nuclear authority in a given country [2][4][27][28].

At the beginning of the development of GFR fuel qualification procedure useful consultations [3][4] were held with SFR experts, but this type of document was not compiled even for the operated SFR reactors.

In this chapter the available fuel qualification methods, approaches will be reviewed. The steps of GFR refractory fuel qualification will be proposed making use of the combination of existing methodologies. Finally, the points of decision making in the qualification process will be identified.

5.1 TRL APPROACH

A Technology Readiness Assessment (TRA) evaluates technology maturity using the TRL scale and was pioneered by NASA in the 1980s for space technology. In 2007 the Department of Energy (DoE) adopted the TRLs and applied the methodology to nuclear fuels and material systems.

The TRL concept is used as a program management and communications tool and is not meant as an absolute quantitative measure of maturity. There is naturally a level of subjectivity in defining and in evaluating the TRLs. Carmack et al. [29] provided proposed attributes and categorization for nuclear fuel system technology readiness level definition. The used TRL scale ranged from 1 (basic principles observed) through 9 (total system used successfully in project operations).

- TRL levels 1-3 correspond to **proof-of-concept phase**. A new fuel concept is proposed (TRL 1). The technical options have been identified and preliminary evaluation is underway (TRL 2). Concepts are verified through laboratory scale experiments and characterization (TRL 3).
- The **proof-of-principle phase** (TRL levels 4-6) requires establishing fabrication capability for representative material at least at the laboratory scale and progressing to in-pile irradiation testing. At TRL 4 fabrication of samples using stockpile materials at bench-scale yielding small fuel elements, rodlets, and small scale pin configurations. TRL 5 includes the fabrication of full scale fuel elements using laboratory scale fabrication capabilities with subsequent pin-scale irradiation testing conducted in relevant prototypic steady-state irradiation environments. At TRL 6 fabrication of engineering-scale test pins using prototypic feedstock materials is conducted. Fuel pin irradiation testing and performance verification is conducted in prototypic irradiation environments.
- In the **proof-of-performance phase** (TRL levels 7-9) the scale of fabrication reaches engineering and commercial scales. TRL 7 represents the established capability to fabricate test assemblies using prototypic feedstock materials at engineering-scale and using prototypic fabrication processes. TRL 8 designates that a few core loads of fuel have been fabricated and full core operation of a prototype reactor with such fuel has been accomplished. TRL 9 designates that the fuel technology is routinely conducted at commercial-scale and normal operations are underway.

D. Sheperd (National Nuclear Laboratory - NNL) pointed out [30] that the original NASA TRLs were defined for systems for individual space missions and the terminology is not always suitable for nuclear industry applications. It was proposed to introduce an additional level: TRL 10 to cover the experience from operating many actual systems (long term use of fuel in nuclear power plants). Similar proposal was developed by Straub [31] for aerospace applications, too (proven operation).

TRL	Definition & Description
10	**Long term operation of many actual systems** *e.g. long term use within commercial reactor fleets i.e. many thousands of hours operating experience*
9	**Successful operation of actual system** *e.g. assemblies successful after irradiation in reload quantities (at least one third of a full reactor core)*
8	**Actual system constructed and commissioned** *e.g. assemblies fabricated in reload quantities, may include irradiation with only limited success*
7	**Prototype successfully demonstrated** *e.g. lead use assemblies successful in a prototype or commercial reactor*
6	**Prototype construction (much more representative than basic system)** *e.g. lead use assemblies irradiated in a prototype or commercial reactor but with only limited success*
5	**Basic system successfully demonstrated** *e.g. test rods have been irradiated and performed successfully in a test reactor*
4	**Integration of components into a basic system** *e.g. test rods subjected to out-of-reactor tests and/or irradiated in test reactor but with only limited success*
3	**Basic components successfully demonstrated** *e.g. fuel and/or cladding components have been tested out-of-reactor and/or irradiated as a component only*
2	**Practical applications suggested** *e.g. fuel and cladding designs have been established*
1	**Research into the principles underlying the technology** *e.g. promising materials have been identified*

Table 1: Summary of TRL definitions in the UK [32].

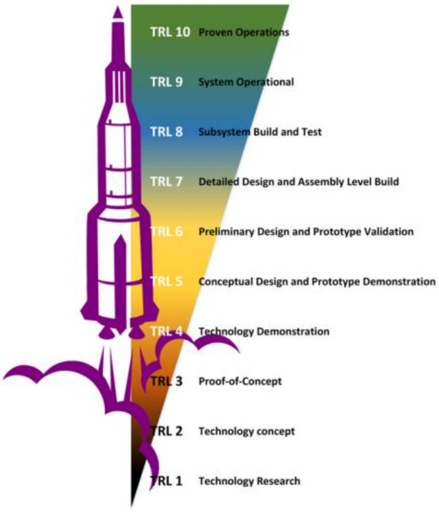

Figure 11: TRL levels defined by NASA [31]

OECD started to applied this scale on innovative fuels for GENIV reactors and also for ATF [33].

The NNL conducted a TRL assessment on the advanced fuels for deployment in Gen III/III+ and IV systems in 2015 [30][32]. These down-selections were made by applying pre-existing knowledge of the relevant systems incorporating the results from a literature review, conference attendance, relevant facility visits and discussion with partners in the international nuclear community.

The NNL assessment included also those GFR fuel types, which are considered for the refractory core of ALLEGRO:

- The SiC_f/SiC cladding is characterised by TRL 3 for all reactor types.
- The carbide fuel had also TRL 3 for GFR conditions.

This assessment showed that the application of SiC_f/SiC cladding and carbide pellets in a GFR reactor needed large efforts on development and qualification in 2015. As will be shown in chapter 5.3, this more or less holds until today.

5.2 The OECD CNRA report

The CNRA Working Group on the Safety of Advanced Reactors produced a valuable technical report [1] on Regulatory Perspectives on Nuclear Fuel Qualification for Advanced Reactors, which describes the regulatory perspectives on nuclear fuel qualification for advanced reactors and identifies topics that should be investigated in the frame of advanced reactor fuel regulation.

The topic of fuel qualification was recognized as a challenging topic from a regulatory point of view due to a lack of clarity regarding the definition and scope of fuel qualification, and uncertainty associated with the regulatory basis for fuel qualification. The regulatory process for fuel qualification is generally implicit in the overall licensing of a nuclear facility. The following common positions were identified by the authors of the report:

1) Fuel qualification requirements are derived from higher level nuclear power plant requirements (e.g., protection of "first barrier", accident source term, assurance of coolable geometry). However, guidance specific to the fuel qualification process is not generally available.

2) Process for qualifying fuel is generally implicit in the overall licensing of the nuclear facility.
3) An essential part of fuel qualification is to define a test envelope to cover expected operating, transient, and accident conditions to assess fuel performance and validate fuel performance codes.
4) An irradiation testing program, which includes testing of the integral fuel design, is necessary to identify fuel failure and degradation mechanisms. This testing should provide irradiation covering the exposure or burnup limits of the fuel.
5) Fuel qualification requires transient testing to assess fuel performance under transient and accident conditions.

A systematic evaluation of the requirements for qualifying nuclear fuel has been performed and a list of criteria has been identified to support a determination that nuclear fuel is qualified for use. The tables below provide a concise list of all the criteria for the following categories:

- manufacturing and safety limits (Table 2),
- evaluation models (Table 3),
- experimental data (Table 4).

The criteria highlighted in gray are identified as objective criteria for which direct evidence is needed to determine that the criteria are met. Higher level criteria in white are satisfied by satisfying all the lower level supporting criteria.

GOAL	**Fuel element is qualified for use**			
G1	Fuel is manufactured in accordance with a specification			
	G1.1	Key dimensions and tolerance of fuel components are specified		
	G1.2	Key constituents are specified with allowance for impurities		
	G1.3	Microstructure attributes for materials within fuel component are specified for otherwise justified		
G2	Margin to safety limits can be demonstrated with high confidence			
	G2.1	Margin to design criteria under conditions of normal operation, including the effects of AOOs		
		G2.1.1	Fuel performance envelope is defined	
		G2.1.2	Evaluation model (go to EM Assessment Framework)	
	G2.2	Margin to radionuclide release limits for accident conditions		
		G2.1.1	Fuel performance envelope is defined	
		G2.2.1	Radionuclide retention requirements are specified	
		G2.2.2	Criteria for barrier degradation and failure	
			(a)	Conservative criteria
			(b)	Experimental data is appropriate (go to ED Assessment Framework)
		G2.2.3	Radionuclide retention and release from fuel matrix	
			(a)	Conservative model
			(b)	Experimental data is appropriate (go to ED Assessment Framework)

	G2.3	Ability to achieve a safe state can be ensured		
		G2.3.1	Criteria specified for ensuring coolable geometry	
			(a)	Criteria to ensure coolable geometry are specified
			(b)	Criteria are shown to provide conservative prediction of coolable geometry loss
			(c)	Criteria are supported by experimental data (go to ED Assessment Framework)
		G2.3.2	Control element insertion can be demonstrated with high confidence	
			(a)	Criteria provided to ensure control element insertion path is not obstructed
			(b)	Evaluation model (go to EM Assessment Framework)

Table 2: List of Goals in Fuel Qualification Assessment Framework [1]

GOAL	**Evaluation model is acceptable for use**		
EM G1	Evaluation model contains the appropriate modelling capabilities		
	EM G1.1	Geometry	
	EM G1.2	Materials	
	EM G1.3	Physics	
EM G2	Evaluation model has been adequately assessment against experimental data		
	EM G2.1	The data used for assessment is appropriate (go to ED Assessment Framework)	
	EM G2.2	The evaluation model has demonstrated the ability to predict fuel failure and degradation mechanism over the test envelope	
		EM G2.2.1	Evaluation model error is quantified through assessment against experimental data
		EM G2.2.2	Evaluation model error is determined through the fuel performance envelope
		EM G2.2.3	Sparse data regions are justified
		EM G2.2.4	Evaluation model is restricted to use within its test envelope

Table 3: List of Goals in Evaluation Model Assessment Framework [1]

GOAL	**Experimental data used for assessment is appropriate**
ED G1	Assessment data is independent of data used to develop/train the evaluation model
ED G2	Data has been collected over a test envelope that covers the fuel performance envelope
ED G3	Experimental data have been accurately measured

	ED G3.1	The test facility has an appropriate quality assurance program
	ED G3.2	Experimental data is collected using established measurement techniques
	ED G3.3	Experimental data accounts for sources of experimental uncertainty
ED G4	Test specimens are representative of prototypical fuel	
	ED G4.1	Test specimens are fabricated consistent with the prototypical fuel manufacturing specification
	ED G4.2	Distortions are justified and accounted for in the experimental data

Table 4: List of Goals in Experimental Data Assessment Framework [1]

5.3 STEPS OF REFRACTORY GFR FUEL QUALIFICATION

In this section the TRL methodology will be applied with the combination of CNRA categories in order to identify the qualification steps for GFR refractory fuel. The present list is much more detailed than it was proposed in the first qualification report [34].

- The **pre-existing knowledge** for each TRL level was collected from available information in different open sources.
- The **further actions** were identified for each TRL level considering the ongoing GFR fuel related activities and literature sources in order to determine the missing information in the qualification process.
- The four CNRA categories (**manufacturing** / **safety limits** / **models** / **experiments**) were considered in the identification of pre-existing knowledge and further actions. These four categories were assigned to TRL levels as shown in Table 5. The categories are used in an extended way:

  MANUFACTURING – both laboratory and industrial scale, technologies.
  SAFETY LIMITS – safety limits, general requirements, design values.
  MODELS – code capabilities, validation, support of experiments, reactor applications.
  EXPERIMENTS – in-pile and out-of pile testing, on-line measurements and PIE.

The details on pre-existing knowledge and further actions are listed in Appendix D, summary is provided in Table 6.

