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Methodology for a Large Gas-Cooled Fast Reactor Core Design and Associated Neutronic Uncertainties

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A feasibility study of a 2400 MWth gas-cooled fast reactor using neutronic and thermo-hydraulic constraints has been performed. Previous feasibility studies [1, 2] were performed on 600 MWth cores. Considering larger cores do not imply any change in the safety approach but relax some of the design constraints on the fuel technology, on the fuel residence time and on the power density. These changes allow an enhanced economy competitiveness. The reference core has a 100 MW/m^3 power density, is based on a dispersed fuel with a fuel-to-matrix volume ratio of 50/50 and achieves a breeding ratio of 1.0 without fertile blankets. This concept possesses enhanced safety features due to a large Doppler effect owing to the presence of carbon in the SiC matrix. The possibility to remove the decay heat out of the core by natural circulation of the gas under a minimum back-up pressure is kept by limiting the core pressure drop. Numerical validation of deterministic calculations by comparisons with Monte-Carlo results are presented and uncertainties due to nuclear data are quantified. It is shown that the specificities of gas-cooled fast reactors keep bias and uncertainties within reasonably limits which are sufficient for current pre-design studies. Definite uncertainties for detailed design studies will be available after the dedicated experimental program ENIGMA in the MASURCA facility be completed.

KEYWORDS: .Gas-Cooled Fast Reactor, Design, Numerical Validation, Uncertainties, Nuclear Data

1. Introduction

Gas-Cooled Fast Reactors (GFR) appear as interesting candidates for meeting Generation IV goals on economics, safety and reliability, sustainability and proliferation resistance. First exploratory studies have been concentrated on small size GFR cores of 600 MWth power and their feasibility has been demonstrated [1, 2]. In order to enlarge screening studies in support of the fuel concept selection and to increase the overall plant performances and economics, it was also decided to investigate large power reactors.

This paper describes the approach followed for designing optimized power reactors of significant size with a within-hand fuel technology and meeting Generation IV design criteria.

These GFR present specific characteristics, compared to standard sodium-cooled oxide fast reactors (soften spectrum, presence of new materials with Si and C elements and Zr_3Si_2 reflector). In order to confirm the validity of the methods and data used for getting these results, a standard two step approach is being used. In a first step, a validation of the standard deterministic calculation method used for the GFR studies is performed using the TRIPOLI

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Monte-Carlo code. In a second step, uncertainties on several parameters due to nuclear data are quantified in an attempt to evaluate the level of GFR neutronic characteristics uncertainties compared to those of sodium-cooled oxide fast reactors.

2. Feasibility Study of a Large Size GFR Core

Exploratory studies in support of the design of a GFR have been concentrated on small size 600 MWth power GFR cores with no blanket material [2]. The plutonium inventory necessary to operate the reactor is preserved by increasing the breeding gain of the core to near-zero value. These previous studies has concluded that dense fuels like carbides or nitrides help to achieve the high actinide content of the core required to achieve the fissile self-generation. CERCER dispersed fuels, in which the actinide compound volumetric fraction can reach 50% to 70%, with the rest being occupied by the inert material (SiC matrix) playing the role of first barrier against fission product release, look to be interesting candidates. However, important power density values (up to 100 MW/m^3) need high actinide compound volumetric fraction (70%) and imply difficult constraints on the fuel fabrication and on the role of first barrier of this type of fuel. Limiting the fuel-to-matrix ratio to 50/50 relaxes these constraints but requires a low power density (56 MW/m^3). In the following we will see that larger power plants provide margins which are making the plant more economic attractive.

First assessment of cores designed with these characteristics have also the potential to meet enhanced economic competitiveness (particularly with the reduction of maintenance costs) and a still enhanced safety behavior due to their large Doppler effect owing to the presence of carbon in the SiC matrix and increased thermal conductivity between fuel and cladding. For representative transients like reactivity insertion, loss of flow and various coolant depressurizations, fast detections combined with inherent core features are providing sufficient margins in order to shut down the reactor before reaching temperature levels leading to a significant release of fission products. The same characteristics could be kept for larger power reactors and the decay heat removal out of the core by natural circulation of the gas under a minimum back-up pressure could be possible as it was for the 600 MWth cores, assuming the same low maximum core pressure drop values (typically close to 0.5 bar).

An interim burn-up objective in the range 5% FIMA with a fuel residence time limited to 10 years has been chosen for 600 MWth cores but higher burn up levels might be required to achieve the competitive system fuel cycle cost goal of Gen IV system. Regarding this burn-up objective and the fuel loading, some margins could be also recovered by considering large cores of 2400 MWth.

