Low Void Effect (CFV) Core Concept Flexibility from Self-breeder to Burner Core

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Abstract – *In the frame of the French strategy on sustainable nuclear energy, several scenarios consider fuel cycle transition toward a plutonium multirecycling strategy in Sodium cooled Fast Reactor (SFR). Basically, most of these scenarios consider the deployment of a 60 GWe SFR fleet in two steps to renew the French PWR fleet. As scenarios do investigate long term deployment configurations, some of them require tools for nuclear phase-out studies. Instead of designing new reactors, the adopted strategy does focus on adaptation of existing ones into burner configurations. This is what was done in the frame of the EFR project at the end of the 90's using the CAPRA approach (French acronym for Enhance Plutonium Consumption in Fast Reactor). The EFR burner configuration was obtained by inserting neutronic penalties inside the core (absorber material and/or diluent subassembly). Starting from the preliminary industrial image of a SFR 3600MWth core based on Low Sodium Void concept (CFV in French), a "CAPRA-like" approach has been studied. As the CFV self-breeding is ensured by fertile blankets, a first modification consisted in the substitution of the corresponding depleted uranium by "inert" or absorber material leading to a "natural burner" core with only small impact on flux distribution. The next step toward CAPRA configuration was the substitution of 1/3 of the fuel pins by "dummy" pins (MgO pellets). The small spectrum shift due to MgO material insertion leads to an increase Doppler constant which exceeds the value of the reference case. As the core sodium void worth value is conserved, the CFV CAPRA core "safety" potential is quite similar to the one of the reference core.*
Fuel thermo-mechanical requirements are met by both nominal core power and fuel time residence reduction. However, these reduction factors are lower than those obtained for EFR core. The management of the enhanced reactivity swing is discussed.

I. INTRODUCTION

For the next generation fast reactor design, the Generation IV International Forum (GIF) defined global objectives in terms of safety improvement, sustainability, waste minimization and non-proliferation.. Among the possibilities studied at CEA, Sodium cooled Fast Reactor (SFR) are studied as potential industrial tools for next decade's deployment. Many efforts have been made in the last years to obtain advanced industrial core designs that comply with both sustainability and safety concerns[1,2].

Large power core designs studies focused on Low Sodium Void Core concept (CFV acronym in French) for which sodium void worth is close to zero, even negative, at end of cycle. This "low sodium void value" is achieved by a heterogeneous geometrical design involving both internal fertile blankets and upper sodium plenum. This configuration not only yields a near zero void effect by also a global improvement on the thermal sodium expansion feedback coefficient. This leads to global improvement toward a "natural safe behavior" of the core for unprotected loss of flow transients.

This specific design has been used for SFR deployment scenarios studies in the current century. As a matter of flexibility scenarios have to investigate several strategies including net nuclear energy growth and nuclear phase out. In both cases, inherent properties of SFR can bring solutions in terms of design adaptation to meet the associated requirements.

In the case of phase out studies, the main goal is to be able to switch from self-breeder core to plutonium burner for inventory minimization concerns.

Starting from the current design of the large power industrial core used in the scenario a preliminary image of a plutonium burner core has been investigated following guidelines of the CAPRA[3] approach issued from EFR[4] project in the 90's.

II. SCENARIO NEEDS

II. Phase out

The reference deployment of next generation nuclear reactors consider fuel cycle based on plutonium multirecycling strategy in Sodium cooled Fast Reactor (SFR). Basically, most of the current scenarios[5] consider the deployment of a 60 GWe SFR fleet in two steps to renew the French PWR fleet. Fig.1 shows a typical transition from PWR (involving today second and third generation of these reactors) to SFR, these latter being deployment around 2040.

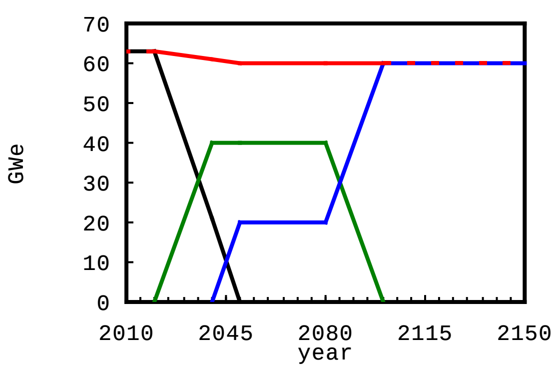


Fig. 1. Typical transition scenario for the 60 GWe French fllet (dark=current PWR, green=EPR, blue=SFR).