TRL	Objectives and definition by Carmack [29] and Sheperd [30]	MANUFACTURING	SAFETY LIMITS	MODELS	EXPERIMENTS
1	Promising materials were identified • Research identifies the basic principles that underlie the technology e.g. promising materials and/or geometry have been identified. A new concept is proposed. Technical options for the concept are identified and relevant literature data reviewed. Criteria developed.		+		

TRL	Objectives and definition by Carmack [29] and Sheperd [30]	MANUFACTURING	SAFETY LIMITS	MODELS	EXPERIMENTS
2	Fuel and cladding designs were selected • Practical applications suggested and concepts formulated e.g. fuel, cladding and/or fuel assembly designs have been established. Technical options are ranked. Performance range and fabrication process parametric ranges defined based on analyses.	+		+	
3	Fuel and cladding was successfully tested in reactor • Basic components fabricated and successfully demonstrated e.g. fuel and/or cladding components have been manufactured and tested out-of-reactor and/or irradiated as a component only. Concepts are verified through laboratory-scale experiments and characterization. Fabrication process verified using surrogates.	+		+	+
4	Fuel rod was fabricated and tested • Integration of components into a basic system e.g. representative assembly sections have been manufactured and subjected to out-of-reactor tests and/or test reactor irradiation trials of individual rods have been conducted with only limited success. Fabrication of samples using stockpile materials at bench-scale irradiation testing of small-samples (rodlets) in relevant environment. Design parameters and features established. Basic properties compiled.	+	+	+	+
5	Fuel rod was successfully tested in reactor • Basic system successfully demonstrated e.g. test rods have been irradiated and performed successfully in a test reactor (demonstrated by in-reactor instrumentation and/or post irradiation examination (PIE) and/or post irradiation mechanical testing). • Fabrication of pins using prototypic feedstock materials at laboratory-scale. Pin-scale irradiation testing at relevant environment. Primary performance parameters with representative composistions under normal operating conditions quantified. Fuel behaviour models developed for use in fuel performance code(s).	+	+	+	+
6	Fuel assembly was fabricated and tested • Prototype construction fully tested out pile. • Fabrication of pins using prototypic feedstock materials at laboratory-scale and using prototypic fabrication processes. Pin-scale irradiation testing at relevant and prototypic environment (steady-state and transient testing). Predictive fuel performance code(s) and safety basis establishment.	+	+	+	+
7	Fuel assembly successfully tested • Prototype successfully demonstrated e.g. lead use assemblies have performed successfully in a test reactor or prototype reactor.	+	+	+	+

TRL	Objectives and definition by Carmack [29] and Sheperd [30]	MANUFACTURING	SAFETY LIMITS	MODELS	EXPERIMENTS
	• Fabrication of test assemblies using prototypic feedstock materials at engineering-scale and using prototypic fabrication processes. Assembly-scale irradiation testing in prototypic environment. Predictive fuel performance code(s) validated. Safety basis established for full-core operations.				
8	Fuel assembly was fabricated for ALLEGRO and irradiation started • Actual system constructed and commissioned e.g. assemblies fabricated in reload quantities, may include irradiation with only limited success. • Fabrication of a few core-loads of fuel and operation of a prototype reactor with such fuel.	+			+
9	Fuel assembly successfully irradiated in ALLEGRO reactor • Successful operation of actual system e.g. assemblies have performed successfully under irradiation in reload quantities (demonstrated by surveillance programme). • Routine commercial-scale operations. Multiple reactors operating.				+
10	Long term successful use of fuel • Widespread, reliable and long-term operation of many actual systems e.g. long-term use of a fuel within a commercial reactor fleet/fleets with many thousands of hours of operating experience and data. • Long term use of fuel in nuclear power plants.				+

Table 5: TRL levels and the corresponding categories from CNRA review

TRL	Objectives and definition by Carmack [29] and Sheperd [30]	ALLEGRO / GFR fuel status
1	*Promising materials were identified*	This is done for ALLEGRO fuel with 2 exceptions – selection between oxide and carbide fuel and decision on fuel cycle.
2	*Fuel and cladding designs were selected*	Very basic exists for refractory fuel element, less than it for pin-type refractory fuel assembly. CEA design of honeycomb fuel assembly exists. Specific manufacturing route has not been selected yet and detailed design does not exist.

TRL	Objectives and definition by Carmack [29] and Sheperd [30]	ALLEGRO / GFR fuel status
3	*Fuel and cladding was successfully tested in reactor*	Limited existing base is available. Oxide fuel by itself is well tested (some testing is still desirable for high-Pu content MOX). Carbide fuel experience sufficient for basic understanding of its behaviour, but no modern product has been manufactured or tested according to ALLEGRO specifications. Which do not exist. SiC_f/SiC experience is growing fast under LWR temperatures. Extension to GFR temperature and fast spectrum is needed (incl. joint leak tightness) No data on long term fuel-cladding interactions for refractory fuel at GFR parameters. No data on fission product retention by SiC_f/SiC at GFR temperatures.
4	*Fuel rod was fabricated and tested*	Not reached yet. Out of pile tests show that the hermetic sealing on SiCf/SiC tubes is possible up to 750°C (LWR program, higher temperatures not impossible, but were not tested so far).
5	*Fuel rod was successfully tested in reactor*	Not reached yet. Unfueled SiC_f/SiC rodlets tested in MITR at LWR temperatures. Oxide fuelled SiC_f/SiC to be tested in ATR at LWR temperatures. No data on fuel rod scale under GFR conditions for either oxide or carbide fuel in SiC_f/SiC .
6	*Fuel assembly was fabricated and tested*	Not reached yet.
7	*Fuel assembly successfully tested*	Not reached yet.
8	*Fuel assembly was fabricated for ALLEGRO and irradiation started*	Not reached yet.
9	*Fuel assembly successfully irradiated in ALLEGRO reactor*	Not reached yet.
10	*Long term successful use of fuel*	Not reached yet.

Table 6: TRL levels and ALLEGRO/GFR fuel status

6 SUMMARY AND CONCLUSIONS

Key element of the GFR development is the introduction, testing and qualification of appropriate refractory fuel that can reliably withstand the high temperature and high dose conditions in the reactor for long time and allows for the safe operation of the reactor (i.e. provides sufficient grace times for reactor safety systems in the abnormal and accidental conditions).

The concept of the ALLEGRO as an European GFR demonstrator unit assumes that the refractory (ceramic) fuel system design will finish its qualification process by the irradiation in several experimental positions in the starting ALLEGRO core(s), which will use SFR based fuel system (UOX/MOX steel clad cores). This process is feasible, but low melting point of steel clad cores is challenging from the safety point of view. An alternative solution could be the start-up of the ALLEGRO with full refractory core. In that case the full qualification procedure of the refractory fuel has to take place before the start-up of ALLEGRO.

However, even if the MOX-UOX steel clad ALLEGRO cores are used for the final steps of the refractory fuel system qualification, the irradiation of the refractory fuel system components and eventually whole pins must start well ahead in other research reactors, otherwise the qualification procedure of refractory core will take decades after the ALLEGRO start.

The qualification of the fuel us not limited to irradiation experiments by far, but this type of the experiments will be most difficult to perform due to limited number of the available facilities and their technical and safety limitations. Three main families of irradiation experiments needed were identified:

- Fuel behaviour with burnup (thermal properties, swelling, creep, microstructural evolution, fission product behaviour…).
- SiC and SiC-SiC behaviour under irradiation (mechanical properties including fatigue, growth, creep, thermal properties, strength and hermeticity of joints…)
- Semi integral and integral pin experiments (gross failure limits, leak-tightness, long-term fuel-cladding chemical interactions….)

In principle, two main emerging material test reactors could play a major role in these experiments - JHR and MBIR (or BOR-60 for material irradiation). However, the development of the experimental devices used to achieve GFR conditions on water cooled thermal research reactors (aiming at JHR) or in sodium cooled fast reactors (aiming at MBIR and BOR-60) should start as soon as possible.

JHR is currently focused on the LWR experiments and raising temperatures to GFR is a challenge. While irradiation of the structural materials (SiC) will be readily possible up to 450°C and in future up to 1000°C with a reasonable dpa rate, possibilities of the fuel irradiation will be limited to small samples for separate effect studies, unless significant effort is focused towards design of a dedicated GFR test vehicle. Even with such device, fast spectrum will not be attainable for integral fuel element studies. Nonetheless, since both CVR and UJV are members of the JHR consortium with a share in the reactor irradiation time, this option will be pursued further, at least to the level of SiC irradiations and small scale fuel testing. Also, Euratom controls a share in JHR and should support the development of GenIV testing capabilities of JHR.

Ideally, a helium-cooled loop should be constructed in a fast-reactor. Preliminary negotiations were started (memorandum of understanding was singed) between V4G4 consortium and

Russian partners on the construction of a He loop in the MBIR reactor, which is under construction in Dimitrovgrad. The possibility of carrying out transient tests may be also planned in MBIR and the Dimitrovgrad hot cell facilities could be used for post-irradation examinations. The potential capabilities of MBIR due to its fast netron spectrum are very attractive. However, the organisation of international research programmes around MBIR is in a much less developed state compared to JHR, and further discussions would be necessary to specify how the GFR fuel qualification needs can be supported by MBIR tools.

SFR fuel development leveraged on irradiation of new fuel element designs or fuel assembly designs in existing SFRs. Similar irradiation of GFR refractory fuel in SFR would provide data on the fuel behaviour in limited temperature range, but at large scale. Such process may be also feasible – to qualify refractory fuel up to SFR temperatures, use it to start ALLEGRO at lower parameters (equivalent to now envisaged steel clad cores, but with higher safety margins) and gradually extend the qualification up to higher temperatures in experimental positions with reduced coolant flow rate.

The main steps fuel qualification process for GFR fuel assembly with SiC cladding and carbide fuel pellets were identified using the TRL methodology and taking into account the categories introduced by the CNRA working group.

The following decision making points could be identified:

a) The selection between UC or mixed carbide pellet could be made taking into account fabrication capabilities and proliferation issues. Oxide fuel has to considered as a near term solution.
b) In the design process the main core parameters and some of fuel assembly design and fuel criteria will be set, but they can be changed based on qualification results. And vice versa, change in reactor design will influence the fuel qualification requirements.
c) The decision on fabrication process and the selection of fuel supplier will have important impact on the whole qualification process. The products of the given supplier produced by the agreed technology will have to checked, even if similar products were already tested,
d) Decision on the necessity of irradiation testing of a full fuel assembly in a separate He cooled loop in a fast reactor, or testing the full assembly only in the ALLEGRO reactor.
e) The selection of computer codes and their application strategy will be an integral step of fuel qualification.

The listed “pre-existing knowledge” items in the Appendix D were collected from open literature sources. It should be emphasized that the collection of available technical information from the past experience with fast reactor fuel still needs significant efforts and beyond the literature review it should be based on international co-operations and data exchange. Significant “pre-existing knowledge” for the GFR refractory fuel exists up to TRL 3, since the carbide pellets and SiC cladding tubes were fabricated, tested, irradiated separately for other reactor types. The development of SiC_f/SiC cladding is moving forward thanks to the LWR ATF programmes. The oxide fuel technology is well mastered.

The specified “further actions” include different activities (e.g. design, technology development, production, in-pile and out-of-pile testing, post irradiation examination, numerical modelling) and some of them may need launching individual projects to reach the given objectives. In the present document these items are not discussed in details, only the general requirements are mentioned. Necessary “further actions” were specified for all TRL levels in Appendix D, including TRL 1, since the design of the ALLEGRO refractory core is still under development.

The presented qualification procedure is shown as a straightforward process. Iterations are not indicated, but they may take place after unsuccessful steps. Some actions may have to be changed and repeated with other conditions. This is a natural concomitant of such a complex procedure.

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APPENDIX A - SELECTION OF PELLET MATERIAL

There are several fuel types that theoretically could be applied in the refractory core of ALLEGRO. Their advantages and limitations were reviewed in a recent study [7]. The review covered oxide, carbide and nitride fuel pellets as candidates for the refractory core fuel in ALLEGRO core (Table 7).