With this objectives, the feasibility of a large size GFR core is studied, taking into account both neutronic and thermo-hydraulic constraints. The search for reference characteristics for this large power gas cooled reactor will therefore be done with the following constraints:

- a CERCER dispersion fuel with a carbide fissile phase and a SiC inert matrix support with a ratio of 50/50,
- a 2400 MWth reactor with a power density close to 100 MW/m3,
- a discharge burn up equal to 10%,
- a maximum fuel temperature of 1200°C,
- a pressure drop close to 0.5 bar

2.1 Neutronic Domain of Feasibility

For achieving proliferation resistant and fuel cycle competitiveness, the search for a core achieving a fissile self-sufficient core without blanket materials is performed. This neutronic constraint is achieved by keeping the breeding gain (defined as the change of isotopic concentration which is normalized to equivalent 239Pu concentration) close to zero.

The core volume is fixed by the reactor power and the power density but the exact dimensions are determined by varying the H/D ratio (H is the fissile height of the core and D his diameter). In fact, for a fixed gas proportion, it is also possible to determine, with the ERANOS deterministic code system [3], the minimum H/D value for which the level of neutrons leakage verifies the constraint.

Therefore, the variation of H/D ratio as a function of the gas proportion leads to a neutronic domain of feasibility which is above a curve of minimum values of H/D acceptable (Figure 1).

2.2 Thermo hydraulic Domain of Feasibility

Two constraints are considered for the determination of the thermo hydraulic domain of feasibility: a maximum fuel temperature of 1200°C (in nominal conditions in order to assume the role of the matrix as a first barrier against fission product release) and a maximum core pressure drop close to 0.5 bar (in order to allow a decay heat removal out of the core by natural circulation of the gas under a minimum back-up pressure).

Associated to a fixed power density, it can be shown that, for a given type of fuel, the first constraint implies that the hydraulic diameter (key-parameter for the sub-assembly definition) depends only on the gas proportion in the core. The second constraint gives an expression of the H/D ratio as a function of the gas proportion and the hydraulic diameter.

Finally, with the combination of the two constraints, it is possible to determine, with the COPERNIC tool system [4], a curve of maximum values of H/D, as a function of the gas proportion, defining a thermo hydraulic domain of feasibility which is under this curve (Figure 1).

2.3 Global Domain of Feasibility

The possible common part of these two neutronic and thermo hydraulic domains of feasibility gives a global domain in which a precise core can be retained.

In fact, with the strict considered values of power density (100 MW/m^3) and maximum pressure drop (0.5 bar), the two domains of feasibility are disjointed (Figure 1). However, it is possible to obtain a common part with a slight decrease of the power density or a slight increase of the maximum pressure drop. For these studies, it was shown to fix the power density to a value of 96 MW/m^3.

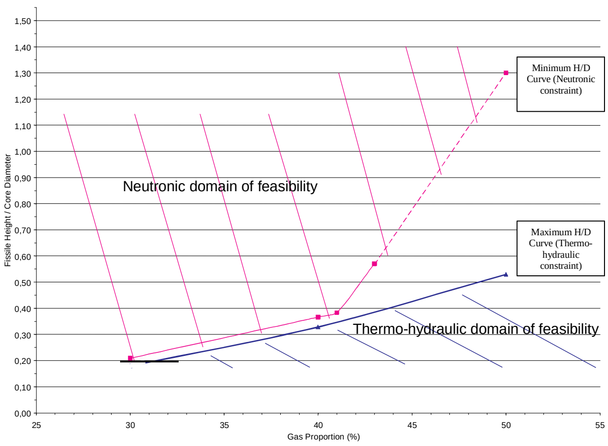

Fig.1 Neutronic and thermo-hydraulic domains of feasibility

Finally, the characteristics of the retained core are summarized in Table 1.

Table 1 Summary of the core characteristics

Reactor power (MWth)	2400	
Power density (MW/m^3)	96	
H/D ratio	0.34	
Fissile Height / Core Diameter (m)	1.55 / 4.54	
Fuel type	CERCER 50/50	
Reactor material		Core volume fractions (%)
Fuel	(U+Pu)C	25
Matrix	SiC	25
Coolant	Helium	40
Structure	SiC	10
Reflector material	Zr_3Si_2	
Pu enrichment (% volume)	15.4	
Cycle length (FPD)	2898	
Fuel burnup (wt% depletion)	10.0	
Depressurisation reactivity EOC (pcm)	212	
Doppler reactivity EOC (pcm)	1278	
Pressure drop (bar)	0.5	

The neutronic characteristics of this core do not differ significantly from the ones of the 600 MWth core and we can anticipate similar transient behaviors with in particular for LOCA accident Doppler effect overtaking very quickly the positive Depressurisation effect

3. Numerical Validation of the ERANOS Deterministic Calculation Method

The physic parameters of the core are calculated using the ERANOS deterministic code system [3] which has various computing modules for reactor physics and fuel cycle analyses.

