THE GENERATION IV PROJECT "GFR FUEL AND OTHER CORE MATERIALS"

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I. INTRODUCTION

The GFR combines the advantages of a fast neutron spectrum with those of high temperatures. [1] It can be deployed for closed fuel cycles for the minimisation of wastes, when the minor actinides are recycled. Furthermore, the effective utilisation of uranium resources is increased dramatically compared with today's light water reactor (LWR) fleets. For sustainability, self generation of Pu in the core is required, and can be provided by the fast neutron spectrum, a high power density (ca 100 $MW.m^{-3}$) and dense fuels. Proliferation risk is minimised through the use of a core without blankets. The high outlet temperatures potentially provide improved economy in the power conversion units and also permit process heat applications, just as for its thermal counterpart the very high temperature reactor (VHTR). Indeed, it is possible to gain synergy with VHTR projects, as many system components outside the core will be common to both reactor types.

Inside the core, however, there will be little similarity between the VHTR and GFR. The fuel concept for the VHTR requires coated particle fuel, which is distributed in a graphite matrix. The consequent low power density, and thermalised neutron spectrum would be inconsistent with the needs of the GFR. Initially coated particle designs (larger kernels, thinner layers) were considered for the GFR, but are no longer pursued.

Given the high core outlet temperatures exceeding 800°C, it is clear that conventional metallic core structures and fuel cladding will be unable to meet the demanding requirements, under both normal and off-normal operating conditions. Thus, only refractory metals or ceramic components can be considered for these purposes, as temperature excursions above 1 600°C have to be foreseen. Core preservation during severe accidents is a necessity, and safety considerations require that there is a limited fission gas release during transients.

The road to the first GFR demonstration reactor foresees the implementation of a test reactor, ALLEGRO, with a power of about 50 MW. For this reactor two cores are foreseen

The startup core will operate at lower outlet temperatures than planned for the GFR, and will incorporate systems to ensure its safe operation, while utilising steel cladding materials and structures. Standard MOX fuel is also considered. This initial core will also be fitted with experimental locations where ceramic fuel (plate or pin) sub assemblies can be inserted and tested.

The second core will then be fully ceramic with an advanced fuel, possibly mixed metal carbide (MC).

The design and development of an innovative refractory fuel in an advanced ceramic cladding remains a fundamental goal of the GFR system. Today, the focus is on SiC composite as structural and cladding material, with carbide (MC) fuel taking first priority over both oxide and nitride fuels as backups.

II. FUEL AND FUEL ELEMENT DESIGN OPTIONS

The classical fast reactor fuel concept consists of a fuel pin into which fuel pellets are loaded. Key design parameters are the pellet cladding gap (necessary to allow fuel swelling), and its concomitant rather poor thermal conductivity due to the helium gas bond. A plenum is available and dimensioned to collect the fission gas (and helium if minor actinide (MA) bearing fuels should be deployed). It is a goal of this Generation IV project to test new and radically innovative concepts, such as the plate type, where the fuel is in the form of a disk (low height to diameter ratio). Initially monolithic SiC was considered as a cladding material, but was dismissed in favour of composite materials such as fibre reinforced SiC, denoted SiC-SiC$_f$, which exhibits superior mechanical properties. Substantial efforts are needed not only in the design of the fuel and its cladding, but also in the full fuel sub assembly, necessitating a multitude of studies on various fuel / cladding configurations.

The situation is in fact far more complex, as compatibility tests have shown that SiC and mixed metal carbide fuels (MC) react, neces–sitating the introduction of a protective liner made of W or other refractory material. This W liner actually has a dual role, namely to inhibit fuel clad interaction, and to act as a fission product barrier. Thus it must be sealed at the fabrication stage. The surrounding SiC-SiC$_f$ cladding then provides the mechanical support.