- The low heat conductivity of **oxide** fuel may limit both fuel element design and power density. Thanks to relatively low power density of ALLEGRO, this is not a showstopper there, but problems may arise in future commercial GFRs with higher power density. Also, the long-term chemical interactions between oxide fuel and SiC at high temperatures have to be considered. Despite these shortcomings, technologically mature oxide fuel is promising near-term solution (some open questions remain for high Pu content MOX fuel, but R&D is ongoing)
- The **nitride** fuel has high fission material content, much better heat conductivity than that of oxides. The operational experience, however, is very modest. The nitride fuel dissociation into metallic phases and nitrogen at less than 2000 K is a significant disadvantage. The production of ^{14}C isotope through the $^{14}N(n,p)^{14}C$ reaction might be also a problem from the point of view of radioactive source term. It is intended to use nitride fuel in the Russian BREST and the Indian FBTR fast reactors. Their experience could be taken into account in later fuel designs.
- The **carbide** fuel has also high fission material content and good thermal conductivity. It was tested and used in much more core than that of nitrides. The carbides are pyrophoric, which complicates the production technology. The carbide fuel has a high swelling and a poor thermal creep, therefore there seems to apparent burnup limit caused by long - term pellet-cladding mechanical interaction concerns.

Taking into account the aspects of fuel production and irradiation experience it was proposed to consider carbide fuel in the design of the refractory ALLEGRO core. The availability of more data on irradiated fuel and the larger fabrication experience supports this decision. Important further step of the GFR pellet development would be the selection of optimal composition of carbide pellet (UC only or mixed type, if mixed type the U-Pu content has to be specified).

The oxide fuel needs to be considered as an alternative, even though it is not a perfect solution in long term, the mature manufacturing process and sufficient quantity of experimental data and models are significant benefits.

	Oxide	Nitride	Carbide
Melting temperature	high	high	high
Thermal conductivity	low	high	high
Swelling due to irradiation	moderate	high	high
Pyrophoricity	no	no	yes
Dissociation	no	at 2000 K	no
Operational experience	wide	very moderate	moderate
Experimental testing	wide	very moderate	moderate
Fabrication experience	wide	very moderate	moderate
Reprocessing	PUREX	PUREX	PUREX non applicable

Table 7: Comparison of oxide, nitride and carbide fuel [7]

APPENDIX B – SiC_F/SIC CLADDING DEVELOPMENT

SiC Ceramic Matrix Composite (CMC) is a candidate GFR cladding material due to its desirable high temperature mechanical properties and small neutron absorption cross-section [8]. Due to its high temperature strength and no-creep behaviour, SiC cladding may maintain fuel rod integrity under accidental conditions without radioactive materials release and structural degradation well beyond the safety limits of the SS cladding. Radiation damage in SiC typically saturates at around 1 dpa which is about 6 months of irradiation in a commercial LWR reactor, the saturated volumetric swelling drops from 2% at 300°C to 0.5% at 1000°C to. Thus, SiC cladding can maintain its dimensional stability and mechanical strength after ≈1 dpa [9]. It must be noted that these results are strongly temperature dependent and also no irradiation in fast spectrum has been performed on recent SiC materials.

Manufacturing processes of SiC/SiC are assumed to be chemical vapor infiltration (CVI), polymer infiltration and pyrolysis (PIP), melt infiltration (MI), nano-infiltration, and transient eutectic-phase (NITE). Various types of production methods were developed to give fibres, interphases and matrices suitable for the nuclear industry. Different SiC_f/SiC composites are produced worldwide but there are only a few companies which are able to produce SiC_f/SiC for the nuclear industry [10]. There is a challenge to produce long SiC_f/SiC composites which satisfy the very strict geometrical standards for straightness and other geometrical parameters of the ceramic tubes.

The main identified challenges are the low pseudo-ductility, and relatively poor thermal conductivity under neutron irradiation. The potentially low thermal conductivity of SiC_f/SiC composites leads to elevated centreline temperatures of the fuel, but more importantly to increased temperature at the point of fuel-cladding contact. With already high coolant temperature (850°C), the chemical interaction between the fuel and the fission products and the cladding become important. In HTR, release of silver and to lesser extent other metals through the SiC barrier of the TRISO particles is the most prominent fuel performance issue. The dimensional scales are more favourable in GFR (almost 1 mm thick cladding tube vs several tens of micometers thick layer), but this factor must be considered nonetheless.

The resistance of SiC_f/SiC cladding to PCMI failure has to be studied because of SiC_f/SiC susceptibility to brittle failure. SiC_f/SiC cladding would not balloon because it is not susceptible to creep or strength degradation at high temperature [11]. Accordingly, the size of a rupture opening is expected to be very small even if SiC_f/SiC cladding ruptures due to excess internal pressure, etc. Regarding postulated accidents, tests simulating LOCA and RIA, and also separate effects tests focusing on the critical phenomena that could be damaging to the coolable geometry are necessary in order to develop safety criteria for SiC_f/SiC clad fuel.

Several actions are needed for the introduction of SiC_f/SiC as cladding for GFR fuel, some of them are listed here:

- Since the SiC_f/SiC cladding tubes are under development for the accident tolerant fuel (ATF) of LWRs [12], the ongoing ATF related activities can produce valuable results for GFRs too, especially in the area of the technology of the SiC joining.
- Experiments and calculations should be performed in order to identify limits and criteria to avoid the loss of cladding integrity and fuel structure, and also to make sure that the model predictions are accurate enough to be used for licensing analysis.
- Optimal fabrications technology should be developed for cladding and assembly structural components made of SiC.

- Risk of severe accidents would be decreased by using SiC_f/SiC cladding. Data on the behaviour of SiC_f/SiC cladding at very high temperature are necessary for safety evaluations.

APPENDIX C – DETAILED REVIEW OF THE JHR EXPERIMENTAL CAPABILITIES WITH RESPECT TO ALLEGRO/GFR NEEDS

<table>
<tr><th>Family of experiments</th><th>Topic</th><th>Experimental objectives</th><th colspan="2">GFR perspective</th></tr>
<tr><td rowspan="4">Fuel material basic properties</td><td>Fuel Thermal properties</td><td>Thermal properties evolution under irradiation</td><td>Non-oxide fuels only</td><td rowspan="4">The soft-spectrum of JHR does not pose significant limitation in data interpretation here. It will be important to reach representative temperatures.

Design of simplified fuel irradiation device for JHR is ongoing. This device will allow irradiation of larger quantity of small fuel samples without complex in-pile instrumentations and will be beneficial for fast scoping studies, for example for the investigation of impact of the fabrication process parameters on key fuel performance indicators (swelling, fission product retention...). Such device will extremely useful for non-oxide fuel qualification on small sample scale. However, no roadmap for introduction of this device exists so far and its design has not been finished up to date.

The only device which will be available for these tests shortly after JHR startup is MADISON. Double - clad pins will be needed to allow testing of non UO2/MOX fuels for the safety reasons and also to provide representative temperatures. Such design is in theory possible for small samples, but will require design modifications and licensing efforts.</td></tr>
<tr><td>Fuel Swelling and creep</td><td>Stress free growth

Thermal and irradiation creep properties (under loading)</td><td>Non-oxide fuels only</td></tr>
<tr><td>Fuel Fission product (FP) effects</td><td>Fuel swelling
He and/or gaseous FP migration and release
Fission product compounds and local thermo-chemistry</td><td>Non-oxide fuels only</td></tr>
<tr><td>Specific Fission Product / Fissile Material effects / fuel - cladding interactions</td><td>Fission products distribution in fuel
Permeation mechanisms through a barrier (SiC…)
Role of FP in fuel-clad gap (JOG creation mechanisms…)
Fissile-fertile interaction</td><td>High priority for both oxide and non-oxide fuels.</td></tr>
<tr><td>Cladding and structural materials basic properties</td><td>Irradiation effects on cladding and structural materials</td><td>Dimensional and structural stability and microstructure, mechanical properties</td><td colspan="2">Small samples could be irradiated in MICA device within the JHR core in the epithermal spectrum with up to 12 dpa/year with temperature up to 450°C (650°C and later 1000°C will be possible with design modifications).</td></tr>
</table>

	Irradiation effects on joints	Joint strength	"Fast" spectrum possible with shielding, but at the cost of dpa rate. MICA device allowing on-line load control for in-pile creep studies will not be available at the start of the JHR
Integral pin testing – normal operation	Fuel Rod / Pin: Integral performance of selected fuel pin design under normal operating conditions: Thermal-mechanical aspects and thermal-chemical aspects	Global thermal performance, fission gas release FP chemical behaviour Fuel-cladding compatibility FP chemical interaction with cladding Specific compounds formation and release: CO, CO_2…	Significant design modifications would be needed to reproduce GFR clad temperatures in the LWR loop of JHR devices. It is not impossible from the technical point of view (AGR pins had been tested in Halden reactors in nearly representative conditions), but safety restrictions might prevent it. **Design of dedicated GFR experimental device would be needed.** Also, the neutron spectrum will be different affecting cladding dimensional changes. In summary - limited studies could be performed at current fleet JHR devices, but not with fully representative spectrum (impacting cladding behaviour) and temperatures (impacting both cladding and fuel behaviour)
	Performance of cladding and fuel assembly structural materials in normal conditions - Geometrical deformations	Irradiation induced growth Cladding creep Pin / Plate bowing	Pin-scale experiments would be impossible with GFR conditions (temperature and spectrum). The creep and growth of cladding and FA structural materials is strongly dependent on these parameters. He loop would be needed. Even with He loop, the spectrum would not be representative of GFR.
Integral pin testing – abnormal operation and accident conditions	Power ramps and transients, power to fuel melting	Key properties and phenomena characterization in abnormal and accidental: **Fission gas and volatile FP release for source term determination** Cladding integrity, failure initiation Fuel creep, plasticity, restructuring Fuel element integral thermal-mechanical behaviour Fuel to cladding chemical interaction	Test up to the fuel failure in ADELINE device are foreseen only for the UO2/MOX fuels so far. Testing of non-oxide fuel up to failure in water loop would be problematic. A double - clad pin would have to be used be used in ADELINE (this possibility must to be confirmed by design safety analysis, the device is limited to 12.5 mm capsule OD) or dedicated He loop would be needed.

	Failed fuel pin behaviour in permanent conditions	Long term behaviour of failed fuel pin in normal operation including transients	ADELINE loop will be used to study the release and transport of the activity from the LWR fuels. For GFR pins, a He loop would be needed for a full scale experiments. However, it must be noted that the motivation for this family of experiments is much lower in GFR. In LWR, cladding and fuel degrade after the primary failure. In GFR, pin failure will not have any impact apart from direct activity release.
	LOCA conditions, LOFA conditions	Safety criteria confirmation Source term determination	GFR LOCA is different from LWR LOCA in term of fuel experimental needs. For GFR LOCA, the scenario is basically an adiabatic heat-up. Most of the studies may be conducted out of pile (cladding thermomechanical response, fuel-cladding chemical interaction...). An in-pile test would be beneficial for the validation purposes and for the source term quantification only. A new dedicated experimental device would have to designed for JHR, but such experiment is in theory possible.

APPENDIX D – DETAILED OVERVIEW OF GFR FUEL TRL: CARBIDE PELLETS IN SiC_F/SiC CLADDING

TRL 1: RESEARCH IDENTIFIES THE BASIC PRINCIPLES THAT UNDERLIE THE TECHNOLOGY

Objective: promising materials were identified

Pre-existing knowledge:

SAFETY LIMITS

- The following general requirements can be applied to the refractory GFR fuel.
    - High enough fissile content in the fuel to allow economical operation of the reactor (note that this is currently a grey area since the fuel cycle type has not been decided)
    - Low neutron absorption and scattering cross section for the structural materials.
    - Irradiation resistance.
    - Mechanical strength at target GFR temperatures.
    - High melting point, good thermal conductivity, stability at high temperature.
    - Fuel-cladding chemical compatibility.
    - Reprocessability of materials in order to close the fuel cycle.
    - Capabilities to withstand long term storage of spent fuel.
- The geometry of GFR fuel was identified as fuel bundles (subassemblies) with cylindrical pellets and cladding tubes [4]. The geometry is similar to that of SFR subassemblies, but the materials must be different.
- The GFR design assumes operation at very high temperature (850 °C core outlet temperature) and high dose rates (22 dpa on SiC cladding) [4].
- Since the traditional LWR or SFR fuel cannot be applied, other materials were reviewed. The potential pellet materials were the oxides, carbides and nitrides [7].
- Among the cladding materials the $SiC–SiC_f$ tubes are the first candidates[36].