In order to achieve sustainability, SFR core has to exhibit self-breeding performances to master the plutonium inventory when equilibrium is reached. As scenarios do investigate long term deployment configurations, some of them require tools for nuclear phase-out studies. For these particular scenarios, the need does focus on nuclear systems that enable to reduce the fleet plutonium inventory in an efficient way. To do so, one can choose either to develop dedicated tools such as Accelerator Driven System or to rely on inherent flexibility of SFR.

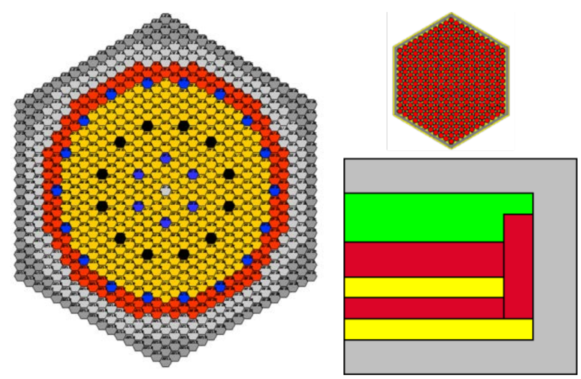
This "exercise" has already been performed during the 90's in the frame of the European Fast Reactor (EFR) within the French Act on Waste management. The main purpose was to have fast reactor core that could "burn" plutonium coming from multirecycling MOX PWR. The core design, labeled CAPRA (French acronym for Enhance Plutonium Consumption in Fast Reactor), was derived from self-breeder EFR configuration. The CAPRA approach involving many core changes leads to a burner core.

II.A. Reference core

The reference core is a MOX 3600 MWth, axially heterogeneous core based on the Low Void Concept (hereafter labeled CFV core) proposed by CEA. The Fig. 2 displays schematic axial and radial cuts of core. The radial plot shows

- mixture of fissile (red) and fertile (yellow) layers for the inner plutonium content.
- lower fertile zone for the outer plutonium content,
- a large upper sodium plenum.

This combination leads to an enhanced neutrons leakage in the sodium voided configurations and also improvement in the coolant thermal expansion feedback coefficient. Self-breeding is achieved by internal fertile zones (fertile pellets inside the fuel pin).



	Inner Diluant S/A		Control Rod S/A
	Inner Fuel S/A:		Shutdown Rod S/A
	Outer Fuel S/A		Reflector S/A:

Fig. 2. Radial and axial cut of the reference core.

Core performances were calculated using a plutonium quality corresponding to a representative isotopic content available around 2035 (hereafter labeled "Pu2035", see Table I.) for the start of the deployment scenario.

TABLE I

Isotopic content of the Pu2035 (%mass)

Isotope	Pu238	Pu239	Pu240	Pu241	Pu242	Am241
(%)	3.57	47.39	29.66	8.23	10.37	0.78

The main core characteristics are presented in Table II. The core configuration leads to a self-breeder core with a low reactivity swing. The reference cladding material is a representative martensitic Oxide Strengthened Steel (ODS) that enable to reach higher maximal damage rate than ferritic steel like AIM1 used by EFR concept. The total sodium void worth is slightly negative at the end of the cycle.