II.1 Plate Type Fuel

The basic design of plate type fuel is based on two ceramic plates, which enclose a honeycomb structure containing cylindrical pellets made of the mixed carbide fuel. This design is shown schematically in Figure 1. The individual plates can then be stacked in a fuel assembly as shown in Figure 2. [2]

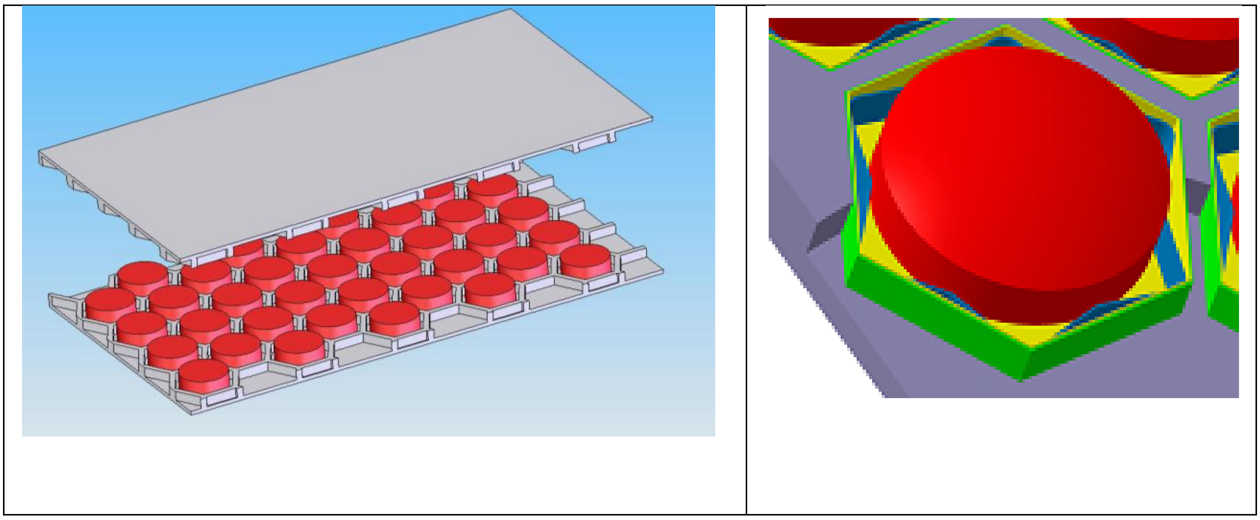

Figure 1: GFR plate fuel design

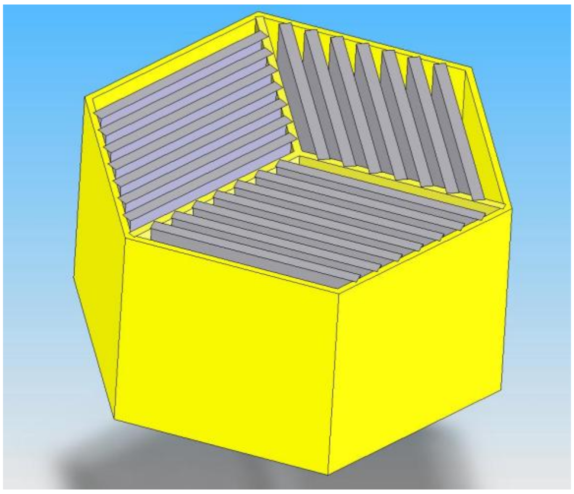

Figure 2: Plate fuel sub assembly concept

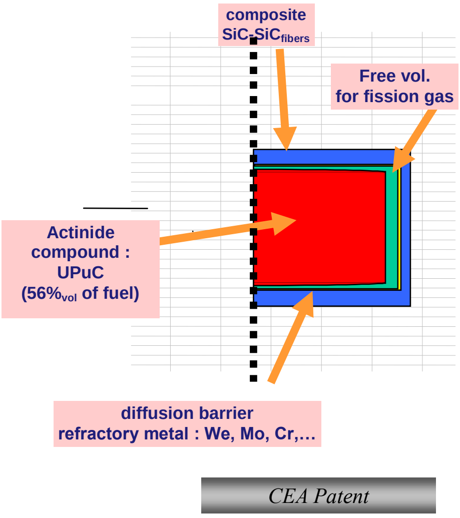

Figure 3: Cell dimensions in plate type fuel

The design of the plate fuel concept has been determined in a set of detailed thermo-mechanical assessments made at the CEA. Parameters investigated include the cell dimensions, fuel disk geometry, radial and axial gaps, and free volume along with irradiation behaviour laws derived for (U,Pu)C pellets and SiC-SiC_f cladding (see Table 1 and Figure 3).

