Fine 3D neutronic characterization of a gas-cooled fast reactor based on plate-type sub-assemblies

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Abstract

CEA neutronic studies have allowed the definition of a first 2400MWth reference gas-cooled fast reactor core using plate-type sub-assemblies, for which the main neutronic characteristics were calculated by the so-called ERANOS "design calculation scheme" relying on several method approximations. The last stage has consisted in a new refine characterization, using the reference calculation scheme, in order to confirm the impact of the approximations of the design route. A first core lay-out taking into account control rods was proposed and the reactivity penalty due to the control rod introduction in this hexagonal core lay-out was quantified. A new adjusted core was defined with an increase of the plutonium content. This leads to a significant decrease of the breeding gain which needs to be recovered in future design evolutions in order to achieve the self breeding goal. Finally, the safety criteria associated to the control rods were calculated with a first estimation of the uncertainties. All these criteria are respected, even if the safety analysis of GFR concepts and the determination of these uncertainties should be further studied and improved.

KEYWORDS: Gas-Cooled Fast Reactor, Neutronic Characterization, Numerical Validation, Control Rods Impact, Safety Criteria

1. Introduction

CEA neutronic studies in support of gas-cooled fast reactor core design have allowed to define a first 2400MWth reference core using plate-type sub-assemblies [1,2]. These neutronic studies are included in a CEA global design approach, based in particular on fuel and material design studies, thermo-hydraulic behaviour in standard and transient conditions, power plant scenarios and safety analysis. According to the Generation IV criteria, the reference concept is based on a self-breeder core, with carbide fuel and SiC inert matrix, and a power density of 100 MW/m^3.

The sub-assembly consists in an hexagonal wrapper containing the plate bundle; there are three CERCER plate sub-bundles arranged with 30 degrees of inclination in order to obtain an hexagon; internal devices come to enchase the plates (see Figure 1). The plates consist in a CERCER component, surrounded by two claddings.

After preliminary thermo-mechanical studies, an advanced design for the CERCER plate was proposed:

- the claddings (as the hexagonal wrapper) would consist of SiC based material,
- the fuel component (without claddings) is an advanced CERCER (U,Pu)C – (SiC + gaps); the SiC has the function of matrix; gaps will allow to accommodate the fissile

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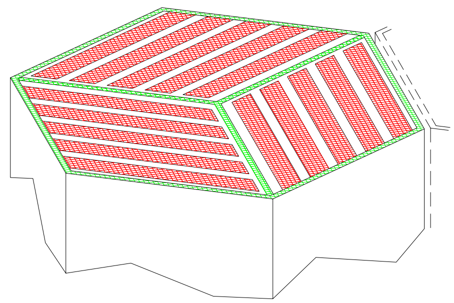
phase swelling and gaseous fission products release.

Figure 1: Fuel plate sub-assembly concept



During the preliminary phases, the main neutronic characteristics were calculated by the so-called ERANOS "design calculation scheme" relying on several method approximations (typically homogeneous cell calculations and RZ description of the core geometry) [3]. The last stage, described in this publication, has consisted in a new fine characterization, using reference calculation schemes, in order to confirm the first series of results and to quantify the impact of the control rods introduction in the core lay-out .The safety criteria were finally calculated.

2. Analysis of method approximations

2.1 The cell calculation scheme

The first CEA neutronic calculations were performed with an homogeneous fuel cell model, all the materials being associated to the mean fuel temperature.

The plate-type innovative concept of fuel sub-assembly has needed specific developments in the ERANOS code in order to take into account the exact description of the heterogeneous fuel cell (stripes of fuel and gas coolant, included in a wrapper tube). In this heterogeneous model, all the different materials are also calculated with their specific temperatures.

The effect on the reactivity of this fine description concerns two different impacts which have almost the same amplitude but opposite signs:

- a "geometrical improvement", due to the heterogeneous description but with all materials at the same temperature: impact on the core reactivity of –634 pcm
- a "temperature improvement", due to the calculation of the materials with their own temperatures: impact on the core reactivity of +562 pcm.

Finally, the global impact on the core reactivity of this fine core cell description is –72 pcm, coming from these two important opposite effects.

2.2 The core Hexagonal-Z geometry and the core calculation scheme

The "design calculation scheme" used for the first GFR design studies is based on a RZ core description with a flux calculation using the diffusion approximation in a finite difference scheme. The "reference calculation scheme", used for this study dedicated in particular to the quantification of the control rods effects and the safety criteria, requires a Hexagonal-Z description of the core and a flux calculation using the transport theory.