Further actions:

SAFETY LIMITS

- During the ALLEGRO design process the expected irradiation doses, temperatures, thermal and mechanical loads could be precised.

TRL 2: Practical applications suggested and concepts formulated

Objective: fuel and cladding designs were selected

Pre-existing knowledge:

MANUFACTURING

- The potential pellet materials were the oxides, carbides and nitrides. Their physical properties, operational and fabrication experience, reprocessibility were compared and the carbide was selected [7].
- UC and mixed carbide pellet fabrication technology exists, but needs special equipment and conditions (e.g. inert atmosphere).
  - In 1960, when research on carbide fuel was initiated, three different methods were followed, namely melting casting, metal hydriding–dehydriding, and carbothermic reduction of oxide. The carbide produced by the latter two techniques is processed further by powder metallurgy techniques for the manufacture of fuel pellets [37][38][39].
- SiC_f/SiC tube production technologies have been developed [36].
  - Processing routes presently available for industrial production of SiC composites are chemical vapour infiltration (CVI), nano-infiltration and transient eutectic-phase process (NITE), melt infiltration (MI) or occasionally termed reaction sintering (RS) or liquid silicon infiltration (LSI) and polymer-impregnation and pyrolysis (PIP) [36][40][41][42].

MODELS

- The main phenomena that take place in the carbide pellets were identified. The physical properties, like thermophysical properties and thermochemistry of carbide fuels were determined [39][43][44].
- Basic material properties for modelling purposes are available for SiC cladding (mechanical properties, thermal properties, chemical stability under normal and off-normal operation conditions, hermeticity, and irradiation resistance) [45].
- Several computer codes include material properties for carbide pellets:
  - TRANSURANUS includes subroutine to determine the structures of carbide and mixed carbide fuel. A special subprogram gives the heat of melting, correlations for the thermal conductivity, melting temperatures, swelling and thermal expansion of carbide and mixed carbide [46].
  - In the new version FUROM-FBR-2 some material properties (melting point, thermal conductivity) are available for carbide or mixed carbide pellets [47].
  - Fuel property data (density, thermal conductivity, thermal expansion, Young's modulus, Poisson ratio, creep, densification, fission gas diffusivity) for carbide fuel were introduced into the FEMAXI-6GA code [48] .
- A number of computer codes include material properties for SiC cladding:
  - The dependence of thermomechanical properties for each SiC layer on temperature and neutron fluence is considered in the BISON fuel performance code. [49]
  - SiC properties were implemented in FRAPCON and FRAPTRAN codes to simulate U_3Si_2-SiC design during normal, power ramp and RIA conditions [50].

    - The FROBA code was updated into FROBA-ATF, transient heat transfer model, multi-layer model and models of cladding material properties were implanted to simulate performance of UO_2-SiC fuel rods under normal and accident condition [51].

Further actions:

MANUFACTURING

- Optimal technology must be selected for production of long SiC_f/SiC and for their sealing tubes.
- The carbide fuel type has to be selected (UC or mixed, U and Pu containing carbide).
- Optimal technology must be selected for production of carbide pellets.
- Reprocessing technology and long term storage solutions must be developed.

MODELS

- Computational analyses (fuel behaviour simulations) are needed for ALLEGRO conditions and scenarios to identify the typical parameters and material property ranges for the carbide pellets and SiC cladding in order to support the experimental programmes with establishing test matrices.
- Further extension of computer codes with SiC, UC and mixed carbide pellet properties, correlations and models is needed.
- Data and models on high temperature fission product retention in SiC–SiCf are needed

TRL 3: BASIC COMPONENTS FABRICATED AND SUCCESSFULLY DEMONSTRATED

Objective: fuel and cladding were successfully tested in reactor

Pre-existing knowledge:

MANUFACTURING

- SiC–SiCf cladding tubes were produced in several laboratories:
  - Since the monolithic SiC has low fracture toughness, composite structures were introduced with using strong SiC fibres that reinforce a SiC matrix to form a SiC_f/SiC composite [36][52][53].
  - Triplex tube samples, monolith-only samples, and SiC/SiC bonding samples were fabricated in the USA [54].
  - Nuclear-grade SiC components were manufactured at the Kyoto University [55].
  - KAERI fabricated nuclear grade SiC_f/SiC duplex and triplex cladding tubes [41].
  - Sandwich technology was developed in France to produce leaktight structure [56][57].
- UC and mixed carbide pellets were produced in several countries:
  - UC pellets of high and low (4000 and 400 ppm) oxygen content were made in a fabrication line in Belgium [58]
  - $(Pu_{0.7}U_{0.3})C$ and $(Pu_{0.55}U_{0.45})C$ pellets were produced for driver fuel in the Fast Breeder Test Reactor (FBTR) in India [59].
  - Mixed carbide fuels were manufactured in Germany for development of fast reactor technology [60].
  - (U,Pu)C carbide fuel fabrication technology was developed at CEA, France [61][62].
  - Mixed carbide fuel for a joint (US-Swiss) irradiation tests in the US Fast Flux Test Facility (FFTF) were produced in the USA via the powder-pellet (dry) route, and in Switzerland using the internal gelation (wet) route [63].

MODELS

- Numerical simulation of irradiation tests was carried out for several reactor measurements with carbide fuel using the FEMAXI-6GA code [48].

EXPERIMENTS

- The carbide pellets were tested in EBR-II, TREAT, FFTF, JMTR, OSIRIS, RAPSODIE, PHENIX and FBTR reactors with different stainless steel claddings [39].
  - During the 1970s and early 1980s over 470 MC fuel rods were irradiated in EBR-II using a range of parameters, sodium or helium bonding, and cladding made from Type 316 stainless steel, PE-16 (a nickel-based alloy used in the U.K.), D9 stainless steel, or D21 stainless steel [39][64][64][65].
  - Ten transient-overpower tests involving MC fuels were conducted in TREAT using fuel irradiated in EBR-II to burn-ups ranging from 0 to 12 at.%, primarily for the purposes of establishing that cladding breach would occur at a margin above that of the FFTF plant protection system settings (at 115% and 125% overpower) [39][64][64][65].
  - Over 200 MC fuel rods were irradiated in FFTF in two assemblies: the ACN-1 experiments with rods fabricated using Type 316SS and D9 cladding and the FC-1 test, which was a full-size, 91-rod FFTF assembly using Type 316SS and D9 cladding and ducts [39][64][65].

- The AC-3 fuel bundle was irradiated in the Fast Flux Test Facility (FFTF) during the years 1986–1988 for 630 full power days to a peak burn up of ~8 at.% fissile material [66]. The test was composed of 91 full-size, D9-clad rods of which 25 rods contained sphere-packed fuel and 66 rods contained pellet fuel. That assembly was irradiated to the goal 9 at% burn-up without breach [39]. All of the pins, irradiated at linear powers of up to 84 kW/m, with cladding outer temperatures of 465 °C appeared to be in good condition when removed from the assembly [66].
- Irradiation test program of uranium-plutonium mixed carbide fuels at JAERI is shown in the next table [67][68]:

Fuels	**Irrad. No.**	**Objective**	**Linear heat rate (W/cm)**	**Burnup (MWd/kg)**	**Pin No.**	**Diameter (mm)**	**Irradiation year**	**Reactor**
$MC_{1.0}$* and $MC_{1.1}$*	1	Preliminary	450	10	2	6.5	1983	JRR-2
	2	Medium heat rate	650	13	2	9.4	1985-1986	JRR-2
	3	Medium burnup	650	26	2	9.4	1986-1989	JMTR
	4	High burnup (Chamfered pellets)	650	50	2	9.4	1986-1990	JMTR
	5	High burnup (Thermally stable pellets)	650	50	1	9.4	1988-1994	JMTR
MC**	8	High heat rate and medium burnup (Fast neutron irradiation)	800	30	3	8.5	1993-1994	JOYO

* M= $PU_{0.2}+U_{0.8}$
** M= $PU_{0.2}+{}^{235}U_{0.2}+U_{0.6}$, U= natural uranium

- 27 fuel test irradiation rigs in total have been irradiated in Joyo on purpose to investigate irradiation behaviours in various compositions and sizes of FR fuels. The maximum burn-up (pellet peak) achieved to 140GWd/t. The maximum linear heat rate exceeds 700W/cm in irradiation tests of the mixed nitride and the mixed carbide fuel [69].
- Between the years 1960 and 1970, about 80 MC fuel pins (80% sodium-bonded), including two NIMPHE pins, were irradiated in MTRs (OSIRIS, SOLOE) and then in Rapsodie and PHENIX reactors [39][70]. The second NIMPHE 2 irradiation is for CEA and ITU nitride pins, and also for carbide pins manufactured by ITU, all helium bonded, which should allow comparison of the behaviour of carbide and nitride, in this case at a higher linear power of 730 W/cm [71]. In France, the (U,Pu)C fuel pins with 71% T.D. smear density reached a burn-up of 12 at.% with clad deformation of 1 to 3% [38].
- The FBTR reactor in India uses an internationally unique fuel in the form of Pu rich carbide [72]. The carbide fuel of FBTR has seen a burn-up of 155 MWd/kg. The

current core is rated for 30 MWt. Mixed carbide fuel, being a unique fuel of its kind without any irradiation data, it was decided to use the reactor itself as the test bed for this driver fuel. Hence, the core was redesigned as a small carbide core. As against the original design of 65 MOX fuel subassemblies rated for 40 MW t, the small carbide core had 22 fuel subassemblies with 70% PuC and 30% UC composition (designated as Mark-I fuel) during first criticality. This small carbide core was rated for 10.2 MW t, with the peak linear heat rating limited to 250 W/cm. With a view to raise the reactor power to 40 MW t, it was decided, in 1995, to go in for a full carbide core of 78 fuel subassemblies. The fuel composition chosen was 55% PuC + 45% UC (designated as Mark-II fuel). The Mark-I fuel in the centre was retained to continue the irradiation for assessing its ultimate burn-up capability before phasing it out. Mark-II fuel was added at the periphery. The allowable peak linear heat rating of the Mark-I fuel has also been revised up to 400 W/cm and burn-up limit of 25 GWd/t was raised to 155 GWd/t based on the fuel performance.

  - The German mixed carbide fuel irradiation program (75% TD smear density, 800 W/cm) was successfully tested under power cycling and transient conditions [38] [39][73]. The fuel pins were irradiated in KNK II rector. The pin concept with cold-worked austenitic steel (1.4970) cladding, pellet diameter 7.0 mm, pellet density 84% TD, fuel–cladding gap of 400 mm, helium bond, smear density 75% TD, pin diameter 8.5 mm, and clad wall thickness of 0.55 mm evolved. The 19 carbide pins were irradiated in the fast neutron flux of the KNK II reactor to a burn-up of about 7 at% without any failure in the centre of a KNK “carrier element” at a maximum linear rating of 800 W/cm [74].
  - In UK, the mixed carbide pin irradiation program was successful with low smear density (70% T.D.) vibro-packed fuel of about 1000 W/cm, with target burn up of 100 GWd/t [38].