TABLE II

Reference core characteristics

Characteristics	Unit	Value
Power	MWth	3 600
Average PuO_2 Content	%vol	22
Pu Mass	t	11.9
Cycle length	efpd	388
Number of batches		5
$\Delta\rho$ cycle	(pcm)	-450
β_{eff}	(pcm)	365
Sodium void worth	($)	-0.4
Fissile Doppler constant	($)	-1.7
Max linear heat rating	(W/cm)	475
Max Burnup	(GWd/t)	165
Max damage rate	(DPA)	148

II.A. Scenario Requirements

The phase out study needs specific assumptions for fuel cycle management (nuclear material flow, reprocessing and manufacturing capacities, reactor shutdown schedule, etc..). For reactor, there is a need of neutronic system with efficient plutonium burning capacities. In order to minimize the impact on the reactor design, the burner core configuration has to be obtained while conserving some of the reference core characteristics such as:

- geometrical data:
    - core diagrid dimensions
    - control rod number and associated positions
    - S/A heigth
- fuel and pin performances :
    - « Acceptable » maximal burnup (~185 GWd/t)
    - Maximal damage rate compatible with ODS cladding (200 DPA)
    - 300 K merging to fuel melting (~485 W/cm)

Concerning the nominal core performances, the CAPRA configuration has to comply with some specific requirements:

- "Manageable" cycle reactivity loss. The value has to be less that 20 pcm/efpd which correspond to Phenix case.
- « no degradation » of feedback coefficient. Sodium void worth and Doppler constant (K_D) could be taken as global estimators to quantify performances.

III. FROM BREEDER TO BURNER CORE

III.A. Philosophy of CAPRA approach

There are several ways to move from a self-breeder configuration into a burner one but all needs some more or less in depth adaptations of the core design:

- change in the power density (by means of initial fuel S/A in the core), as for PRISM or ABR concept[6,7],
- change in the fuel nature (plutonium on inert matrix),
- insertion of neutronic penalties in the core.

The latter case offers a wide range of possibilities and was the basis of the CAPRA approach used in the frame of the EFR project. "Neutronic penalties" can be obtained by either modifying the geometry of the core to enhance neutron leakage or inserting non-heavy nuclei material ("inert" or absorber material) to decrease the fissile volume fraction. These modifications lead to increase the fissile content in order to reach acceptable cycle length. As the plutonium content reaches high level, the U^{238} proportion goes down and the core exhibits less abilities to produce fissile material.

In the present case, the CFV reference core has several fertile zones inside the fuel element itself and thus shows additional degree of freedom for CAPRA approach compared to EFR homogeneous core.

Fig 3 displays schematic radial and axial cut of core and subassembly for reference and CAPRA core configurations. The main adaptations consist in:

- fertile material (yellow) substitutions by inert material,
- insertion of "inert" pins in the fuel subassembly (green pins) .

Two solutions investigated in the CAPRA project were not implemented here:

- MOX fuel nature change to nitride or uranium-less fuel (inert matrix). As the phase-out scenario is based on reprocessing strategy, MOX fuel nature is the most suitable form for industrial purpose.
- Insertion of diluent (with or without absorber material) subassembly. As local penalties involve perturbation in the flux distribution compared to the reference core, a fine tuning of the core characteristic's is needed especially in order to meet several basic requirements such as maximum linear heat rating, reactivity swing during cycle, safety margin for shutdown issues, etc.. This solution could be investigated in a next phase of the study.

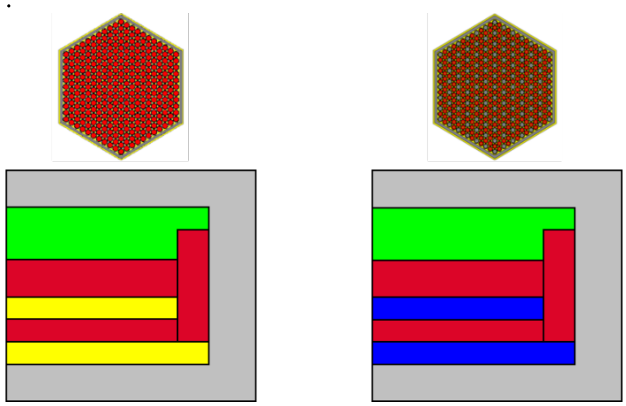


Fig. 3. Radial and axial cut of the reference (right) and CAPRA (left) cores.

III.B. Calculation tools

Calculations were performed using the ECCO/ERANOS[8] code package together with JEFF3.1.1 cross sections library[9]. The transport option has been used for all calculations with a 33-group working library which has been generated from a 1968-group master library. For comparison, fine-group (1968 groups) ECCO cell calculations were also performed, and provide an accurate description of the reaction thresholds and resonances. The fine-group calculation has been performed for the 2-dimensional description of a sub-assembly. Because the broad 33-group library has been generated from this master fine group library, it has been found that differences on the core parameters are small.