A particularly important design parameter is the closed axial gap between the pellets and SiC-SiC_f cladding at begin of life (BOL), which provides a mechanical bond decreasing the thermal barrier and consequently the fuel operating temperature. The calculated thermal distribution at BOL (1/24 cell) is shown in Figure 4. The hottest part of the fuel is at about 1500 K and remains constant throughout the irradiation. The temperature gradient in the cladding material is higher than in the fuel itself. This is a consequence of the poor thermal conductivity of the SiC-SiC_f cladding, and clearly, a solution to ameliorate this situation is required. The high temperature gradient in the cladding also results in inhomogeneous thermal expansion, and combines with pellet-clad interaction, to give a compressive stress in the cladding of 300 MPa at the so called P12 point. [2]

Table 1: Plate Fuel Cell Parameters

Fuel plate thickness	ep =8.4 mm
Across flat cell	a = 14 mm
Cladding thickness	eg = 0.85 mm
Pellet-clad axial gap	ja = 0.05 mm
Pellet-clad radial gap	jr = 0.75 mm
Wall thick in honey comb	ev = 1.115 mm
Pellet diameter	D = 11.285 mm
Pellet height	hc = 6.5 mm
Fissile vol by cell	23%

The much smaller thermal gradient in the fuel limits thermally induced stresses. It is assumed that fuel swelling will be accommodated by creep, which due to the operating pressure on the plate (70 bar) should occur in the radial direction and not cause any additional axial stress on the cladding. The thermomechanical calculations are encouraging, but it is clear that further evaluations are needed, and in particular both integral and separate effect irradiation tests are required to prove the feasibility of the concept. Fabrication of such a complex structure with individually sealed alveoli will be a key issue to be mastered.