In a first step, we have defined an Hexagonal-Z core geometry without control rods, directly derived from the RZ core geometry, in order to quantify the impact of the "design" cylindrical model on the reactivity, the introduction of the control rods being studied in a further paragraph. The impact on the reactivity due to the “RZ => Hex-Z” transposition is equal to –241 pcm.

The reference flux calculation method in ERANOS for this type of hexagonal-Z geometry is based on a nodal method. In this first step, for which there is no control rod in the core lay-out, it is sufficient to use a simplified nodal calculation scheme with the following options:

- transport calculation simplified spherical harmonic,
- order of polynomial expansion for nodal source of 1,
- order of polynomial expansion for even flux of 6,
- order of polynomial expansion for partial current (leakage) of 0,

The impact on the core reactivity coming from the “Finite Difference Diffusion Scheme => 261 Transport Nodal Scheme” is equal to +455 pcm.

The geometry models were validated step by step by comparisons with the TRIPOLI Monte-Carlo code [4], using the same nuclear data set. This numerical validation was performed with the following geometries:

- Infinite fuel sub-assembly
- Axially finite fuel sub-assembly
- Bare Hexagonal-Z core consisting of 10 sub-assembly rows of a limited height of 2 meters
- Reflected Hexagonal-Z core consisting of 10 sub-assembly rows of a limited height of 1.3 meters. The radial and axial reflectors are made of Zr_3Si_2 material as for the reference core.

In all cases, the sub-assemblies are described in an heterogeneous way (exact description of the fuel plates in the wrapper tube). The discrepancies between the reactivities calculated by ERANOS and TRIPOLI4 are lower than 300 pcm.

2.3 The nuclear data

The first neutronic calculations were performed with the adjusted ERALIB1 nuclear data of the ERANOS neutronic code [5]. The ERALIB1 data set is based on a formal adjustment procedure of JEF2.2 cross-sections which takes into account a very wide range of integral data. In the available data base, different types of integral data are considered, such as critical masses, bucklings, spectral indices, response function data for neutron transmission… In total, 35 integral parameters from 71 different systems (thermal, epithermal, fast) have been used. It is important to note that this library was not specifically produced for the GFR concept studies. So, even if the integral data base was large, it will be necessary to confirm the validity of this library for this type of reactors. This is one of the goal, with the experimental qualification of calculation methods, of an experimental program proposed in MASURCA facility, called ENIGMA (Experimental Neutronic Investigation of Gas-Cooled Fast Reactor Configurations in MASURCA).

Considering the schedule of this experimental program, a new adjusted nuclear data set could be defined in the future on the basis of the recent JEFF3.1 nuclear data set. So, in a first approach, it was decided to use these JEFF3.1 data for the study related in this publication.

The impact on the reactivity due to the “ERALIB1 => JEFF3.1” data is equal to +219 pcm. This is a rather change but uncertainties associated to these results are a matter of concerns as it will be discussed further in the following.

2.4 The fission products description

The design calculation scheme of the ERANOS code is based on 6 JEF2.2-based lumped fission products, initially produced for the Na-Cooled Fast Reactors and representative of the absorption of the fission products of the mean heavy nuclides: ^{235}U, ^{238}U, ^{239}Pu, ^{240}Pu, ^{241}Pu and ^{242}Pu. The validity of these data for GFR applications was analysed during this study by comparing the reactivity swing obtained with these lumped fission products or calculated with 88 explicit fission products. This analysis is important because the reactivity swing is mainly due to the fission products is the case of GFR self-breeder cores.

The results are reported in Table 1 and show that the current lumped fission products of the ERANOS code are not valid for GFR applications. So, for this study, the explicit description of the fission products was adopted. For future design studies, new lumped fission products dedicated to GFR cores will be processed.

Table 1 Reactivity swing results

	Simplified calculation (6 lumped fission products)	Reference calculation (88 explicit fission products)
Reactivity swing For 3×831 EFPD	- 4923 pcm	- 3957 pcm
Linear reactivity swing	- 1.97 pcm/EFPD	- 1.59 pcm/EFPD

3. Impact of the control rods introduction in the core lay-out

A Hex-Z (3D) representation of the core lay-out, with the introduction of control rod sub-assemblies, was defined. In this first 3D study, the core lay-out corresponds to the EFR design one, with 24 principal control rods (CSD) and 9 shutdown sub-assemblies (DSD), arranged in two independent systems (G1 and G2) as shown in the Figure 2.

In a first step, only the rod followers, supposed constituted by 91% of helium coolant and 9% of steel structure, are introduced in the core lay-out in order to quantify the impact of these void channels on the core reactivity. These 33 rod followers induce a reactivity decrease of 2222 pcm, which was validated by TRIPOLI comparison.