- SiC_f/SiC cladding developments are in progress in many countries.
  - The SiC_f/SiC cladding applicability is under investigation for different reactor types and is a candidate for accident tolerant fuel material in LWRs [36].
  - French SiC_f/SiC cladding samples irradiated in BOR-60 [53].
  - A number of SiC/SiC samples were exposed to PWR coolant and neutronic conditions using an in-core loop in the MIT research reactor (MITR-II)[54].
  - Testing of SiC_f/SiC cladding in high temperature He was carried out in Hungary, focusing on the effect of impurities. The applicability of SiC_f/SiC cladding in high temperature He was confirmed [75].
  - The SiC/SiC composites were investigated as structures and flow channel insert (FCI) for fusion reactor blankets, control rod sheath in advanced gas-cooled thermal reactors, core components in gas-cooled fast reactors (GFR), and fuel cladding for various fission reactors, including the light water reactor (LWR) [43].
  - Initially developed as fuel cladding materials for the Fourth generation Gas cooled Fast Reactor (GFR), this material has been recently envisaged by CEA for different core structures of Sodium Fast Reactor (SFR) which combines fast neutrons and high temperature (500°C) [53].
  - The Advanced High-Temperature Reactor (AHTR) is a new reactor concept that uses a liquid fluoride salt coolant and a solid high-temperature fuel. Several alternative fuel types are being considered for this reactor. One set of fuel options is the use of pin-type fuel assemblies with silicon carbide (SiC) cladding [40].

  - Silicon carbide (SiC) has been investigated for use in both fission and fusion applications and recently has been considered as cladding material for advanced light water reactors (ALWR) working with accident tolerant fuel (ATF) [10][12][41] [76][77][78][79][80].
  - SiC_f/SiC type duplex and triplex type claddings were produced in KAERI for nuclear fuel. This cladding type is a candidate material for the refractory core of the ALLEGRO reactor. High temperature testing in He atmosphere with different impurities, detailed scanning electron microscope analyses of some cladding samples and mechanical testing of all samples were carried out at MTA EK [75][81].
  - SiC material is used in gas-cooled high temperature pebble bed reactors as one layer in the TRISO fuel [82]. The TRISO coatings were applied at ORNL. Nominal coating thicknesses were 100 μm for the porous carbon buffer, 40 μm for the inner pyrolytic carbon (IPyC) layer, 35 μm for the SiC layer,and 40 μm for the outer pyrolytic carbon (OPyC) layer.
- The applicability of SiC cladding in gas cooled reactor was addressed in several experimental programmes in the past:
  - SiC_f/SiC cladding tubes are produced at KAERI and their behaviour is tested in the framework of extensive experimental series [83]. The work performed at KAERI with SiC_f/SiC composites for nuclear applications includes the development of light water reactor (LWR) fuel cladding and in-core components for very high temperature reactors (VHTR). One series of KAERI tests focused on the investigation of behaviour CVD (Chemical Vapour. Deposition) SiC and SiC_f/SiC composite in the oxygen containing He and air. In air atmosphere positive mass gains were observed above 1100 °C. It was concluded by KAERI experts that long-term experiments and tests at higher temperatures are required to verify the chemical compatibility of SiC_f/SiC composites with the VHTR/Fusion relevant He coolant chemistry [83].
  - The chemical compatibility aspects of CVD β-SiC and SiC_f/SiC composites with a VHTR specific helium coolant were examined at KAERI in another test series [84]. The specimens were exposed to helium gas containing 20 Pa H2, 5 Pa CO, 2 Pa CH4, and 0.02–0.1 Pa H2O, which is an expected VHTR coolant chemistry. Oxidation tests were carried out at 900 °C and 950 °C for up to 250 hours. β-SiC and SiC_f/SiC composites had an excellent compatibility with the expected VHTR helium coolant environment. The oxidation of β-SiC as a matrix material of the SiC_f/SiC composite reacted in a passive oxidation regime owing to the presence of water vapour. A condensed version of the oxide SiO2 formed at an early stage of oxidation and the growth of this oxide layer was very limited as the oxidation time increased up to 250 h. The recession of the pyrolytic carbon interphase of SiC_f/SiC composite was not observed [84].
  - High temperature (1300–2000 K) tests on massive SiC samples (sintered and CVD) were performed in France. The tests were coupled to SEM (Scanning Electron Microscopy), XPS (X-Ray Photoelectron Spectroscopy) and XRD (X-Ray Diffractometry) analyses before and after oxidation. It was found that the level of oxidizing species had an important impact on the physico-chemical behaviour of SiC. The investigated SiC samples maintained their structural integrity at high temperature in helium environment with low oxygen partial pressure [85].
  - High-temperature tests of silicon carbide composite cladding under GFR (Gas Cooled Fast Reactor) conditions were performed at KIT (Karlsruhe Institute of

Technology) [86]. In particular, the feasibility of silicon carbide composites is investigated in helium with low amount of impurities (H_2, CO, N_2, O_2, H_2O, CH_4 and CO_2) by means of a thermogravimetric device. The SiC_f/SiC composites specimens were provided by CEA. The temperatures of the tests were in the range of normal operation conditions (900–1000 °C) and accident conditions (up to 1500 °C) of a gas fast reactor. Passive oxidation was detected at 900 °C and 1200 °C, whereas the samples underwent active oxidation and mass loss at 1300 °C, 1400 °C, and 1500 °C. Overall, the results meet the requirements for the aimed application, since the transition temperature from passive to active oxidation is 300 °C higher than the nominal working conditions of GFR.

- Duplex and triplex type SiC_f/SiC cladding samples with 9.9 mm length were produced in KAERI laboratories and provided to MTA EK in the framework of scientific collaboration [75]. The cladding tubes were tested at 1000 °C for 7 h in helium atmosphere with and without gas impurities. The mass gain measurements showed that in case of pure helium atmosphere and with hydrogen or nitrogen impurities small mass reduction was observed. In case of methane impurities the decomposition of methane and the formation of carbon deposits lead to mass gain of the SiC samples

- Several carbide pellets were irradiated in different reactors (EBR-II, TREAT, FFTF, JRR2, JMTR, OSIRIS, RAPSODIE, PHENIX, FBTR) [39].

Fuel type	**Reactor**	**Country/organization**	**Bond**	**Density (%TD)**	**Burnup (at.%)**	**Clad**
MC	RAPSODIE	France/CEA	Na	91.5	12	-
MC	BOR 60	USSR	Na	-	-	OX16H15M3G
MC	EBR II	United States	Na	-	12	PE16
MC	RAPSODIE	TUI	He	86	5	-
MC	KNK II	FZK, Germany	He	85	7	1.4970
MC	EBR II	United States	He	80/87	12	316.20 cw
MC	EBR II	United States	He	81/87	16-20	316.20 cw
UC/MC	BOR 60	USSR	He	-	10	OX16H15M3G-
MC	FFTF	United States	He	80	10	D9
MC	PX	CEA/TUI	He	80/82	-	15/15 Ti
MC	FFTF	DOE/PSI	He	-	10	D9
MC	FBTR	India	He	91/86	16	SS316 cw

Further actions:

MANUFACTURING

- Companies for SiC_f/SiC cladding tubes fabrication should be identified and the details of technology (monolith or fiber, sandwich structure, duplex, triplex, fiber winding patterns and angles, additives) should be fixed.
- Companies for carbide pellet fabrication should be identified and the details of technology should be fixed.

MODELS

- Numerical models should be validated against completed reactor tests with carbide pellets including the burnup effects, wide power ranges, transients.
- The planning of new measurements should be supported by computer code calculations, and post-test calculations should be performed.

- The capabilities of severe accident codes should be extended to cover the behaviour of carbide pellets and SiC_f/SiC cladding in DEC conditions.

EXPERIMENTS

- The carbide pellets produced with the selected technology must be tested out-of-pile and in-pile.
  - Basic material properties of carbide pellets have to be measured.
  - For carbide pellet irradiation tests research reactor capabilities (see irradiation facilities in chapter 5) should be reviewed
  - Irradiation programmes must be defined. Structural changes due to irradiation must be identified. Burnup dependent material properties should be measured.
  - Both on-line measured data and post-irradation examination (PIE) results must be evaluated.
- The SiC cladding produced with the selected technology must be tested out-of-pile and in-pile.
  - Basic material properties of SiC cladding have to be measured.
  - The irradiation behaviour of SiC_f/SiC cladding needs further examinations (e.g. those samples that were irradiated in the BOR-60 reactor recently).
  - The potential irradiation damage should be identified for different SiC structures.
  - The effect of different components (e.g. BN) in SiC_f/SiC cladding should be evaluated.
- Compatibility of SiC_f/SiC cladding and carbide pellets should be proven at high temperature.
- The high temperature behaviour of carbide pellets, SiC_f/SiC cladding and their interactions must be covered by measurements.

TRL 4: Integration of components into a basic system

Objective: fuel rod was fabricated and tested

Pre-existing knowledge:

MANUFACTURING

- Fuel rod with carbide pellets in SiC cladding has not been produced yet.
- Blind-end SiC cladding tube closing technology and buffer bond of high porosity C-based braid were patented by CEA [56][57].

SAFETY LIMITS

- A recent OECD NEA opinion paper pointed out that new performance metrics and regulatory criteria to preserve coolable geometry need to be developed for SiC_f/SiC cladding [87].
- Rod-like fuel geometry was selected considering the planned ALLEGRO core design [88][89][90][91].

MODELS

- Based on the PLEIADES software platform, the fuel performance code CELAENO was developed by CEA for the simulation of fuel elements for gas-cooled fast reactor with SiC_f/SiC cladding and mixed uranium-plutonium carbide pellets [92].

EXPERIMENTS

- Recent results of the development of ATF fuel with SiC_f/SiC can be used. US report [45] summarized various physical, mechanical, and chemical compatibility properties of SiC/SiC composites for LWR cladding applications.
- Carbide fuel pellets in SS cladding with He bonding designs are available from SFRs. Several irradiation were performed with this fuel type all over the world, so in the USA, India, Germany, UK, Russian Federation [37][39].
- Post-test examination of carbide fuel was performed. The performance analysis of the mixed carbide fuel can be best understood on the basis of their burnup period, the structural changes occurring during the burnup and subsequently the swelling of the pin [39][65][66][73][93][94][95].

Further actions:

MANUFACTURING

- The technology of refractory fuel fabrication must be developed. Production of fuel rod with carbide pellets and SiC_f/SiC cladding has to be demonstrated**.**
- Key dimensions and tolerance of fuel components have to be specified.
- Key constituents have to be specified with allowance for impurities.
- Microstructure attributes for materials within fuel component have to be specified for otherwise justified.
- Short fuel rodlets with SiC_f/SiC cladding tubes have to be produced for in-pile experimental purposes.

- SiC tube closing has to be solved for the selected cladding type.
- The need for internal spring in the fuel rod to fix the fuel column should be defined.

SAFETY LIMITS

- The main design parameters (fuel and cladding geometry, enrichment, gas volume, gap size) of ALLEGRO fuel rod have to be fixed.
- Fuel performance envelope has to be defined.
- Fuel criteria for ceramic ALLEGRO fuel rods have to be defined.
    - Radionuclide retention requirements have to be specified
    - Criteria for barrier degradation and failure have to be specified.
    - Criteria have to be specified for ensuring coolable geometry.
    - Criteria has to be provided to ensure control element insertion path is not obstructed
    - The limits should prevent fuel failure and release of radionuclides from the fuel rod.
    - RIA failure limits should be specified.
    - LOCA and severe accident failure mechanisms should be identified and the corresponding limits should be defined.
- Margins to safety limits have to be demonstrated with high confidence
    - Margin to design criteria under conditions of normal operation, including the effects of AOOs.
    - Margin to radionuclide release limits for accident conditions.
    - Ability to reach safe state has to be ensured.

MODELS

- Ceramic fuel rod models have to be developed with appropriate modelling capabilities on
    - Geometry,
    - Materials and
    - Physics.
- Numerical models should be applied to carry out simulation of fuel behaviour in steady state and transient conditions in the ALLEGRO reactor. The calculations should include reactor physics, thermal hydraulics and fuel behaviour aspects.
- The codes should be validated against experimental data:
    - The data used for assessment has to be appropriate.
    - The evaluation model has to demonstrate the ability to predict fuel failure and degradation mechanism over the test envelope.
    - Evaluation model error should be quantified through assessment against experimental data
    - Evaluation model error should be determined through the fuel performance envelope.
    - Sparse data regions have to be justified.
    - Evaluation model has to be restricted to use within its test envelope.

