Improvement of minor actinides transmutation performances in fast reactors using fissile material

T. Kooyman, L. Buiron, G. Rimpault

IMPROVEMENT OF MINOR ACTINIDES TRANSMUTATION PERFORMANCES IN FAST REACTORS USING FISSILE MATERIAL

Timothée Kooyman[1,*], Laurent Buiron[1], Gérald Rimpault[1]

1- *CEA,DEN,DER Cadarache 13108 Saint Paul lez Durance CEDEX France*
**Corresponding author: timothee.kooyman@cea.fr*

In Fast Reactor systems the heterogeneous minor actinides transmutation is a promising solution to transmute minor actinides and reduce the long-term radiotoxicity burden associated with nuclear waste without impairing core operations. In this approach, minor actinides are loaded in dedicated targets using uranium dioxide as support matrix at the core periphery. However, due to their location those targets are exposed to a lower flux level and fuel temperature inside is lower than for standard fuel. This has a negative effect on transmutation performances as those depend on the neutron fluence on the targets, as well as on fuel behavior under irradiation due to limited fuel restructuration and potential high solid swelling coming from important helium production (alpha decay of minor actinide). Additionally, plutonium breeding in the blankets leads to a consequent shift in power in the blankets during irradiation, from 0.5 MW up to 1.5 MW per assembly which has a potentially significant impact on core thermal hydraulics.

To address these concerns, the use of small quantities of fissile material in the blankets is discussed here. Several options such as various plutonium or uranium isotopic vectors are investigated in terms of impact on minor actinides transmutation performances. Impacts on fuel behavior, fuel cycle and core power redistribution are also investigated. In a first time, transmutation performances are analyzed and it is found that the down blending of 5 %vol of weapon-grade plutonium increases americium consumption by 50 % without significant impacts on core feedback coefficients such a sodium void worth. Introduction of a degraded plutonium isotopic vectors leads to improvements in transmutation rates ranging from 65 to 40 % depending on the amount added, with ^{235}U yielding intermediate results. The use of reprocessed uranium as support matrix for the blankets and the relative impacts on assembly dose rate are also characterized. Fuel cycle impacts are also limited in terms of target decay heat and neutron source due to a competition between an increase in curium production from higher fluence and a decrease in capture cross sections from a faster spectrum. The effect of fissile addition in the blankets on fuel temperature and core radial power profile is also investigated. Small power redistribution towards core periphery is observed with power variations ranging from 2 to 3 MW during irradiation. This redistribution could be smoothened by core fissile content adaptation in order to limit the increase in total plutonium inventory. The impact on plutonium breeding in the blankets is also investigated and it is found that using degraded plutonium to speed up the transmutation process has a positive impact on both the plutonium isotopic vector and proliferation by breeding ^{238}Pu and ^{239}Pu. Finally, this solution is compared to the alternate approach based on the addition of moderating material in the blankets such as ZrH_2 or MgO to increase transmutation performances.

I. INTRODUCTION

Minor actinides are three elements, namely neptunium, americium and curium which are produced by successive captures on uranium or plutonium isotopes in nuclear reactors. In the case of a closed fuel cycle, where the plutonium produced during irradiation is recovered and reused as fuel, these nuclei make up almost the entirety of the heavy nuclides that can be found in the waste. Due to their very long half lives compared to most of the fission products, these nuclei and their daughter are going to drive the long term radiotoxicity of the nuclear waste. Additionally, due to their high intrinsic decay heat, the will also be the main contributor to ultimate waste packages thermal load.

Minor actinides transmutation is the process of removing those nuclei from the final waste by submitting them to a neutron flux so that they can undergo fission and turned in to shorter-lived fission products. If a near complete removal from the waste can be achieved (a small fraction around 0.1 % being still discarded as waste due to losses during reprocessing), a reduction of the waste radiotoxicity by a factor 200 could be achieved [1] along with a 33 % reduction of the total volume to be excavated in a deep geological repository [2].

Considering the requirement for a closed fuel cycle and additional spectrum considerations regarding neutron spectrum, fast reactors are more suited to this task that thermal reactors, as it is discussed in [3]. Two main approaches can be identified for loading minor actinides into a fast reactor core [4].

In the so-called homogeneous approach, minor actinides are dispersed in the fuel and loaded directly inside the reactor core. In this configuration, minor actinides consumption is maximal due to the high flux level to which they are exposed; however, the presence of such nuclei in the core leads to a neutron spectrum hardening which degrades the core feedbacks coefficients. Consequently, transient behavior of the core is negatively impacted and additional safety measures must be considered (lower power, additional active safety systems etc.). This approach also exhibits the drawback of "polluting" the entire fuel cycle with minor actinides.