It was shown it was necessary to use refined options for the nodal flux calculation, due to the heterogeneity brought by these sub-assemblies in the core:

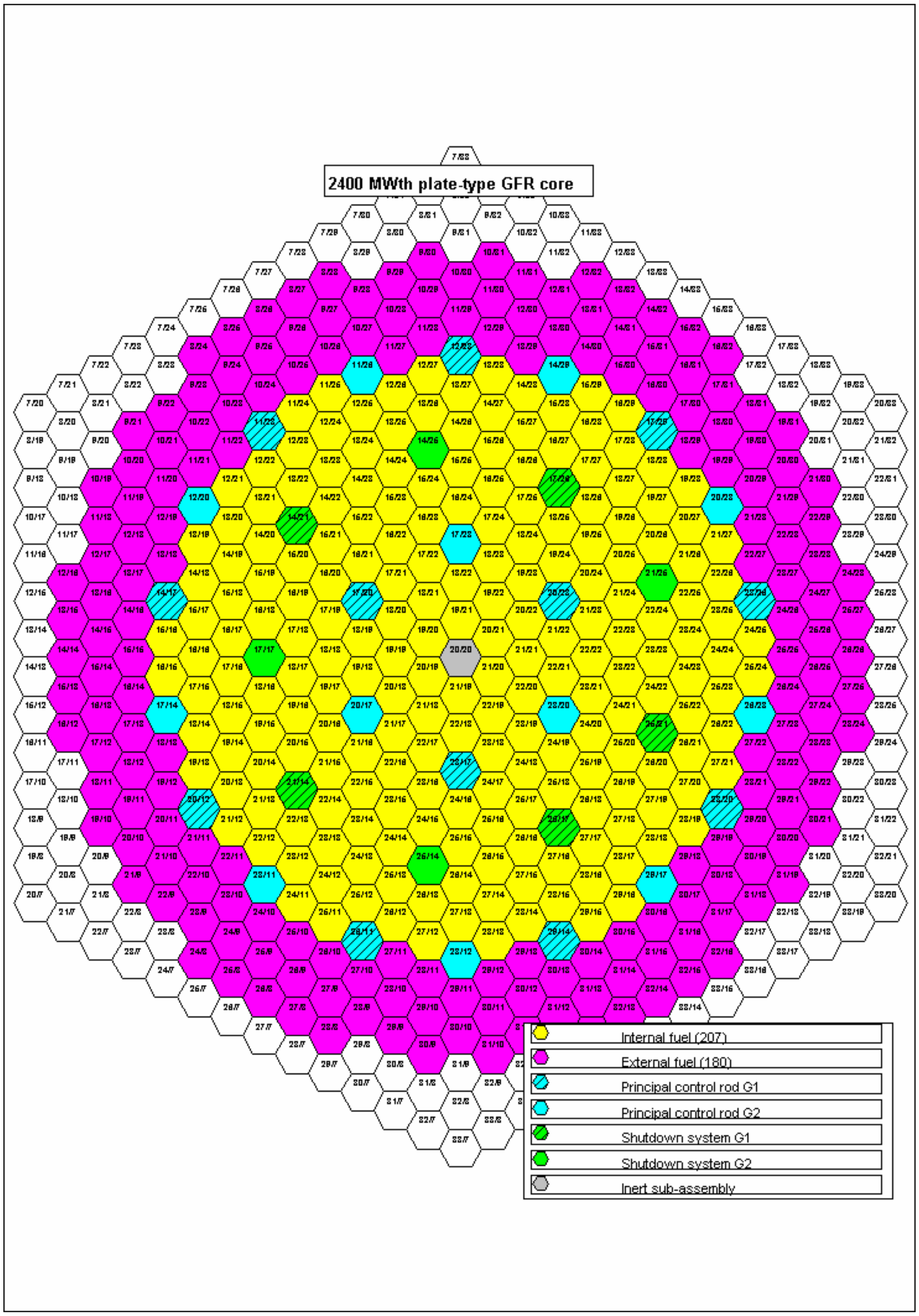
- exact transport calculation with flux and leakage expansion orders equal to 3,
- order of polynomial expansion for nodal source of 2,
- order of polynomial expansion for even flux of 6,
- order of polynomial expansion for partial current (leakage) of 1,

In a second step, the absorber zones of all the control rods were included in the upper part of these control sub-assemblies (with the interface between the follower and the absorber zones placed at the top of the fissile column). This new perturbation induces a second reactivity decrease of 1213 pcm.