EXPERIMENTS

- Mechanical load bearing capabilities (especially PCMI) and leak tightness must be addressed in experimental series.
- Failure mechanisms (loss of cladding integrity) must be investigated in wide range of parameters.

- Test series are needed to support criteria development.
- Assessment data must be independent of data used to develop/train the evaluation models.
- Data has to be collected over a test envelope that covers the fuel performance envelope.
- Experimental data have to be accurately measured.
    - The test facility must have an appropriate quality assurance program.
    - Experimental data must be collected using established measurement techniques.
    - Experimental data must account for sources of experimental uncertainty.
- Test specimens have to be representative of prototypical fuel.
    - Test specimens should be fabricated consistent with the prototypical fuel manufacturing specification.
    - Distortions should be justified and accounted for in the experimental data.

TRL 5: Basic system successfully demonstrated

Objective: fuel rod was successfully tested in reactor

Pre-existing knowledge:

MANUFACTURING

- None.

SAFETY LIMITS

- None.

MODELS

- None.

EXPERIMENTS

- None.

Further actions:

MANUFACTURING

- Improvement of fuel rod production technology on the basis of learnings from tests in TRL 4.

SAFETY LIMITS

- Update of safety limits may be needed on the basis of learnings from tests in TRL 4.

MODELS

- Conservative model has to be developed to simulate radionuclide retention and release from fuel matrix.
- Conservative model has to be developed to simulate barrier degradation and failure.
- Conservative model has to be developed to predict loss of coolable geometry.
- Numerical models should be used to support the preparation of in reactor tests. Wide range of boundary conditions (e.g. linear heat rates, inlet temperatures) must be covered. Special attention should be devoted to start-up and power changes.
- Post-test calculations of performed experiments must be carried out. Further model developments may be necessary.

EXPERIMENTS

- Helium irradiation loop or capsule has to be constructed in the selected fast reactor
- Irradiation of test rods in a fast reactor should be carried out in high temperature gas loop up to high dpa values.
- Experimental data have to be accurately measured.
  - The test facility must have an appropriate quality assurance program.
  - Experimental data must be collected using established measurement techniques.
  - Experimental data must account for sources of experimental uncertainty.

- Test specimens have to be representative of prototypical fuel.
    - Test specimens should be fabricated consistent with the prototypical fuel manufacturing specification.
    - Distortions should be justified and accounted for in the experimental data.
- The results of irradiation programme have to be evaluated and the operability of fuel rod has to be demonstrated.
- Transient testing for operational transients, incidents and accidents will be needed.

TRL 6: PROTOTYPE CONSTRUCTION

Objective: fuel assembly was fabricated and tested

Pre-existing knowledge:

MANUFACTURING

- None.

SAFETY LIMITS

- The main elements of the fuel assembly design are the hermetically closed cladding tubes, cylindrical pellet column, large free volume inside of the fuel element for gases and the use of fuel assemblies to guarantee the fixed geometrical arrangement and simple handling [7][36][39][41].
- The bundle of pins will be housed in a hexagonal wrapper tube (shroud) made of SiC. The distance between fuel pins should be kept by the application of SiC spacer grids [90][91].
- In the earlier ceramic core design of the ALLEGRO reactor, there are 87 ceramic fuel assemblies. Each assembly contained 90 fuel rods. The active length of the rods was 0.86 m [4][88][89][90][91].

MODELS

- Detailed thermal hydraulic modelling of the ALLEGRO ceramic assembly was performed in Hungary [91].

EXPERIMENTS

- The applicability of SiC type BWR channel box is under investigation. SiC_f/SiC composites are being considered for applications in the core components, including BWR channel box and fuel rod cladding, of light water reactors to improve accident tolerance. In the extreme nuclear reactor environment, core components like the BWR channel box will be exposed to neutron damage and a corrosive environment. To ensure reliable and safe operation of a SiC channel box, it is important to assess its deformation behaviour under in-reactor conditions including the expected neutron flux and temperature distributions [96].

Further actions:

MANUFACTURING

- Key dimensions and tolerance of fuel assembly components have to be specified.
- Key constituents have to be specified with allowance for impurities.
- Microstructure attributes for materials within fuel assembly component have to be specified for otherwise justified.
- The fuel assembly fabrication process must be demonstrated, new workshops/laboratories are needed.
- ALLEGRO refractory subassemblies have to be produced.

SAFETY LIMITS

- Detailed design of the ALLEGRO fuel assembly has to be performed and the main dimensions have to be specified (fuel rod active and total length, diameters, internal pressure, pitch size, number of pins in the assembly, spacer grid geometry and positions).
- Criteria for subassemblies must be specified for normal operational conditions (e.g. allowable change of dimensions).
- Fuel assembly specific failure mechanisms (e.g. fretting, bowing) should be identified, if such phenomena exist.
- Criteria has to be provided to ensure control element insertion path is not obstructed.
- Limitations on mechanical damage (due to e.g. earthquake) should be investigated.
- Subcriticality, coolability and shielding conditions must be specified for transport and storage conditions (e.g. in water pool).

MODELS

- Validation against fuel assembly experimental data.

EXPERIMENTS

- Out-of-pile testing should be carried out. Part of testing (e.g. thermal hydraulic) can be done without carbide pellets.
- Integral bundle test with simulant pellet materials are needed to demonstrate the fuel assembly behaviour under LOCA and severe accident conditions.

TRL 7: PROTOTYPE SUCCESSFULLY DEMONSTRATED

Objective: fuel assembly successfully tested in reactor

Pre-existing knowledge:

MANUFACTURING

- None.

SAFETY LIMITS

- None.

MODELS

- None.

EXPERIMENTS

- None.

Further actions:

MANUFACTURING

- Improvement of fuel assembly production technology on the basis of learnings from tests in TRL 6.

SAFETY LIMITS

- Definition of ALLLEGRO start-up procedure and normal operational conditions (not only for fuel, but for all systems).

MODELS

- Conservative model has to be developed to confirm that the control element insertion is not obstructed.
- Conservative model has to be developed on the prediction of loss of coolable geometry.
- Supporting calculations of reactor start-up (including not only fuel behaviour, but reactor physics, thermal hydraulics).
- Checking of fuel safety criteria for ALLEGRO scenarios (the corresponding scenarios, covering cases must be defined).

EXPERIMENTS

- Refractory subassemblies have to be successfully irradiated in the gas loop of a fast reactor (e.g. MBIR) with in-core monitoring or in prototype reactor (ALLEGRO)
- PIE of irradiated subassemblies have to be carried out (with both non-destructive and destructive procedures).
- The state of the assembly must be checked for meeting the fuel assembly criteria.

TRL 8: ACTUAL SYSTEM CONSTRUCTED AND COMMISSIONED

Objective: fuel assembly was fabricated for ALLEGRO and irradiation started

Pre-existing knowledge:

MANUFACTURING

- None.

EXPERIMENTS

- None.

Further actions:

MANUFACTURING

- Subassemblies for testing in the first core of ALLEGRO have to be produced in reload quantities.

EXPERIMENTS

- Start of irradiation of fuel in the ALLEGRO core.
- Construction of ALLEGRO core (after inactive testing of all ALLEGRO technological components).
- Start-up of the reactor core using detailed on-line measurements for core-monitoring.
- Regular or on-line activity concentration measurements in the primary coolant.
- Shut-down of the reactor for refuelling.
- Appropriate hot cell capabilities have to be established.
- PIE of irradiated fuel assemblies.

TRL 9: Successful operation of actual system

Objective: fuel assembly successfully irradiated in ALLEGRO reactor

Pre-existing knowledge:

EXPERIMENTS

- None.

Further actions:

EXPERIMENTS

- Continuation of irradiation of refractory subassemblies in ALLEGRO.
- Fuel assemblies have to perform successfully under irradiation in reload quantities (demonstrated by surveillance programme).
- Regular PIE of irradiated fuel assemblies.

TRL 10: WIDESPREAD, RELIABLE AND LONG-TERM OPERATION OF MANY ACTUAL SYSTEMS

Objective: long term successful use of fuel

Pre-existing knowledge:

EXPERIMENTS

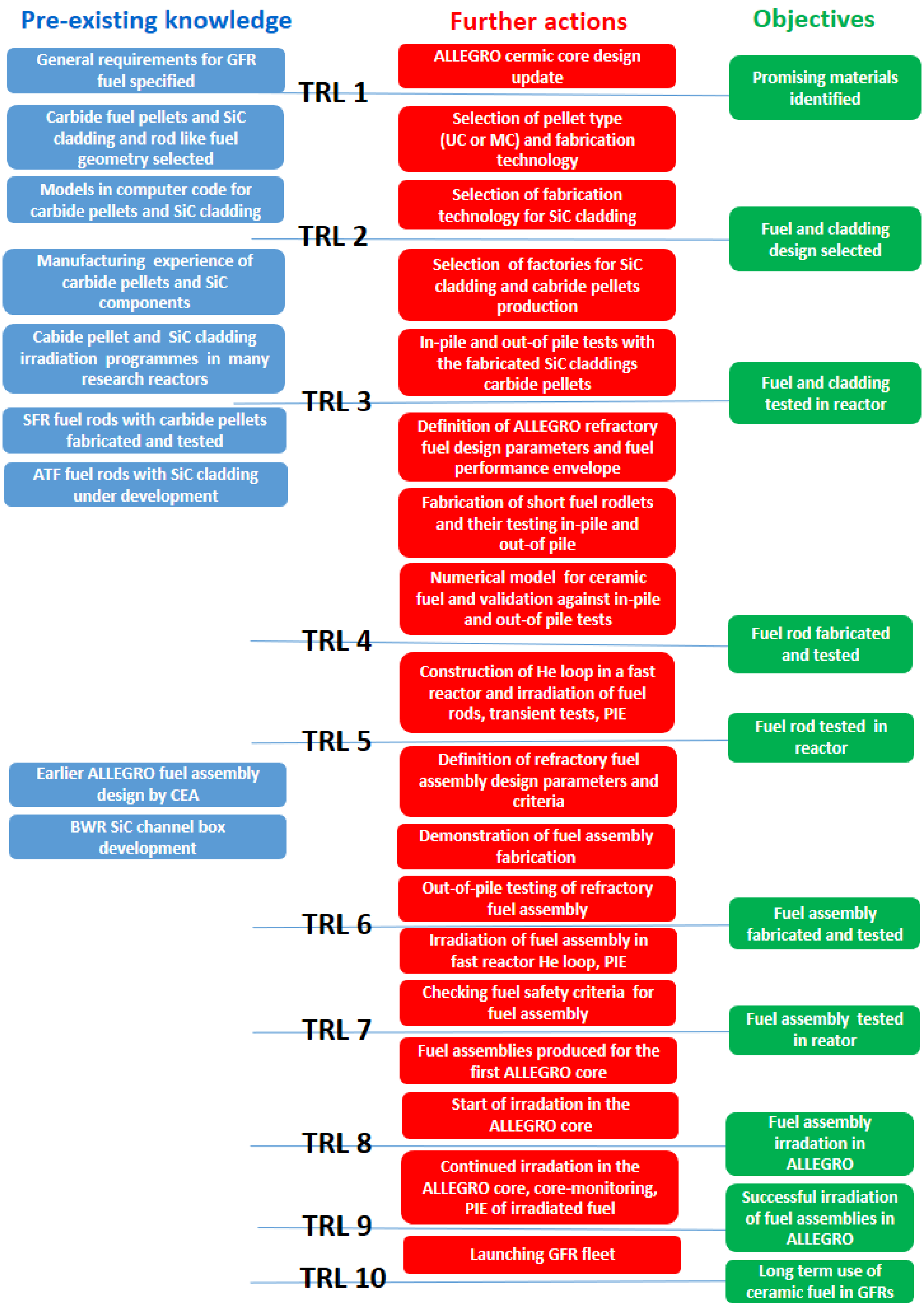
- None.

Further actions:

EXPERIMENTS

- This step can be reached only with the launching of reactor fleet with several GFR reactors.