In the second approach, usually called the heterogeneous approach, minor actinides are loaded in dedicated targets located at the core periphery. As these targets, denominated "Minor Actinides Bearing Blankets" or MABB, are located in a relatively lower flux level zone, the impact on core feedback coefficients are smaller but the minor actinides consumption is reduced compared to the homogeneous case due to this lower flux. Consequently, minor actinides bearing blankets tend to be loaded with higher amount of minor actinides compared to the fuel and to stay longer in the core than standard core MOX fuel, typically twice the residence time of a fuel assembly, as described in [5] for instance. This work will focus here on addressing one of the drawbacks of the heterogeneous approach, to know the low flux level at the core periphery.

II. PROBLEM DESCRIPTION AND METHODOLOGY

One of the main drawbacks of heterogeneous minor actinides transmutation is the low flux level seen at the core periphery which yields comparatively lower performances than in the homogeneous approach. The basic physics of the transmutation process for ^{241}Am and ^{243}Am can be written as shown in Equation 1, with N the amount of americium in the blankets, considering that the production of americium by successive captures on uranium nuclei can be neglected and that no significant amount of heavier nuclei decay to the aforementioned americium isotopes. This equation can be directly solved and the transmutation rate τ, or the fraction of americium consumed during irradiation, can be defined as done in Equation 2, with T being the total irradiation time.

$$\frac{dN}{dt} = -N\phi\sigma_a \tag{1}$$

$$\tau = 1 - e^{-\sigma_a \phi T} \tag{2}$$

Considering Equation 2, it can be seen that various solutions are possible to increase the transmutation rate. The most straight forward one is to increase the irradiation time of the blankets assembly, typically by irradiating blankets assemblies twice longer than standard fuel, as it is considered for instance in [5]. It should be mentioned here that increasing the fluence generally leads to an increase in the activity of the irradiated blankets related to the enhanced curium production. However, above a given limit, accumulated curium starts to be consumed as well and the overall activity of the spent blankets starts to decrease. This is illustrated below in Figure 1 for the reduced ^{243}Am/^{244}Cm couple. However, this approach is mainly limited by cladding resistance to the high resulting dpa dose and overall mechanical behavior of the assembly and it may not be possible to reach the very long residence time necessary to overcome this so-called "curium peak".

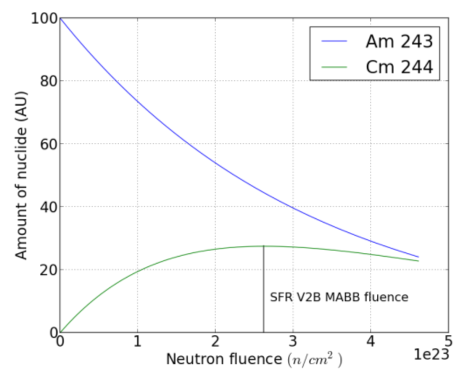


Figure 1 : Illustration of the so-called curium peak effect. The vertical line corresponds to the fluence in a 3600 MWth industrial reactor taken from [6]

Another option which has been extensively discussed in the past is to consider the introduction of a moderating material in the blankets in order to soften the neutron spectrum and increase the reaction rate, as it is discussed in [7] or [8]. The addition of hydrogenated moderating material in the blankets could increase the absorption cross sections of americium by a factor 4 due to the softer spectrum achieved this way. However, this approach exhibits several limitations:

- Self-shielding in the blankets will have a negative impact on the flux level "seen" by the blankets, and thus the actual increase in the total reaction rate is going to be limited to a factor 2.
- Addition of moderating material to the blankets is done either at the expense of sodium volume fraction, which requires an increase in the flow rate in order to extract the same amount of heat

from the assembly, or at the expense of fuel volume fraction. In this case, either the total amount of fuel in the blankets has to be decreased, which decreases the absolute transmutation performances, or the americium content in the fresh targets has to be increased, which has negative impact on manufacturing [9] and transportation of fresh targets.