The lattice parameters (cross sections) are generated by the ECCO module, using JEF2.2 nuclear data library and a homogeneous representation of the core cell. Fine-group (1968 groups) ECCO cell calculations are performed, providing an accurate description of the reaction thresholds and resonances, and results are condensed in a 33 broad-groups scheme.

In a second step, a cylindrical (RZ) model is used for the geometry model of the core and the calculation of the neutronic flux and derived parameters is performed with the BISTRO Sn module, using transport theory in a 33 broad-groups scheme. Two core zones with different plutonium enrichments are considered in order to flatten the power shape in the reactor.

The numerical validation of the ERANOS results was based on comparisons with the TRIPOLI Monte-Carlo code [5] for the mean neutronic spectrum in the core, radial reaction rate traverses (in the core and the Zr_3Si_2 reflector) and the reactivity.

The first comparison concerns the mean neutronic spectrum (Figure 2). The comparison between the two codes shows a very good agreement for this parameter on the complete energetic domain. The discrepancies between ERANOS and TRIPOLI values are lower than 1% in all energetic groups. This confirms the correct treatment of the slowing-down process in ERANOS code which is important in this concept due to the presence of C and Si elements.

The second comparison concerns radial reaction rate traverses, representative of high energy neutrons (^{238}U fission), intermediate neutrons (^{235}U fission) and low energy neutrons (^{238}U capture). These traverses are compared in the core and in the reflector on Figures 3-5.

All the reaction rate traverses show a very good agreement in the core with discrepancies lower than 2%, confirming the first tendency obtained for the mean spectrum in the core.