For the core geometry model of the core, a cylindrical (RZ) model is adopted. Validity of the RZ model was assessed against the results of heterogeneous (hexagonal-Z) calculation using the TGV/VARIANT[10] 3-dimensional nodal transport code. Heavy nuclei depletion calculations are performed with an extended chain up to Cf252 isotope.

Sodium void reactivity effects have been obtain by set sodium concentration near zero for the fuel zones including fissile and fertile or inert medium zones of the inner plutonium content. The upper sodium plenum has also been considered.

For Doppler constant estimation, only fissile zones were considered in order to have a coherent comparison between configurations.

IV. BURNER CORE CONFIGURATION

IV.A. Impact of incremental changes

In order to evaluate individual impact of any modification from the reference configuration, incremental change strategy was applied. The evaluated impacts on core characteristics as well as plutonium consumption (assuming 1450 MWe electricity gross) are displayed in Table III.

TABLE III

CAPRA configurations characteristics

Conf.	Ref	A	B	C	D	E
Power (MWth)	3600	3600	3600	3600	3600	3000
Pu Content (%)	22.3	21.7	25.4	36.5	37.5	35.8
Max BU (GWd/t)	163	171	194	233	220	187
$\Delta\rho$ cycle (pcm)	-800	-2600	-3600	-5500	-5500	-4600
β_{eff} (pcm)	365	361	350	324	323	325
Sodium Void ($)	-0.40	-1.6	-0.7	-1.1	-0.92	-1.15
K_D ($)	-1.70	-3.1	-2.9	-2.5	-2.2	-2.4
Cons. (kg/TWeh)		-23.4	-34.0	-55.1	-57.5	-55.6

Configuration A:

Starting from the reference core, the main geometrical modifications applied consisted in:

- Substitution of fertile pellets by MgO pellets
- Reduction of the fuel residence time to 3x500 efpd to meet criterion on maximum burnup

The choice of the MgO inert material was driven by safety considerations. Its low atomic number composition lead to a soften neutron spectrum that bring important margins for both sodium void worth and Doppler constant. As further geometrical modifications will lead to higher initial plutonium content, these margins will help keeping the same feedback level as what is observed in the reference core.

Beside improvement in the feedback coefficients, the main impact of these modifications relies on the cycle reactivity loss which increases up to 7 pcm/efpd.

Configuration B and C:

In order to introduce inert material inside the fissile part of the subassembly, fuel bundle has to been redesign. As the new configuration has to meet the maximum linear heat rating criteria the fuel pins number has to remain quite constant. To do so, pin diameter was first decreased to be able to move from 331 pins to 469 (configuration B).

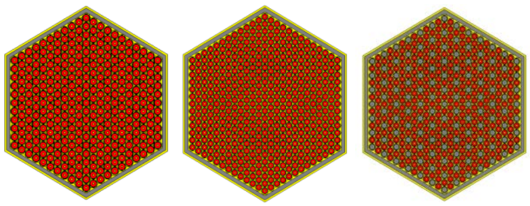


Fig. 4. Radial of the fuel bundle: 331 pins (left); 469 pins (center) and 342+127 CAPRA (rigth).

This new bundle leads to new fuel, structures and sodium volume fractions:

- Fuel volume fraction goes down from 44% to 36 %
- Sodium volume fraction increases from 28 to 32 %.

This leads to higher plutonium content that does enhance the plutonium consumption, but also degrades feedback coefficients as well as cycle reactivity loss compared to previous configuration.

Configuration C corresponds to substitution of around one third of the fuel pins by inert pins (MgO pellets instead of MOX ones). The corresponding fuel volume fraction inside the fuel bundle drops to 26%. The associated plutonium content is 35% which leads to a reactivity loss around -18 pcm/efpd, close to the acceptable limit. The evaluated maximum burnup value (233 GWd/t) is beyond the required criterion. This high level plutonium content leads to harder neutron spectrum that favors neutron leakage so that sodium void worth level is still below the one of the reference case. Doppler constant value is a bit lower but is still higher but still showes improvement compared to the self-breeder core.