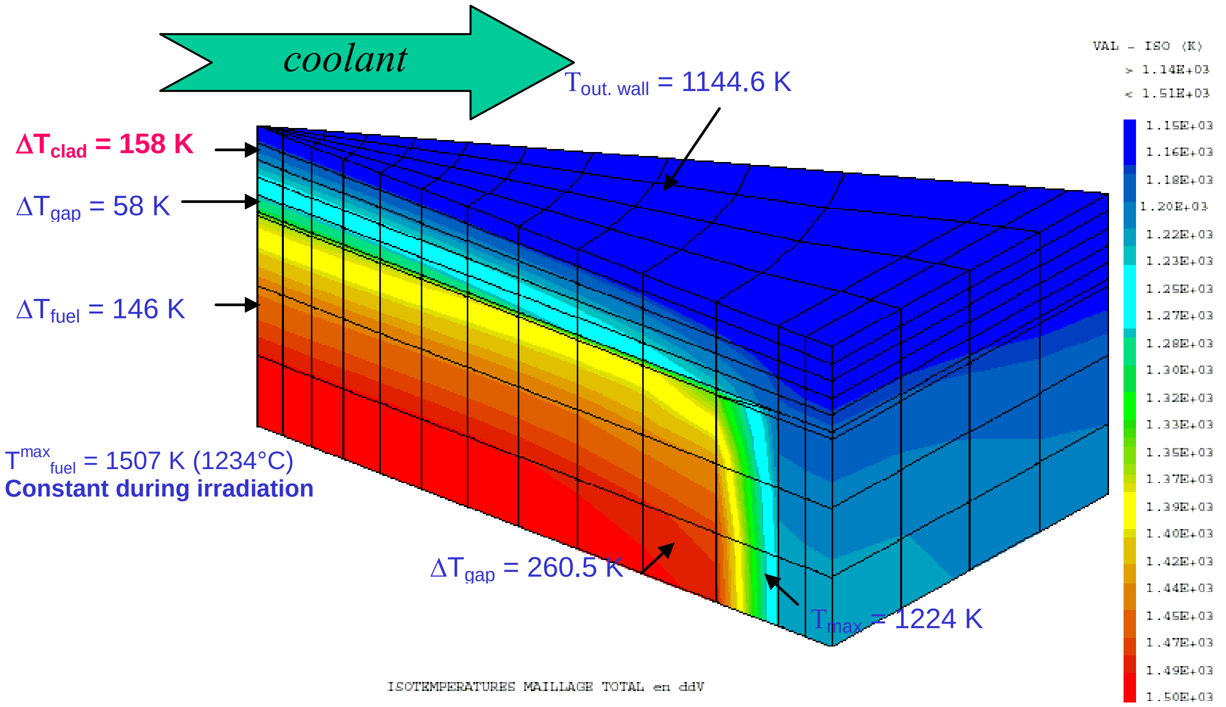

Figure 4: Thermal behaviour of a plate type fuel cell at BOL. Temperature gradients in the cladding, axial and radial gaps, and fuel are shown

II.2 Pin type fuel

Pin type fuel will be similar to current fast reactor designs. A single rod should be formed from two individual components, with independent seals and gas plenums. This improves mechanical stability, and it is believed that manufacturing of short straight tubes will be simpler than a single longer component. Again carbide fuel (with oxide and nitride as back up options) and SiC-SiC$_f$ cladding are considered. Irradiation induced carbide fuel swelling (twice that of the oxide) can cause fuel clad mechanical interaction (FCMI) at relatively low burnups. Slender pins, certainly not greater in inner diameter than 5 mm, are required and the pellet cladding gap should be chosen carefully to allow for fuel swelling and deleterious mechanical interaction. Spherepac fuel would also be advantageous, as it provides a convenient means to accommodate swelling, while limiting stresses on the cladding. Pellet fuel at 85% of the theoretical density can also accommodate swelling internally within its porous structure. A high thermal stability is required, however, to avoid sintering at BOL, which would then result in higher fuel temperatures due to the concomitant increased gap size. The studies on pin type fuel are still at an early stage. Though there are difficulties to master *vis-à-vis* fuel swelling and concomitant FCMI that could ensue, manufacturing of the components and sealing them should be simpler than for plate type fuel.

III. MATERIALS

III.1 Fuel

For neutronic and safety design studies, carbide fuel has advantages over oxide fuel. It has a high fissile element density, and also higher thermal conductivity. In principle, nitride fuel has similar properties, but has the disadvantage of requiring ^{15}N enrichment, to avoid ^{14}C produc–tion during irradiation. Past experience has shown that its fabrication may be somewhat more convenient than carbide, and consequently, it is considered as a reserve option along with the oxide. Some properties of various fuel compositions are summarised in Table 2. A drawback of carbide fuel is its volatility, in particular if MA bearing fuels are considered. The EURATOM FP6 programme GCFR [3] has produced a number of review reports covering fabrication properties, past irradiation programmes and reprocessing of MC and MN fuels. The CEA, ITU and PSI have been actively engaged in the re-establishing fabrication facilities for such advanced fuels. Mainly carbothermal reduction of the oxides is considered for the production of nitride and carbide fuels. It is proposed that actinide losses during production (due to the high vapour pressure of Pu) can be overcome, if a solid solution of (U, Pu)O_2 is used as the starting material. As the solid solution is already formed, the losses should be lower as PuC should not be formed, rather (U, Pu)C directly. In addition, the length of the high temperature processing in the carbothermal reduction step will be shorter. Plans are also afoot at PSI to use such a process for particle production, enabling Spherepac deployment.