Finally, the impact of the introduction of control rods on the reference core reactivity is important: more than 3000 pcm, even when extracting the absorbers.

The Pu content must be significantly increased (from 15.2%$_{vol}$ to 16.7% $_{vol}$), leading to a decrease of the breeding-gain (from –0.04 to –0.09 EOC) which requires to be compensated by design evolution for getting a self-breeder core which is one of the goals of Gen-IV criteria (Table 2).

Figure 2: 2400 MWth plate-type core lay-out



4. Calculation of the safety criteria

This Hexagonal-Z core lay-out and the reference calculation scheme allow the determination of the safety criteria associated to the control rods. As the GFR safety analyses are not presently finalized, this study relies in past studies from other existing reactors or concepts. In a first approach, we will consider EFR safety criteria, but also data derived from

SUPERPHENIX or CAPRA studies.

Table 2 2400 MWth GFR Core – First cycle Neutronic characteristics

	Adjusted core
Average Pvol (MW/m^3) Volume (m^3) Diameter (m) / Height (m)	100 24 4.44 / 1.55
SiC structures* (%vol) Gas He – coolant + gaps (%vol) (U,Pu)C (%vol) SiC matrix (%vol)	19.7 50.2 22.9 7.2
TRU enrichment (%) Heavy nuclides inventory (tons) Pu inventory (tons/GWe)	16.7 60.7 8.6
Core management (EFPD)	$3 \times 831 = 2493$
Average burn up (FIMA) Max damage (DPA SiC)	10.1 162
Average flux level – BOL ($n/cm^2/s$) Max fast flux > 0.1 MeV (n/cm2/s)	15.8×10^{14} 12.6×10^{14}
Breeding Gain – BOL/EOL	-0.18 / -0.09
Doppler constant – BOL/EOL (10^{-5})	-1576 / -1124
He depressurization – BOL/EOL (10^{-5})	192 / 255
Delayed neutron fraction – BOL/EOL (10^{-5})	383 / 343

* claddings + hex. Wrapper + “internals”

4.1 The cold reactivity excess (10$ criterion)

The SUPERPHENIX safety report specifies that the cold reactivity excess must be higher than 10$ when all the absorbers are inserted in the core. In a first approach, this criterion was also calculated for the GFR core, considering a cold temperature of 180°C (see Table 3). The CSD rods are inserted at the critical height in the full power situation and an heterogeneity correction of –30% is applied to the reactivity worth of the absorber rods, which will be rigorously calculated in future studies when the control rod design will be known.

Table 3 The cold reactivity excess (10$ criterion)

	Reactivity effects (pcm)	Uncertainties (pcm)
BOC reactivity at full power (criticality) Allowance for control **Reactivity worth of CSD and DSD** Temperature effect: full power to 180°C Neptunium decay	16 300 **-12403** 1876 80	 **3711** 563 8
10$ criterion	3830	192
Safety margin without uncertainties	-6301 pcm	
Total uncertainty		3758 pcm
Safety margin with uncertainties		**-2543 pcm**

The determination of the uncertainties is not easy because several effects cannot be precisely calculated at the current stage of GFR design studies and must be taken from past studies. This is the case for the uncertainties associated to:

- the reactivity swing,
- the heterogeneity effects,
- the reactivity worth,
- the temperature effect,
- the 10$ criterion,
- the neptunium decay.

The calculated contributions concerns the uncertainty associated to:

- the calculation method of the BOC reactivity deduced from ERANOS/TRIPOLI comparisons,
- the JEFF3.1 nuclear data (uncertainty supposed to be of the same order of magnitude than for JEF2.2 data).

The main contributions concern the uncertainties on nuclear data, the reactivity worth and the reactivity swing.

It is shown that the 10$ criterion is largely respected, with uncertainties.

4.2 The cold reactivity excess (handling error)

The SUPERPHENIX safety report specifies that the margin to criticality must be sufficient in case of a handling error consisting in a fuel sub-assembly loaded in a control rod position. We consider the worse case of an external fuel loaded in an external control rod position (see Table 4).

Table 4 The cold reactivity excess (handling error)

	Reactivity effects (pcm)	Uncertainties (pcm)
BOC reactivity at full power	16	
Allowance for control	300	
Reactivity worth of CSD and DSD –1 ext CSD +1 fuel	**-6848**	**2823**
Temperature effect: full power to 180°C	1876	563
Neptunium decay	80	8
Safety margin without uncertainties	-4576 pcm	
Total uncertainty		2879 pcm
Safety margin with uncertainties		**-1697 pcm**

This handling error criterion is largely respected well within uncertainties.