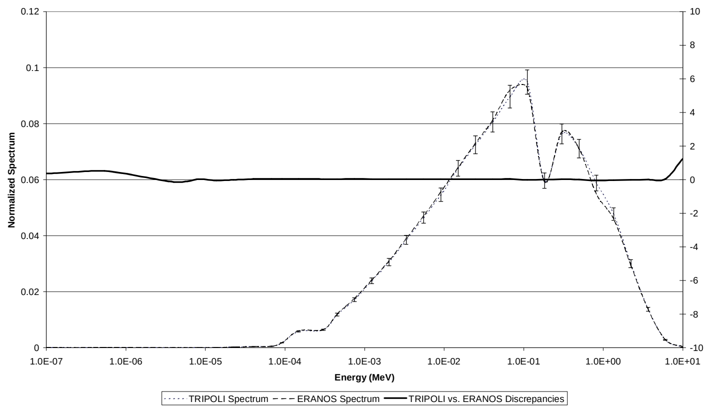

Fig.2 Mean spectrum in the core: ERANOS vs. TRIPOLI comparison

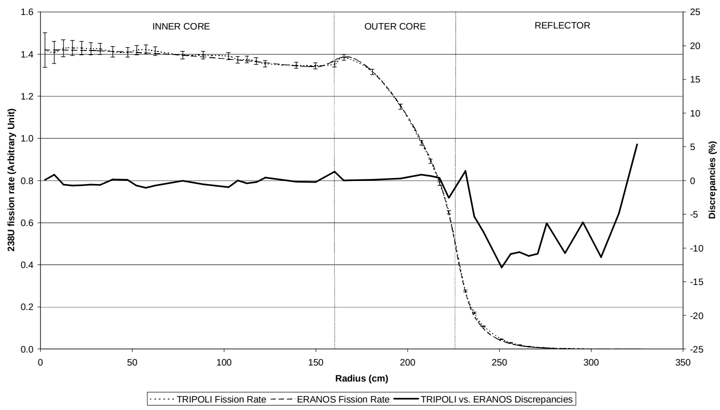

Fig.3 Radial ^{238}U Fission Rate Traverse: ERANOS vs. TRIPOLI comparison

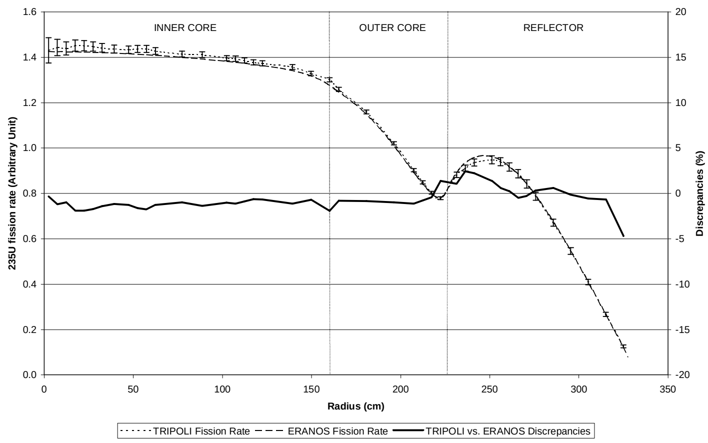

Fig.4 Radial ^{235}U Fission Rate Traverse: ERANOS vs. TRIPOLI comparison

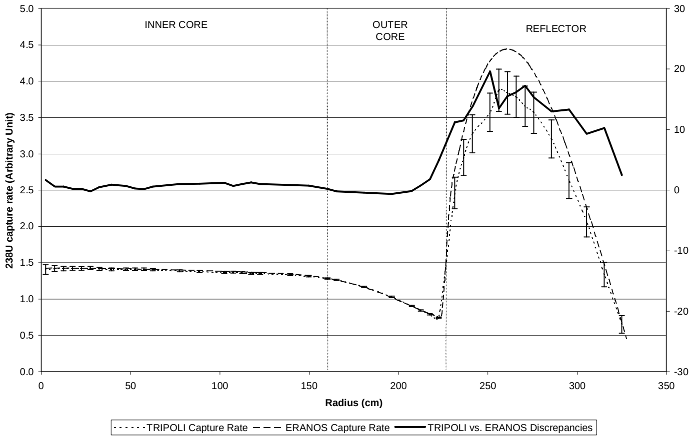

Fig.5 Radial ^{238}U Capture Rate Traverse: ERANOS vs. TRIPOLI comparison

The discrepancies in the reflector are more important (about 10% for ^{238}U fission and 20% for ^{238}U capture). It can be noted, in particular, a lack of neutrons calculated by ERANOS in the high energy range and a too important number in the low energy range. However, the increasing of the ^{235}U fission rate in the reflector, characteristic of the slowing-down of the neutrons in this medium, is well calculated with ERANOS code, due to compensation effects.

These discrepancies are due to the fact that the calculation method has not been optimized for the precise treatment of this medium. They should be reduced by using a reference calculation method for generating broad group reflector cross sections, taking into account their spatial and energetic variations in this medium. However, these discrepancies are not prohibitive for the current GFR studies.

The third comparison concerns the reactivity. The results obtained with ERANOS and TRIPOLI, detailed in Table 2, show a residual discrepancy of 309 pcm between the two codes. which will be investigated, in particular, in relation with the discrepancies in the reflector. Nevertheless, this bias is not prohibitive, considering the current status of the GFR studies.

Table 2 Reactivity results

ERANOS Reactivity (pcm)	6802
TRIPOLI Reactivity (pcm)	7111 ± 42 (3σ)
ERANOS-TRIPOLI (pcm)	- 309

4. Study of Uncertainties due to Nuclear Data

GFR have several specificities compared to standard sodium-cooled oxide fast reactors characteristics: new materials with Si, C and Zr elements and softened spectrum due to the presence of the inert matrix.

It is important to determine if these specificities imply new sources of uncertainties which would impact the preliminary conclusions of the GFR studies. So, uncertainties due to nuclear data have been quantified for the reactivity and different reaction rate ratios in the core, representative of various energetic domains.

Two different nuclear data libraries are used: data based on JEF2.2 [6] evaluation and ERALIB1 [7] adjusted data. 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, 355 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, associated to 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). The ERALIB1 uncertainties calculated here are only tendencies which will be validated.

Perturbation theory is used to combined estimated (and calculated) cross-section uncertainties with calculated sensitivity coefficients to determine the variance of these parameters of interest. The objective is to calculate the effect of a given uncertainty, $\Delta\sigma$, on a specific nuclear design parameter of interest, I; or, in other words, to predict the uncertainty ΔI of a calculated integral parameter due to a given cross-section uncertainty $\Delta\sigma$.