- The most effective moderating materials considered so far are zirconium hydride ZrH_2 or yttrium hydride YH_2. Such materials exhibit the risk of hydrogen desorption in the case of a power transient in the core, which is not acceptable in terms of safety. Alternative material such as MgO or beryllium (under metallic form) have been considered, however their effectiveness as moderating material being lower, their performances are less interesting compared to hydrides.
- The increase in the absorption cross sections from moderation comes from an increase of the capture cross sections, which is inversely proportional to the neutron energy. Consequently, curium production is going to increase with moderation which will have negative impacts on fuel cycle parameters due to the high activity of curium isotopes.

A possible solution to overcome these limitations is to use once-through approach for blankets, which are not reprocessed at the end of irradiation, as discussed in [7] for instance. This option will not be further considered here.
The third option to increase the transmutation rate in the blankets is to increase the flux level to which they are submitted. To achieve this, they can be placed in inner positions in the core, as it is discussed in [5] for instance; however this is an intermediate solution between the homogeneous and heterogeneous approach and negatively impacts core feedback coefficients.
Another possibility, which will be discussed here, is to add fissile material to the target fuel. This exhibits several advantages, as it increases the flux level in the blankets while hardening the neutron spectrum, effectively increasing the absorption rate while limiting production of curium by capture and thus the increase on targets decay heat and neutron source. It also has a positive impact on the evolution of the power in the blankets during irradiation due to lower breeding gain. This power increase may also be beneficial for helium release in the pins free volume, thus limiting fuel swelling rate. However, an increase in the transmutation may lead to an increase in Helium production from alpha decay, which creates additional constraints in term of assembly design. Finally, it may provide a temporary solution to immobilize available fissile material and thus reduce proliferation risks [10] [11]. However, it may also negatively impacts core feedback coefficients and power distribution, which may require adaptation of core enrichment in order to address these issues. The increase in transmutation performances may also be offset by the production of minor actinides due to the fissile element used, depending on the choice.
Various fissile materials can be considered for such a use:

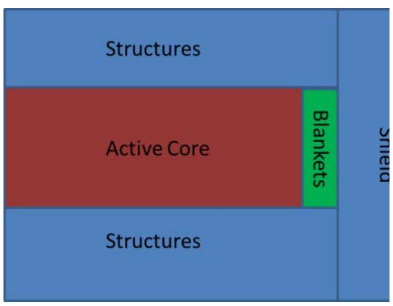
- ^{235}U, which is readily available as PWR fuel
- ^{239}Pu, which is available in a nearly pure form as weapon-grade material
- Any isotopic vector of plutonium coming from reprocessed fuel. Two vectors were considered in this work, namely the Pu2035 and Pu2100 which are deemed representative of the available plutonium stocks in France around years 2035 and 2100 [12] The first one has a lower amount of ^{239}Pu as it is coming from MOX fuel irradiated in PWRs while the second one is representative of the plutonium that can be obtained in a closed fuel cycle with sodium fast reactors. The actual composition of these two vectors can be found in **Erreur ! Source du renvoi introuvable.**.

Table 1: Composition of the considered isotopic vectors for plutonium

%	^{238}Pu	^{239}Pu	^{240}Pu	^{241}Pu	^{242}Pu	^{241}Am
Pu2035	3,57	47,39	29,66	8,23	10,37	0,78
Pu2100	0,61	62,89	30,46	2,54	3,05	0,45

III. ANALYSIS OF VARIOUS FISSILE MATERIALS

The performances of the various materials described before compared to a reference case without fissile material and with a case with ZrH2 as moderating material for completeness. The reference case is the same as considered in [7], e.g. a 3600 MWth sodium cooled fast reactor with an homogeneous core design denominated SFR V2b designed by EDF, Areva and CEA. The first inner ring of reflector is replaced by 84 targets assemblies with a 38.4 % volume fraction of fuel made of $(U_{0.8}Am_{0.2})O_2$. The americium isotopic vector considered is 75 % ^{241}Am and 25 % ^{243}Am. Fuel residence time for this core is 2050 EFPD, so a 4100 EFPD residence time was considered for the targets. All calculations were carried using a 2D-RZ model as shown below in Figure 2, except the calculations related to core power distributions which were done using a complete 3D model in diffusion approximation for trends estimation. The reference assembly design corresponding to the volume fraction aforementioned is given in Figure 2 . The ERANOS code system was used [13] with the JEFF 3.1 nuclear data library [14]. Depletion calculations were carried out using the DARWIN code system [15]. Various isotopic contents in fissile material were considered withcontent ranging from 1 to 5 at%. An irradiation time of 4100 EFPD was considered for the blankets in this analysis.