4.3 Cold shutdown from reactor at full power

The EFR safety report specifies that the reactor must be shutdown to cold state with G2-1G2 rod, starting with all the CSD rods partially inserted in order to fulfill criticality and power flattening conditions. The failure of a backup shutdown rod is the most severe situation (see Table 5).

Table 5 Cold shutdown from reactor at full power (case 1)

	Reactivity effects (pcm)	Uncertainties (pcm)
BOC reactivity at full power	16	
Allowance for control	300	
Reactivity worth of G2-1G2R (DSD)	**-7491**	**2926**
Differential dilatation	500	50
Temperature effect: full power to 180°C	1876	563
Neptunium decay	80	8
Safety margin without uncertainties	-4719 pcm	
Total uncertainty		2980 pcm
Safety margin with uncertainties		**-1739 pcm**

The differential dilatation is supposed to be caused by rods extraction due to differential dilatation between the top and the bottom of the reactor vessel. The value comes from EFR studies but should be precisely calculated in future when the reactor systems design will be known.

This criterion is largely respected with uncertainties.

The EFR safety report also specifies that the reactor must be shutdown to cold state with G1+G2-2 rods, starting with all the CSD rods partially inserted in order to fulfill criticality and power flattening conditions. The failure of two external rods is the most severe case (see Table 6).

This criterion is largely respected within uncertainties.

Table 6 Cold shutdown from reactor at full power (case 2)

	Reactivity effects (pcm)	Uncertainties (pcm)
BOC reactivity at full power	16	
Allowance for control	300	
Reactivity worth of G1+G2-2 ext rods	**-9923**	**3315**
Differential dilatation	500	50
Temperature effect: full power to 180°C	1876	563
Neptunium decay	80	8
Safety margin without uncertainties	-7151 pcm	
Total uncertainty		3363 pcm
Safety margin with uncertainties		**-3788 pcm**

4.3 Hot shutdown from reactor at full power

The EFR safety report specifies that reactor shutdown must be obtained with fully insertion of DSD with failure of one rod, starting with the CSD rods partially inserted. This shutdown must be assured to a hot state, with a temperature lower than the limit for the integrity of the SiC structures. We consider an arbitrary hot temperature of 600°C (see Table 7).

Table 7 Hot shutdown from reactor at full power

	Reactivity effects (pcm)	Uncertainties (pcm)
BOC reactivity at full power	16	
Allowance for control	300	
Reactivity worth DSD –1 rod	**-2839**	**651**
Differential dilatation	500	50
Temperature effect: full power to 600°C	640	192
Neptunium decay	80	8
Safety margin without uncertainties	-1303 pcm	
Total uncertainty		681 pcm
Safety margin with uncertainties		**-622 pcm**

This criterion is respected within uncertainties and the margin is sufficient to consider a hot temperature lower than 600°C, which is not a problem for the integrity of the SiC structures.

5. Conclusion

The first CEA neutronic studies have allowed the definition of a 2400 MWth GFR reference core, based on a plate-type CERCER fuel sub-assembly. The main characteristics of this core were calculated using a design calculation scheme. For the last stage, all the method approximations were quantified and a new reference calculation scheme was defined, taking into account in particular a precise “stripe heterogeneous cell model” in the ERANOS code to represent precisely the heterogeneous description of the core sub assembly.

This reference study has confirmed the CEA preliminary neutronic studies in support of GFR core design, and furthermore extending the study to better modelling schemes with an Hex-Z core geometry, an exact description of the innovative plate-type fuel sub-assemblies and the introduction of control rod sub-assemblies.

The reactivity effects were compared to approximate “design” results and the differences analysed in detail (these important effects will be taken into account in the CEA future design studies dedicated to the optimisation of GFR configurations).

First calculations of safety criteria associated to the control rods were performed, considering an EFR-type core lay-out and safety approach. Even if many assumptions must be deeply discussed, all the safety criteria associated to the control rods are confirmed, including uncertainties. These criteria correspond to different constraints on the reactivity worth of these absorber assemblies, and different situations of cold and hot shutdown with various configurations of rods, including handling errors. Nevertheless, many assumptions must be consolidated: the criteria come from other existing concepts (especially EFR and SUPERPHENIX) and their relevance for GFR concepts must be deeply studied, according to a general GFR safety approach. Most of the uncertainties come from past studies, equally, and must be quantified more precisely. The analysis of the future experimental program ENIGMA in MASURCA facility will also contribute significantly to the overall validation of the GFR neutronic calculations.

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