Table 2: Fuel Properties

	Carbide	Nitride	Oxide
Theoretical density ($g.cm^{-3}$)	13.58	14.32	11.5
Melting point (°C)	2420	2780	2750
Thermal Conductivity ($W.m^{-1}K^{-1}$)	16.5	14.3	2.9

III.2 SiC-SiC$_f$ cladding

Much of the pioneering work on advanced SiC-SiC$_f$ has been made at Kyoto University. This composite material has key advantages in terms of toughness (K_{IC} up to 30 $MPa.m^{1/2}$). Its permeability could necessitate an outer liner to prevent damage of the inner liner by the helium at 70 bar in the primary loop of the reactor. There also remain some concerns about the stability of SiC at very high temperatures that could be encountered during a severe temperature excursion in off normal operating conditions

Several irradiation experiments on monolithic SiC were performed in the UK in the 1960s. The results and experience were gathered together in reports produced within the EURATOM FP6 programme GCFR. [1] For other reactor structural components, *e.g.* the reactor pressure vessel and reflector, synergy is sought with the Generation IV VHTR project on structural materials.

IV. IRRADIATION TESTING PROGRAMMES

Ultimately, all fuel and structural materials need to be tested and qualified in dedicated irradiation experiments. Carbide fuels were tested in various programmes in the past, but their behaviour is not nearly as well known as for oxide fuel. When MAs are present, there is no information at all. Ongoing and planned irradiation programmes are depicted in Figure 5. [4] Post Irradiation Examination on the NIMPHE programmes is in progress at JRC-ITU. These tests showed a relatively good behaviour of both carbide and nitride fuel at intermediate burnups. Recently, the results of an irradiation test on spherepac fuel made in the Fast Flux Test Facility (FFTF) [5] showed excellent behaviour of carbide fuel in this form.

New irradiation tests on carbide and nitride fuel compatibility with SiC-SiC$_f$ are nearing completion in Phenix (FUTURIX CONCEPT), while a new programme in BR2 (IRRDEMO) will be launched shortly. Material tests are ongoing in Phenix and OSIRIS (FUTURIX MI and REA series, respectively).

V. THE FUEL AND OTHER CORE MATERIALS PROJECT

The FCM project arrangement will be signed by France, Switzerland, Japan and EURATOM. The programme consists of the following work packages:

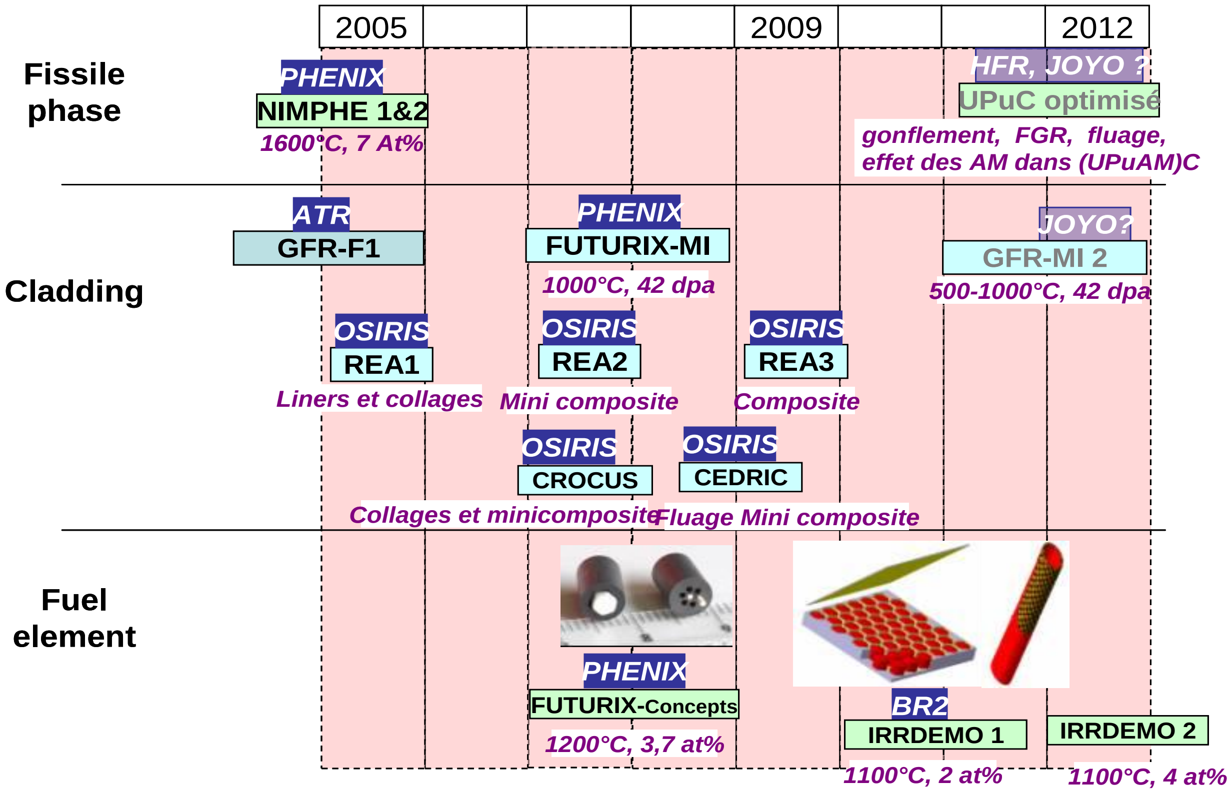

Figure 5: Ongoing and planned irradiation experiments on GFR fuel and materials

1. **Fuels and assemblies modelling and design**, which covers studies on fuels in plate and pin geometries, using fuel performance codes (PLEIADES-CELAENO, TRANSURANUS) and finite element methods. Thermo aerodynamic codes are used for assembly design. Fuels under consideration include MC, MN and MOX, while cladding and structural materials include SiC-SiC$_f$ and oxide dispersion strengthened steel (ODS), the latter being an option for the ALLEGRO start up core.

2. **Basic fuel material studies**, covers irradiation studies on inert materials in Phenix, detailed investigations on fresh and irradiated MC and MN fuels, and interactions between actinide fuels, fission products and inert cladding material.

3. **Basic in core material studies** is dedicated to the structure materials for the core (subassembly, control rods, guide tubes, and reflector) and in particular their ability to satisfy main safety requirements of the reactor in various operating conditions. Out of pile and in pile studies are foreseen, along with the development of appropriate codes and standards. Where possible synergies with the Generation IV VHTR Materials project are sought.

4. **Fuel fabrication process development** covers the comparison and selection of processes suitable for MX fuels with and without minor actinides, testing of these processes with the development of relevant flowsheets, leading to the fabrication of U, U-Pu and U-Pu-MA specimens for property determination and eventually for irradiation testing.

5. **Fuel and assembly development and qualification by irradiation testing** addresses screening, optimisation and validation phases required for fuel deployment. Minor actinide bearing fuels are considered in a first instance in dedicated separate effect studies, possibly

in material testing reactors, before validation in fast reactor systems.

6. **Behavior during off-normal conditions** is concerned with the ability of the fuel and fuel element to retain FPs during a depressurisation accident. This activity covers thermal behaviour of SiC-SiC$_f$ in appropriate conditions, high temperature behaviour of fresh MX fuels, chemical compatibility of all core materials at high temperature, and the tightness of the fuel element at temperatures beyond 1 600°C (corresponding to severe accidents).

The GFR FCM project will provide design data for ALLEGRO and GFR cores. It relies heavily on the French national programme, while Japanese programmes contribute mainly to SiC-SiC$_f$ investigations, and Switzerland and EURATOM contribute mostly on fuel issues.

VI. CONCLUSION

The GFR fuel and other core materials (FCM) project arrangement awaits signature, but even without formal signature the partners have collaborated effectively over the past 5 years. The project programme has now been defined until 2012. At that time decisions on future systems will be made in France and also at the European level in the framework of the Sustainable Nuclear Energy Technology Platform (SNE TP). [6]

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