Dimension	Length (mm)
Inner flat to flat	197.3
Wrapper thickness	4.5
Sodium thickness between assemblies	4.5
Cladding thickness	0.5
Spacing wire thickness	1
Gap	0.15
Plenum height	989
Fissile column height	1000

Figure 2 : 2D-RZ representation of the SFR-V2B core with minor actinides bearing blankets and fuel assembly specifications

III.A. Impact on transmutation performances and plutonium consumption

Transmutation performances were evaluated by using the estimator defined in Equation 2 and the specific consumption, which is defined in Equation 3. P_{core} is the electrical output of the reactor and S_c is expressed in kg/TWeh. The transmutation rate is an estimator of the relative efficiency of the transmutation process while the specific consumption is a metric of the absolute efficiency of the process for a given situation.

$$S_c = \tau * \frac{m_{Am}(t=0)}{P_{core} * T} \qquad (3)$$

The addition of fissile material to the blankets has two impacts. Firstly, the increase in the flux level in the blankets increases the consumption in this region of the core. The combination of the two effects is shown below in Figure 3. Secondly, it decreases the production of minor actinides from the core itself due to power redistribution towards the outer part of the core and the blankets in which minor actinides consumption takes place. This effect is however rather limited with a maximal value around 3 % here for a case where 5% ^{239}Pu is used.

It can be observed Figure 3that the addition of 5 % of pure ^{239}Pu in the blanket assemblies, leads to a twofold increase the total specific consumption. This is due to the very good quality of this isotope as a fissile element. Using either ^{235}U or Pu2100 yields slightly lower results, with a 73 % increase in the total consumption. These two elements yields similar results, with the difference with ^{239}Pu explained either by the lower fission cross section of ^{235}U in a fast spectrum or the lower quality of the plutonium isotopic vector considered. Finally, it can be observed that Pu2035 case exhibits the lowest increase of the four materials compared, but still leads to a 55 % increase in the total specific consumption. This is explained by the lower quality of this plutonium isotopic vector, which leads to a smaller increase in the flux level and to a small amount to the production of minor actinides from the plutonium itself.

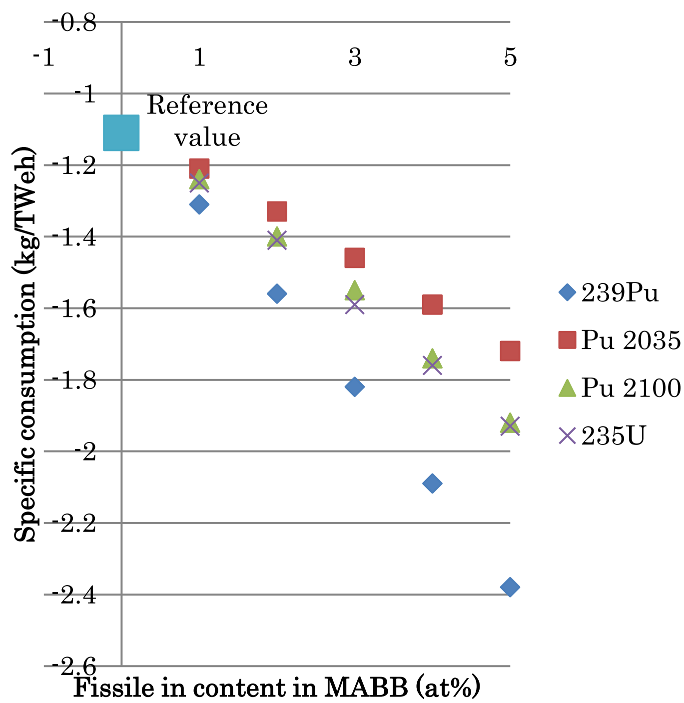


Figure 3 : Specific consumption of minor actinides in blankets vs content in fissile material for various fissile materials

The impact on plutonium inventory in the core and the blankets was also assessed, as it is shown in **Erreur ! Source du renvoi introuvable.**. It can be seen that the initial inventory in the core slightly decreases due to the higher fissile content in the blankets. The additional Pu mass required at loading here is at most 546 kg in the Pu 2035 case, or around 5% of the total core Pu content in the core. A similar impact is found at EOL (End Of Life). In the 235U case, additional 624 kg of ^{235}U is required at the beginning of irradiation. However, due to the lower breeding gain in the cases where plutonium is loaded in the blankets, the final increase in the plutonium core inventory is closer to 2.5 %.