Summary on carbide fuel with SiC_f/SiC cladding qualification process



APPENDIX E – DETAILED OVERVIEW OF GFR FUEL TRL: UOX PELLETS IN SiC_F/SIC CLADDING

TRL 1: RESEARCH IDENTIFIES THE BASIC PRINCIPLES THAT UNDERLIE THE TECHNOLOGY

Objective: promising materials were identified

Pre-existing knowledge:

SAFETY LIMITS

- The following general requirements can be applied to the refractory GFR fuel.
    - High enough fissile content in the fuel to allow economical operation of the reactor (note that this is currently a grey area since the fuel cycle type has not been decided)
    - Low neutron absorption and scattering cross section for the structural materials.
    - Irradiation resistance
    - Mechanical strength at target GFR temperatures
    - High melting point, good thermal conductivity, stability at high temperature.
    - Fuel-cladding chemical compatibility.
    - Reprocessability of materials in order to close the fuel cycle.
    - Capabilities to withstand long term storage of spent fuel.
- The geometry of GFR fuel was identified as fuel bundles (subassemblies) with cylindrical pellets and cladding tubes [4]. The geometry is similar to that of SFR subassemblies, but the materials must be different.
- The UOX pellets were identified on the basis of earlier applications in SFRs operated in Russia (BN-350, BN-600) [109][110].
- The GFR design assumes operation at very high temperature (850 °C core outlet temperature) and high dose rates (22 dpa on SiC cladding) [4].
- Since the traditional LWR or SFR fuel cannot be applied, other materials were reviewed. The potential pellet materials were the oxides, carbides and nitrides [7].
- As an option, UOX pellets with ^{235}U enrichment about 20% can be considered. Oxides present an alternative option possibly allowing faster qualification of refractory cores.
- Among the cladding materials the SiC_f/SiC tubes are the first candidates[36].

Further actions:

SAFETY LIMITS

- During the ALLEGRO design process, the expected irradiation doses, temperatures, thermal and mechanical loads could be precised.

TRL 2: PRACTICAL APPLICATIONS SUGGESTED AND CONCEPTS FORMULATED

Objective: fuel and cladding designs were selected

Pre-existing knowledge:

MANUFACTURING

- Mature UOX pellet fabrication technology exists for LWR with typical enrichment up to 5% [102].
- Mature UOX pellet fabrication technology exists for SFR with high enrichment in Russia [109].
- Reprocessing technology of UOX fuel exist for LWRs [116].
- SiC_f/SiC tube production technologies have been developed [36].
    - Processing routes presently available for industrial production of SiC composites are chemical vapour infiltration (CVI), nano-infiltration and transient eutectic-phase process (NITE), melt infiltration (MI) or occasionally termed reaction sintering (RS) or liquid silicon infiltration (LSI) and polymer-impregnation and pyrolysis (PIP) [36][40][41][42].

MODELS

- Mature numerical models exist for UOX pellets in LWR conditions.
- Basic material properties for modelling purposes are available for SiC_f/SiC cladding (mechanical properties, thermal properties) [45].

Further actions:

MANUFACTURING

- Optimal technology must be selected for production of long SiC_f/SiC tubes and for their sealing.
- Optimal technology must be selected for production of oxide pellets with high enrichment.
- Reprocessing technology for SiC_f/SiC tubes and long term storage solutions must be developed.

MODELS

- Computational analyses (fuel behaviour simulations) are needed for ALLEGRO conditions and scenarios to identify the typical parameters and material property ranges for the oxide pellets and SiC_f/SiC cladding in order to support the experimental programmes with establishing test matrices.
- Careful application of LWR models for GFR conditions is needed taking into account the limitations of applied correlations and material properties.
- Further extension of computer codes with SiC_f/SiC cladding and UOX pellet properties, correlations and models is needed.
- Data and models for chemical interactions between UOX and SiC are needed.
- Data and models on high temperature fission product retention in SiC_f/SiC are needed

TRL 3: BASIC COMPONENTS FABRICATED AND SUCCESSFULLY DEMONSTRATED

Objective: fuel and cladding was successfully tested in reactor

Pre-existing knowledge:

MANUFACTURING

- SiC_f/SiC cladding tubes were produced in several laboratories:
    - Since the monolithic SiC has low fracture toughness, composite structures were introduced with using strong SiC fibres that reinforce a SiC matrix to form a SiC_f/SiC composite [36][52][53].
    - Triplex tube samples, monolith-only samples, and SiC_f/SiC bonding samples were fabricated in the USA [54].
    - Nuclear-grade SiC components were manufactured at the Kyoto University [55].
    - KAERI fabricated nuclear grade SiC_f/SiC duplex and triplex cladding tubes [41].
    - Sandwich technology was developed in France to produce leaktight structure [56][57].
- UOX for fast reactor with higher ^{235}U content is routinely produced in Russia.
    - The BN-350 operated with uranium in the range of 20% enrichment [110]. The BN-600 used fuel assemblies had three different ^{235}U enrichment levels between 17% and 33% [109].
    - The first core of Russia's new BN-800 reactor contains some HEU fuel but the plan is to transition it to 100% plutonium fuel. Russia also has provided initial HEU fuel for China's Experimental Fast (breeder) Reactor [111].

MODELS

- UOX data for modelling purposes are available in open publications [102][103].
- On the base of analysis of experimental observations and critical assessment of existing models for oxide fuel structure evolution under operation conditions of fast reactors, numerical models were developed [99][100][101].

EXPERIMENTS

- Post-test examination of several irradiated UOX fuel from SFRs was performed [97][98].
- The operability of UOX fuel for SFR conditions was tested in out-of-pile facilities and in reactors [98][104][105].
- SiC_f/SiC cladding developments are in progress in many countries.
    - The SiC_f/SiC cladding applicability is under investigation for different reactor types and is a candidate for accident tolerant fuel material in LWRs [36].
    - French SiC_f/SiC cladding samples irradiated in BOR-60 [53].
    - A number of SiC/SiC samples were exposed to PWR coolant and neutronic conditions using an in-core loop in the MIT research reactor (MITR-II)[54].
    - Testing of SiC_f/SiC cladding in high temperature He was carried out in Hungary, focusing on the effect of impurities. The applicability of SiC_f/SiC cladding in high temperature He was confirmed [75].
    - The SiC/SiC composites were investigated as structures and flow channel insert (FCI) for fusion reactor blankets, control rod sheath in advanced gas-cooled thermal reactors, core components in gas-cooled fast reactors (GFR), and fuel cladding for various fission reactors, including the light water reactor (LWR) [43].

- Initially developed as fuel cladding materials for the Fourth generation Gas cooled Fast Reactor (GFR), this material has been recently envisaged by CEA for different core structures of Sodium Fast Reactor (SFR) which combines fast neutrons and high temperature (500°C) [53].
- The Advanced High-Temperature Reactor (AHTR) is a new reactor concept that uses a liquid fluoride salt coolant and a solid high-temperature fuel. Several alternative fuel types are being considered for this reactor. One set of fuel options is the use of pin-type fuel assemblies with silicon carbide (SiC) cladding [40].
- Silicon carbide (SiC) has been investigated for use in both fission and fusion applications and recently has been considered as cladding material for advanced light water reactors (ALWR) working with accident tolerant fuel (ATF) [10][12][41] [76][77][78][79][80].
- SiC_f/SiC type duplex and triplex type claddings were produced in KAERI for nuclear fuel. This cladding type is a candidate material for the refractory core of the ALLEGRO reactor. High temperature testing in He atmosphere with different impurities, detailed scanning electron microscope analyses of some cladding samples and mechanical testing of all samples were carried out at MTA EK [75][81].
- SiC material is used in gas-cooled high temperature pebble bed reactors as one layer in the TRISO fuel [82]. The TRISO coatings were applied at ORNL. Nominal coating thicknesses were 100 µm for the porous carbon buffer, 40 µm for the inner pyrolytic carbon (IPyC) layer, 35 µm for the SiC layer,and 40 µm for the outer pyrolytic carbon (OPyC) layer.

- The applicability of SiC_f/SiC cladding in gas cooled reactor was addressed in several experimental programmes in the past:
  - SiC_f/SiC cladding tubes are produced at KAERI and their behaviour is tested in the framework of extensive experimental series [83]. The work performed at KAERI with SiC_f/SiC composites for nuclear applications includes the development of light water reactor (LWR) fuel cladding and in-core components for very high temperature reactors (VHTR). One series of KAERI tests focused on the investigation of behaviour CVD (Chemical Vapour. Deposition) SiC and SiC_f/SiC composite in the oxygen containing He and air. In air atmosphere positive mass gains were observed above 1100 °C. It was concluded by KAERI experts that long-term experiments and tests at higher temperatures are required to verify the chemical compatibility of SiC_f/SiC composites with the VHTR/Fusion relevant He coolant chemistry [83].
  - The chemical compatibility aspects of CVD β-SiC and SiC_f/SiC composites with a VHTR specific helium coolant were examined at KAERI in another test series [84]. The specimens were exposed to helium gas containing 20 Pa H2, 5 Pa CO, 2 Pa CH4, and 0.02–0.1 Pa H2O, which is an expected VHTR coolant chemistry. Oxidation tests were carried out at 900 °C and 950 °C for up to 250 hours. β-SiC and SiC_f/SiC composites had an excellent compatibility with the expected VHTR helium coolant environment. The oxidation of β-SiC as a matrix material of the SiC_f/SiC composite reacted in a passive oxidation regime owing to the presence of water vapour. A condensed version of the oxide SiO2 formed at an early stage of oxidation and the growth of this oxide layer was very limited as the oxidation time increased up to 250 h. The recession of the pyrolytic carbon interphase of SiC_f/SiC composite was not observed [84].

- High temperature (1300–2000 K) tests on massive SiC samples (sintered and CVD) were performed in France. The tests were coupled to SEM (Scanning Electron Microscopy), XPS (X-Ray Photoelectron Spectroscopy) and XRD (X-Ray Diffractometry) analyses before and after oxidation. It was found that the level of oxidizing species had an important impact on the physico-chemical behaviour of SiC. The investigated SiC samples maintained their structural integrity at high temperature in helium environment with low oxygen partial pressure [85].
- High-temperature tests of silicon carbide composite cladding under GFR (Gas Cooled Fast Reactor) conditions were performed at KIT (Karlsruhe Institute of Technology) [86]. In particular, the feasibility of silicon carbide composites is investigated in helium with low amount of impurities (H_2, CO, N_2, O_2, H_2O, CH_4 and CO_2) by means of a thermogravimetric device. The SiC_f/SiC composites specimens were provided by CEA. The temperatures of the tests were in the range of normal operation conditions (900–1000 °C) and accident conditions (up to 1500 °C) of a gas fast reactor. Passive oxidation was detected at 900 °C and 1200 °C, whereas the samples underwent active oxidation and mass loss at 1300 °C, 1400 °C, and 1500 °C. Overall, the results meet the requirements for the aimed application, since the transition temperature from passive to active oxidation is 300 °C higher than the nominal working conditions of GFR.
- Duplex and triplex type SiC_f/SiC cladding samples with 9.9 mm length were produced in KAERI laboratories and provided to MTA EK in the framework of scientific collaboration [75]. The cladding tubes were tested at 1000 °C for 7 h in helium atmosphere with and without gas impurities. The mass gain measurements showed that in case of pure helium atmosphere and with hydrogen or nitrogen impurities small mass reduction was observed. In case of methane impurities the decomposition of methane and the formation of carbon deposits lead to mass gain of the SiC samples

Further actions:

MANUFACTURING

- Companies for SiC_f/SiC cladding tubes fabrication should be identified and the details of technology (monolith or fiber, sandwich structure, duplex, triplex, fiber winding patterns and angles, additives) should be fixed.
- Companies for oxide pellet fabrication should be identified and the details of technology should be fixed.