It can be established the expression of ΔI using the first-order perturbation theory:

$$\Delta I = [Var(I)]^{1/2} = [\sum_{i,j} \frac{\partial I}{\partial \sigma_i} \frac{\partial I}{\partial \sigma_{ji}} D(\sigma_i, \sigma_j)]^{1/2} \tag{1}$$

where $\frac{\partial I}{\partial \sigma_i}$ are the ERANOS calculated sensitivity coefficients and $D(\sigma_i, \sigma_j)$ are the elements of the variance-covariance matrix.

The JEF2.2 variance-covariance matrix was generated with data based on evaluations (for ^{238}U and ^{239}Pu), data extracted from literature and expert judgement. The ERALIB1 variance-covariance matrix was generated by the adjustment process for 17 main isotopes and especially for ^{235}U, ^{238}U, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu and ^{12}C which are present in the GFR concept studied.

The Table 3 present the uncertainties calculated for the reactivity, the (^{239}Pu Fission /^{235}U Fission) reaction rate ratio and the (^{238}U Fission /^{235}U Fission) reaction rate ratio. These ratios are calculated at the centre of the core and are respectively representative of intermediate and fast neutrons. They also concern important isotopes for the studied GFR concept.

Table 3 Uncertainties due to nuclear data calculated

	Reactivity	(^{239}Pu Fiss /^{235}U Fiss)	(^{238}U Fiss /^{235}U Fiss)
JEF2.2 Uncertainty	1390 pcm	2.4 %	3.3 %
ERALIB1 Uncertainty	312 pcm	0.5 %	1.2 %

The important JEF2.2 uncertainty on reactivity is essentially due to the ^{239}Pu fission for which the sensitivity coefficient is very important. The contribution of this reaction represents 72% of the total uncertainty. With the adjusted data, the contribution of this reaction is decreased and the total uncertainty is better distributed on the different elements.

It can be noted that the total uncertainty on reactivity with the adjusted data is slightly greater than the typical uncertainty for the reactivity of standard sodium fast reactors but remains a low value. Even if the validity of ERALIB1 data must be verified for GFR applications, the obtained tendency is sufficient at the current state of the GFR studies.

The important JEF2.2 uncertainty on (^{239}Pu fission /^{235}U fission) ratio is essentially due to the ^{239}Pu fission which represents the direct effect. This reaction represents 92% of the total uncertainty. With the adjusted data, this direct effect remains the main contribution but the reduction of the variance-covariance data for this isotope implies an important decrease of the total uncertainty.

For the (^{238}U fission /^{235}U fission) ratio, the direct effect (^{238}U fission) represents only 42% of the total uncertainty. Other important contributions are ^{238}U inelastic scattering, Si elastic scattering and Si inelastic scattering. With the adjusted data, the ^{238}U contributions are reduced but the Si contributions remain important because this isotope was not adjusted.

Finally, we can say that these uncertainties are consistent to standard sodium-cooled oxide fast reactors ones and not prohibitive for the current state of GFR studies.

5. Conclusion

A combined neutronic and thermo-hydraulic method has allowed the definition of a large power gas cooled reactor (2400 MWth) with a core achieving a breeding ratio of 1.0 without fertile blanket. The margins recovered by considering larger cores than the 600 MWth cores of previous studies [1] have allowed to limit the constraints on the fuel and to increase the cost effective aspect of the fuel cycle with a maximum fuel burnup up to 10% (also due to the safe fuel technology used) and the power density reaching nearly 100 MW/m^3. The reference configuration is characterized by a CERCER dispersed fuel with a 50/50 fuel-to-matrix ratio and an acceptable core pressure drop (0.5 bar), allowing the removal of the decay heat out of the core by natural circulation of the gas under a minimum back-up pressure.

The ERANOS deterministic calculation method used for the characterization of this type of core was validated by comparison with a reference Monte-Carlo code (TRIPOLI) for several parameters: reactivity, mean neutronic spectrum in the core and various reaction rate traverses in the core and the Zr_3Si_2 reflector. It was shown that the discrepancies between the results of the two codes were limited and not prohibitive for the current state of GFR studies.

The uncertainties due to JEF2.2 and ERALIB1 adjusted nuclear data were calculated for the reactivity and reaction rate ratios at the center of the core. Even if the validity of adjusted ERALIB1 data must be confirmed for GFR applications, it can already be said that the specificities of GFR concepts (softened spectrum and presence of Si element in particular) imply limited additional sources of uncertainties. The data used for designing GFR are judged sufficiently valid for the accuracy level required at this stage of the design studies.

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