Table 2 : Plutonium inventory in the core and blankets with 5 at% of various fissile in the blankets

	REF		^{239}Pu	
kg	BOL	EOL	BOL	EOL
Pu core	11922	11969	11786	11831
Pu blankets	0	1090	631	1557
Am core	94	339	93	331
Am blankets	2413	1461	2413	1277
Total pu	11922	13059	12418	13389
Total Am	2507	1800	2506	1608
	Pu2035		Pu2100	
kg	BOL	EOL	BOL	EOL
Pu core	11841	11884	11831	11871
Pu blankets	627	1564	629	1575
Am core	93	334	93	333
Am blankets	2418	1378	2416	1348
Total pu	12468	13448	12459	13446
Total Am	2511	1712	2509	1682
	^{235}U			
kg	BOL	EOL		
Pu core	11822	11877		
Pu blankets	0 + 624 ^{235}U	1148 + 290 ^{235}U		
Am core	93	333		
Am blankets	2413	1339		
Total Pu	12446	13314		
Total Am	2506	1672		

III.B. Impact on fuel cycle parameters

The impacts on fuel cycle parameters were assessed by considering:

- The time necessary to reach a target decay heat level of 7.5 kW, which is the expected limiting value for sodium washing of the assembly and subsequent handling in air or water [9].
- The neutron source output of the target at this specified time, as this value is an estimator of the necessary precautions required to handle the spent assemblies. For comparison purpose, the neutron source of a spent fuel assembly after 5 years of cooling, which the time currently considered in industrial scenarios [9] is 1.22 10^9 n/s.

It can first be observed that adding fissile material in the blankets increases the cooling time by up to 2.5 years when 5% of fissile is added. This increase can be divided by two contributions, the main one being from the increase in transmutation performances and thus production of heat emitting isotopes such as ^{244}Cm. The second contribution is directly due to the isotopes of plutonium added in the blankets and increase with the fissile content. This contribution can be easily evaluated by comparing the cooling time for the ^{235}U and Pu2100 cases, which have the same transmutation performances but a different cooling time. An interesting point to note here is that the cooling time associated with Pu2035 is as long as the one required for pure ^{239}Pu, even if the performances are much lower. This is explained by the increased production of ^{244}Cm due to successive captures on ^{242}Pu and then ^{243}Am. This phenomenon is less obvious for the Pu2100 case as the ^{242}Pu fraction is lower.

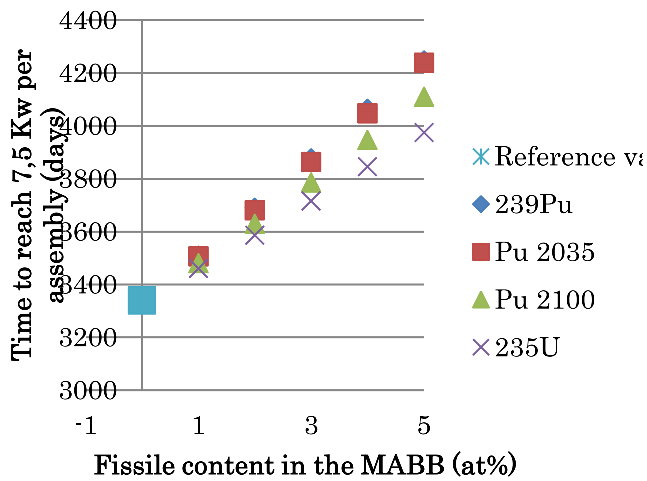


Figure 4 : Cooling time to reach 7.5 kW per assembly vs fissile content in the MABB

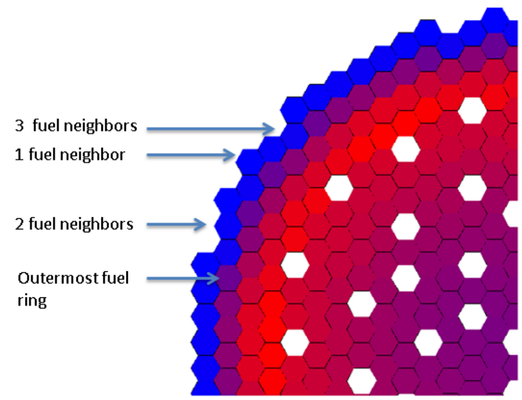


Figure 5 : Position of the various MABB assemblies considered

III.C. Impact on assembly design and power

3D calculations were carried out for cases loaded with 5 % of fissile material. Using 2D-RZ calculations, an assessment of the helium and gaseous fission products was done. It was verified that the reference design described in Figure 2 could be used for each calculation. Each case was evaluated with regards to the relative and absolute power variations during irradiation. In order to limit constraint on the upper part of the core structures and to smooth the temperature distribution in the upper sodium, it is necessary to limit the increase in power in order not to over cool the targets assemblies at the beginning of irradiation. The linear heat rate in the blankets was always below 200 W/cm.