MODELS

- Numerical models are needed to carry out simulation of fuel behaviour in steady state and transient conditions.
- Numerical models should be validated against completed reactor tests with oxide pellets including the burnup effects, wide power ranges, transients.
- The planning of new measurements should be supported by computer code calculations, and post-test calculations should be performed.
- The capabilities of severe accident codes should be extended to cover the behaviour of fuel elements with UOX pellets and SiC_f/SiC cladding in DEC conditions.

EXPERIMENTS

- The oxide pellets produced with the selected technology must be tested out-of-pile and in-pile.
    - Basic material properties of oxide pellets have to be measured.
    - For oxide pellet irradiation tests research reactor capabilities (see irradiation facilities in chapter 5) should be reviewed
    - Irradiation programmes must be defined. Structural changes due to irradiation must be identified. Burnup dependent material properties should be measured.
    - Both on-line measured data and post-irradation examination (PIE) results must be evaluated.
- The SiC_f/SiC cladding produced with the selected technology must be tested out-of-pile and in-pile.
    - Basic material properties of SiC_f/SiC cladding have to be measured.
    - The irradiation behaviour of SiC_f/SiC cladding needs further examinations (e.g. those samples that were irradiated in the BOR-60 reactor recently).
    - The potential irradiation damage should be identified for different SiC structures.
    - The effect of different components (e.g. BN) in SiC_f/SiC cladding should be evaluated.
- Compatibility of SiC_f/SiC cladding and oxide pellets should be proven at high temperature to cover DEC conditions.

TRL 4: Integration of components into a basic system

Objective: fuel rod was fabricated and tested

Pre-existing knowledge:

MANUFACTURING

- Blind-end SiC cladding tube closing technology and buffer bond of high porosity C-based braid were patented by CEA [56][57].
- UOX with SiC_f/SiC cladding is under development for LWR applications [106].

SAFETY LIMITS

- A recent OECD NEA opinion paper pointed out that new performance metrics and regulatory criteria to preserve coolable geometry need to be developed for SiC_f/SiC cladding [87].
- Rod-like fuel geometry was selected considering the planned ALLEGRO core design [88][89][90][91].

MODELS

- The development of fuel behaviour codes for the simulation of UOX SiC_f/SiC fuel in LWRs is in progress[107][108].

EXPERIMENTS

- Recent results of the development of ATF fuel with SiC_f/SiC can be used[106].
- A US report [45] summarized various physical, mechanical, and chemical compatibility properties of SiC_f/SiC composites for LWR cladding applications.

Further actions:

MANUFACTURING

- The technology of refractory fuel fabrication must be developed. Production of fuel rod with oxide pellets and SiC_f/SiC cladding has to be demonstrated in the available workshops/laboratories.
- Key dimensions and tolerance of fuel components have to be specified.
- Key constituents have to be specified with allowance for impurities.
- Microstructure attributes for materials within fuel component have to be specified for otherwise justified.
- Short fuel rodlets with SiC_f/SiC cladding tubes have to be produced for in-pile experimental purposes.
- SiC tube closing has to be solved for the selected cladding type.
- The need for internal spring in the fuel rod to fix the fuel column should be defined.

SAFETY LIMITS

- The main design parameters (fuel and cladding geometry, enrichment, gas volume, gap size) of ALLEGRO fuel rod have to be fixed.
- Fuel performance envelope has to be defined.
- Fuel criteria for ceramic ALLEGRO fuel rods have to be defined.
    - Radionuclide retention requirements have to be specified

  - Criteria for barrier degradation and failure have to be specified.
  - Criteria have to be specified for ensuring coolable geometry.
  - Criteria has to be provided to ensure control element insertion path is not obstructed

MODELS

- Ceramic fuel rod models have to be developed with appropriate modelling capabilities on
  - Geometry,
  - Materials and
  - Physics.
- Numerical models should be applied to carry out simulation of fuel behaviour in steady state and transient conditions in the ALLEGRO reactor. The calculations should include reactor physics, thermal hydraulics and fuel behaviour aspects.
- The codes should be validated against experimental data:
  - The data used for assessment has to be appropriate.
  - The evaluation model has to demonstrate the ability to predict fuel failure and degradation mechanism over the test envelope.
  - Evaluation model error should be quantified through assessment against experimental data
  - Evaluation model error should be determined through the fuel performance envelope.
  - Sparse data regions have to be justified.
  - Evaluation model has to be restricted to use within its test envelope.

EXPERIMENTS

- Failure mechanisms (loss of cladding integrity) must be investigated in wide range of parameters.
- Test series are needed to support criteria development.
- Assessment data must be independent of data used to develop/train the evaluation models.
- Data has to be collected over a test envelope that covers the fuel performance envelope.
- Experimental data have to be accurately measured.
  - The test facility must have an appropriate quality assurance program.
  - Experimental data must be collected using established measurement techniques.
  - Experimental data must account for sources of experimental uncertainty.
- Test specimens have to be representative of prototypical fuel.
  - Test specimens should be fabricated consistent with the prototypical fuel manufacturing specification.
  - Distortions should be justified and accounted for in the experimental data.

TRL 5: BASIC SYSTEM SUCCESSFULLY DEMONSTRATED

Objective: fuel rod was successfully tested in reactor

Pre-existing knowledge:

MANUFACTURING

- None

SAFETY LIMITS

- None.

MODELS

- None.

EXPERIMENTS

- None.

Further actions:

MANUFACTURING

- Improvement of fuel rod production technology on the basis of learnings from tests in TRL 4.

SAFETY LIMITS

- Update of safety limits may be needed on the basis of learnings from tests in TRL 4.

MODELS

- Conservative model has to be developed to simulate radionuclide retention and release from fuel matrix.
- Conservative model has to be developed to simulate barrier degradation and failure.
- Conservative model has to be developed to predict loss of coolable geometry.
- Numerical models should be used to support the preparation of in reactor tests. Wide range of boundary conditions (e.g. linear heat rates, inlet temperatures) must be covered. Special attention should be devoted to start-up and power changes.
- Post-test calculations of performed experiments must be carried out. Further model developments may be necessary.

EXPERIMENTS

- Helium irradiation loop or capsule has to be constructed in the selected fast reactor.
- Irradiation of test rods in should be carried out to reach the target dpa/burnup.
- Experimental data have to be accurately measured.
    - The test facility must have an appropriate quality assurance program.
    - Experimental data must be collected using established measurement techniques.
    - Experimental data must account for sources of experimental uncertainty.
- Test specimens have to be representative of prototypical fuel.

  - Test specimens should be fabricated consistent with the prototypical fuel manufacturing specification.
  - Distortions should be justified and accounted for in the experimental data.
- The results of irradiation programme have to be evaluated and the operability of fuel rod has to be demonstrated.
- Transient testing for operational transients, incidents and accidents will be needed.

TRL 6: PROTOTYPE CONSTRUCTION

Objective: fuel assembly was fabricated and tested

Pre-existing knowledge:

MANUFACTURING

- None

SAFETY LIMITS

- The main elements of the fuel assembly design are the hermetically closed cladding tubes, cylindrical pellet column, large free volume inside of the fuel element for gases and the use of fuel assemblies to guarantee the fixed geometrical arrangement and simple handling [7][36][39][41].
- The bundle of pins will be housed in a hexagonal wrapper tube (shroud) made of SiC. The distance between fuel pins should be kept by the application of SiC spacer grids [90][91].
- In the earlier ceramic core design of the ALLEGRO reactor, there are 87 ceramic fuel assemblies. Each assembly contained 90 fuel rods. The active length of the rods was 0.86 m [4][88][89][90][91].

MODELS

- Detailed thermal hydraulic modelling of the ALLEGRO ceramic assembly was performed in Hungary [91].

EXPERIMENTS

- The applicability of SiC type BWR channel box is under investigation. SiC-SiC composites are being considered for applications in the core components, including BWR channel box and fuel rod cladding, of light water reactors to improve accident tolerance. In the extreme nuclear reactor environment, core components like the BWR channel box will be exposed to neutron damage and a corrosive environment. To ensure reliable and safe operation of a SiC channel box, it is important to assess its deformation behaviour under in-reactor conditions including the expected neutron flux and temperature distributions [96].

Further actions:

MANUFACTURING

- Key dimensions and tolerance of fuel assembly components have to be specified.
- Key constituents have to be specified with allowance for impurities.
- Microstructure attributes for materials within fuel assembly component have to be specified for otherwise justified.
- The fuel assembly fabrication process must be demonstrated, new workshops/laboratories are needed.
- ALLEGRO refractory subassemblies have to be produced.

SAFETY LIMITS

- Detailed design of the ALLEGRO fuel assembly has to be performed and the main dimensions have to be specified (fuel rod active and total length, diameters, internal pressure, pitch size, number of pins in the assembly, spacer grid geometry and positions).
- Criteria for subassemblies must be specified for normal operational conditions (e.g. allowable change of dimensions).

- Fuel assembly specific failure mechanisms (e.g. fretting, bowing) should be identified, if such phenomena exist.
- Criteria has to be provided to ensure control element insertion path is not obstructed.
- Limitations on mechanical damage (due to e.g. earthquake) should be investigated.
- Subcriticality, coolability and shielding conditions must be specified for transport and storage conditions (e.g. in water pool).

MODELS

- Validation against fuel assembly experimental data.

EXPERIMENTS

- Out-of-pile testing should be carried out.

TRL 7: Prototype successfully demonstrated

Objective: fuel assembly successfully tested in reactor

Pre-existing knowledge:

MANUFACTURING

- None.

SAFETY LIMITS

- None.

MODELS

- None.

EXPERIMENTS

- None.

Further actions:

MANUFACTURING

- Improvement of fuel assembly production technology on the basis of learnings from tests in TRL 6.

SAFETY LIMITS

- Definition of ALLLEGRO start-up procedure and normal operational conditions (not only for fuel, but for all systems).

MODELS

- Conservative model has to be developed to confirm that the control element insertion is not obstructed.
- Conservative model has to be developed on the prediction of loss of coolable geometry.
- Supporting calculations of reactor start-up (including not only fuel behaviour, but reactor physics, thermal hydraulics).
- Checking of fuel safety criteria for ALLEGRO scenarios (the corresponding scenarios, covering cases must be defined).

EXPERIMENTS

- Refractory subassemblies have to be successfully irradiated in the gas loop of a fast reactor (e.g. MBIR) with in-core monitoring or in prototype reactor (ALLEGRO)
- PIE of irradiated subassemblies have to be carried out (with both non-destructive and destructive procedures).
- The state of the assembly must be checked for meeting the fuel assembly criteria.

TRL 8: Actual system constructed and commissioned

Objective: fuel assembly was fabricated for ALLEGRO and irradiation started

Pre-existing knowledge:

MANUFACTURING

- None.

EXPERIMENTS

- None.

Further actions:

MANUFACTURING

- Subassemblies for testing in the first core of ALLEGRO have to be produced in reload quantities.

EXPERIMENTS

- Start of irradiation of fuel in the ALLEGRO core.
- Construction of ALLEGRO core (after inactive testing of all ALLEGRO technological components).
- Start-up of the reactor core using detailed on-line measurements for core-monitoring.
- Regular or on-line activity concentration measurements in the primary coolant.
- Shut-down of the reactor for refuelling.
- Appropriate hot cell capabilities have to be established.
- PIE of irradiated fuel assemblies.

TRL 9: Successful operation of actual system

Objective: fuel assembly successfully irradiated in ALLEGRO reactor

Pre-existing knowledge:

EXPERIMENTS

- None.

Further actions:

EXPERIMENTS

- Continuation of irradiation of refractory subassemblies in ALLEGRO.
- Fuel assemblies have to perform successfully under irradiation in reload quantities (demonstrated by surveillance programme).
- Regular PIE of irradiated fuel assemblies.

TRL 10: WIDESPREAD, RELIABLE AND LONG-TERM OPERATION OF MANY ACTUAL SYSTEMS

Objective: long term successful use of fuel

Pre-existing knowledge:

EXPERIMENTS

- None.

Further actions:

EXPERIMENTS

- This step can be reached only with the launching of reactor fleet with several GFR reactors.

Summary on UOX fuel with SiC_f/SiC cladding qualification process