Configuration D:

In depth analysis of the configuration C shows that the increase of maximum burnup value comes from the combined effect of the radial insertion of the MgO inside the fuel part of the subassembly. Large volumes of MgO (both the fuel part and in the medium and lower inert zone of the inner core) lead to strong spectrum transition at the fuel/inert MgO interfaces. The local effect can be seen in the Fig. 5 which represents a 1D Monte-Carlo model of the inner fuel pin. Here, several materials of the central inert zone fuel have been tested for fission form factor evaluation.

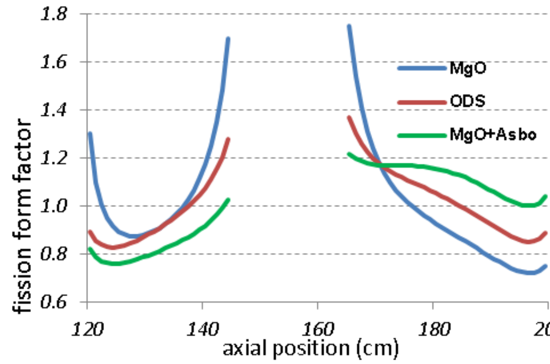


Fig. 5. Fission form factor as a function of the material composition of the central inert zone.

The red curve corresponds to the same steel composition as the cladding material (ODS) while the blue one corresponds to MgO material. Fuel plutonium content was set to 35% according to configuration D characteristics. The light elements of MgO material give a local modification of the flux coming from fissile zones and the fission form factor reaches 1.7 compared to 1.3 for steel composition.

Several solutions have been tested in order to decrease the peaking value. The most efficient consists in inserting absorber material in the transition zone. One example in which 1 cm of steel containing boron carbide have been located in both lower and upper part of the central inert zone. The associated result is shown by the green curve in Fig. 5.

This solution has been applied in configuration. The absorption level of the core is slightly increased and the plutonium content need to be updated. If the maximum burnup value obtained by the RZ calculation scheme does not seem to be impacted, more realistic 3D calculations show a 20% reduction of the peaking factor. However the associated maximum linear heat rating is still beyond the requirement.

Configuration E:

The last parameter that can help reaching the targeted value is the core nominal power. Moving from 3600 MWth to 3000 MWth gives acceptable maximum value without degrading plutonium consumption. In addition, the change in power density gives lower cycle reactivity loss. The associated feedback coefficients values meet the requirement in terms of “safety” level:

- Sodium void worth is still negative
- Doppler constant is not degraded
- Both values show small gain compared to the reference case.

This latter configuration was the basis of a full 3D core description to validate the observed performances.

IV. B. 3D Burner core configuration

Starting from configuration E, cores modifications were included in a 3D core description using “best estimate” calculation scheme:

- 3D flux transport calculation scheme
- Reloading scheme using 1/5 pattern with control inserted at beginning of cycle
- Equilibrium search (each subassembly is replaced twice)

To do so, some adaptations have been performed:

- MgO pellets have been put into inert pins only
- MgO pellets of the lower and central inert zones (former fertile zones) have been replaced by ODS ones to limit fission peaking factor inside the fuel pin. The pins configuration is displayed in Fig. 6.
- Natural boron carbide used in the control rod design has been replaced by enriched B^{10} boron carbide (90% B^{10}) in order to deal with the marge reactivity swing of the burner configuration.

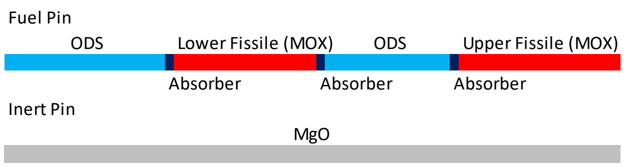


Fig. 6. Schematic axial description of the fuel and inert pin of the fuel bundle.

The main core performances are listed in Table IV. Most of them are very similar to those obtained for configuration E using a “low” level calculation scheme. The main improvement is the 3D validation of the maximum linear heat rating that fits the initial requirement. If the maximum burnup exceeds the value of the self-breeder core, it seems acceptable from the thermo-mechanical point of view. The plutonium consumption evaluated at the end of irradiation is 56 kg/TWeh.

TABLE IV

CAPRA 3D core characteristics based on Pu2035 feeding

Characteristics	Unit	Ref.	CAPRA
Power	MWth	3 600	3 000
Average PuO_2 Content	%vol	22	35
Pu Mass	T	11.9	12.6
Cycle length	Efpd	388	300
Number of batches		5	5
$\Delta\rho$ cycle	(pcm)	-450	-4 650
β_{eff}	(pcm)	365	325
Sodium void worth	($)	-0.4	-1.1
Fissile Doppler constant	($)	-1.7	-2.2
Max linear heat rating	(W/cm)	475	482
Max Burnup	(GWd/t)	165	182
Max damage rate	(DPA)	148	113

The characteristics of the CAPRA core image has to be compared to results obtained for EFR core. One of the main differences relies on the level of the Doppler constant. For EFR CAPRA the Doppler constant was 25% lower than the one of the reference EFR MOX core because of the use of diluent subassemblies inside the core that increased the power density and leads to higher initial plutonium content.

The current CFV CAPRA design based on simple material and geometrical modifications helps improving feedback coefficient while maintaining acceptable plutonium consumption.

IV. CONTROL ROD DESIGN

In order to obtain a realistic design of the core a large amount of work has been done for the fuel element especially in terms of neutronic effects of the different materials involved in the diverse inert zones inside or outside the fuel pin itself.

The same kind of approach needs to be done for the control rod design. The only modification that was applied for rod design was the substitution of the natural boron by enriched one. A lot of work remains to be done to ensure that the current modification is able to satisfy the whole set of criteria relative to reactivity control and reactor shutdown.

If we put aside complex core configurations such as fuel handling errors fro hot and cold core states, control rods efficiency has to comply with:

- reactivity management at beginning of cycle,
- reactor shutdown at beginning and the end of cycle

In addition, control rod efficiency has to be compatible with “reasonable” margin on boron carbide melting temperature. With the modified boron content rod design, equilibrium beginning of cycle criticality is achieved with 25 cm rods insertion in the inner fissile zone. If we assume a conservative value for the ${}^{90\%}B^{10}{}_4C$ conductivity and a representative power peeking factor (~1.5), maximum boron carbide centerline temperature can be estimated. The original rod design has 37 absorber natural born carbide rods corresponding to 24% volume fraction.

Fig. 6 shows the margin to melting evaluated from the maximum center line boron carbide temperature after a certain number of irradiation cycles. Here, several pin designs have been tested for which we made the assumption that the total absorber volume fraction inside the control rod subassembly remains constant. The current design based on 37 pins rods does not show any realistic margin since melting occurs before the end of the first irradiation cycle. With smaller rod pin diameter, solution can be found. At least 91 pins are required to have design that can remain two cycles or more in the core. The use of the natural boron carbide instead of $^{90\%}B^{10}$ boron carbide gives significant margin on fuel melting since 37 pins design is still 1400 K above fuel melting after 5 irradiation cycles.

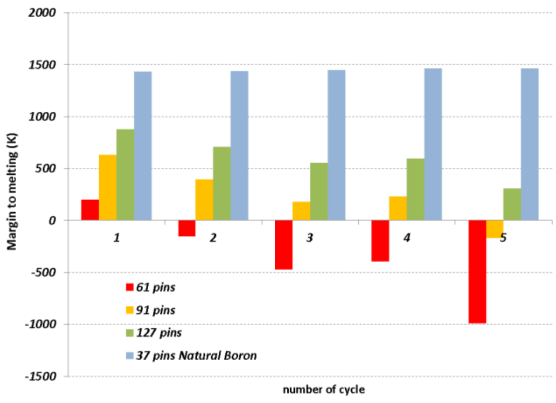


Fig. 6. Boron carbide margin to melting as a function of rod pin number in the control rod design

For reactor shutdown issues, the total control rod worth required for the CAPRA core has to account for the following items:

- reactivity loss during cycle : ~ 4600 pcm
- reactivity Doppler effect from nominal full power state to cold isotherm shutdown : ~1000 pcm
- management of fuel handling errors : ~2000 pcm
- integration of incertainty level : ~1000 pcm

The sum of all these effects gives a total close to 9000 pcm. Assuming that the control rod worth does not depend on the number of pin for an imposed volume fraction, several material and geometrical rod configurations based on 127 absorber rod pins were investigated. Starting from a 127 absorber pin rods design (see Fig. 7) the effect of B^{10} content on control rods worth has been investigated.

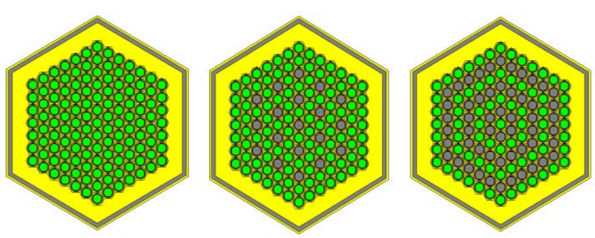


Fig. 7. Schematic view of 127 pins control rod designs with or without moderator (gray pins).

As brief sensitivity study showed that control rod absorption level can be enhanced with the use of moderator material, some versions of the design with ZrH_2 and other similar material has been tested.

TABLE V

Control rod (CR) worth of various design versions

B4C pins	B10 content (%)	Mod. pins	Mod. material	CR worth (pcm)
127	19.8	0	None	-7 700
127	30	0	None	-9 000
127	48	0	None	-10 100
127	90	0	None	-12 500
108	90	19	ZrH_2	-12 300

As shown in Table V, Initial $^{90\%}B^{10}$ configuration leads roughly to a 3000 pcm margin. Criteria on total control rod worth can be achieved with either 30% or 48% B^{10} content. With these configurations the estimated boron carbide maximum centerline temperatures decrease and the gain can be used to find optimized solutions that can meet the whole set of control rods requirements.

The analysis of configurations with moderator material showed that such pins can bring new degree of freedom in order to:

- boost the global absorption rate level by shifting neutrons to lower energy
- decrease the “shadow” effect between absorber pins (spatial self-shielding) and thus decrease the peaking form factor

CFV CAPRA core control rod design is currently on-going using the above results. The optimized solution will have to be validated on the basis of Monte-Carlo calculations.

V. CONCLUSIONS

The reference deployment of next generation nuclear reactors consider fuel cycle based on plutonium multirecycling strategy in Sodium cooled Fast Reactor (SFR). As scenarios do investigate long term deployment configurations, some of them require tools for nuclear phase-out studies. Instead of designing new reactors, the adopted strategy does focus on adaptation of existing ones into burner configurations.

Starting from the low sodium void core configuration (CFV), the self-breeder reference core has been turned into a plutonium burner core using the CAPRA-like approach already used in the 90’s in the frame of the EFR project.

As the CFV self-breeding is ensured by fertile blankets, a first modification consisted of the substitution of the corresponding depleted uranium by “inert” or absorber material leading to a “natural burner” core with only small impact on flux distribution. The next step toward CAPRA configuration was the substitution of 1/3 of the fuel pins by “dummy” pins (MgO pellets). The small spectrum shift due to MgO material insertion leads to an increase Doppler constant which exceeds the value of the reference case. As the core sodium void worth value is conserved, the CFV CAPRA core “safety” potential is quite similar to the one of the reference core.

Preliminary studies on control performances show that the initial design can not meet the whole set of reactor shutdown requirements, especially the margin to melting issues. A set of material and geometrical modifications of the design gives clear insight of possible solutions that have to be confirmed by reference calculations.

The resulting core configuration exhibits acceptable plutonium consumption without degrading basic feedback coefficient values compared to the self-bredeer reference core. Reactivity management as well as reactor shutdown capacity seem achievable with some minor design adaptations.

The move from a self-breeder reference core to a burner core presented here has been performed while keeping the main feedback characteristics of the CFV core concept. Plutonium burning efficiency can be tuned according to scenarios demands and this kind of design offers a high degree of flexibility for nuclear material management.

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