The same values were given for comparison for the "hottest" and "coldest" assembly in the reference core. Depending on their positions in the core, target assemblies can be placed next to 1, 2 or 3 fuel assemblies and consequently exhibit important variations in power produced. As it is shown in Figure 5 for a third of core, one of each case was considered at each step.

Table 3 : Comparison of the power variations during irradiation for various fissile in 5 at% amount in the MABB

		Reference value	^{239}Pu	Pu2035
1 fuel neighbor	Power BOL (MW)	0,36	1,33	0,94
	Power EOL (MW)	0,94	2,26	1,74
	Flow BOL (kg/s)	4,95	11,90	9,16
	ΔT BOL (K)	57	88	81
2 fuel neighbors	Power BOL (MW)	0,62	1,91	1,41
	Power EOL (MW)	1,57	3,12	2,54
	Flow BOL (kg/s)	8,3	16,4	13,4
	ΔT BOL (K)	59	92	83
3 fuel neighbors	Power BOL (MW)	0,83	2,37	1,79
	Power EOL (MW)	2,08	3,88	3,22
	Flow BOL (kg/s)	11,0	20,4	17,0
	ΔT BOL (K)	60	92	83
		Pu2100	^{235}U	
1 fuel neighbor	Power BOL (MW)	0,99	1,24	
	Power EOL (MW)	1,84	1,99	
	Flow BOL (kg/s)	9,69	10,48	
	ΔT BOL (K)	81	93	

2 fuel neighbors	Power BOL (MW)	1,48	1,79
	Power EOL (MW)	2,66	2,8
	Flow BOL (kg/s)	14,0	14,7
	ΔT BOL (K)	83	96
3 fuel neighbors	Power BOL (MW)	1,87	2,25
	Power EOL (MW)	3,36	3,49
	Flow BOL (kg/s)	17,7	18,4
	ΔT BOL (K)	83	97

Using the classical formula $\dot{m}\, c_p \Delta T = \dot{Q}$, the mass flow required to achieve a 150 K temperature difference at EOL was computed. Considering this mass flow, the ΔT at BOL was computed using the same approach. Considering the design of the SuperPhénix reactor, it can be considered that the temperature difference between two neighboring fuels should be below 50 °C at anytime. As the power produced in the outermost ring of fuel assemblies decreases during irradiation, the mass flow per assembly is adjusted to achieve a ΔT of 150 K at BOL. As a consequence, it can be seen that this criterion is not fulfilled for any of the cases above mentioned.
It also appears from this analysis that 235U is the best fissile to limit the power variation in the blankets assemblies. The power variation observed in the reference case corresponds to the contribution of plutonium produced during the transmutation process and bred. Adding 235U increases the initial power without adding initial plutonium. This means that part of the plutonium production will compensate for the consumption of 235U, thus limiting the power increase. On the other hand, for the cases where plutonium is used, the power variation is explained by an important production of ^{238}Pu and ^{239}Pu which comes on top of the initial amount loaded. It is higher for the Pu2100 and 2035 cases as the initial amount of ^{239}Pu is comparatively lower.
It appears from this analysis that in order to limit the power variation in the blankets assembly, ^{235}U is the best fissile element, and that only 5.5 % are necessary to fulfill the constraint that the difference of temperature between two neighboring assembly should be below 50 K.

III.D. Impact on core power and feedbacks coefficients

The impact on core feedback coefficients was assessed by evaluating core Doppler and sodium void worth coefficient at the end of one representative core cycle and at the end of the blankets irradiation time. Core power distribution at the end of the blankets irradiation time was also studied and it was verified that no power inversion took place between the inner and outer core during irradiation. The ratio of power production in each zone of the core during irradiation was computed and it was checked that blankets did not generate more than 10 % of the core total production. Addition of fissile material in the blankets also leads to a flux and power redistribution which tends to a power decrease at the core center.
The maximal variation of sodium void worth was estimated as +0.05 $ and the total Doppler Effect (core + blankets) was not modified by the addition of fissile material in the blankets. This result is consistent with the low power level in the blankets and their position at the core periphery. Regarding power distribution in the core, two power profiles are shown below in Figure 6.

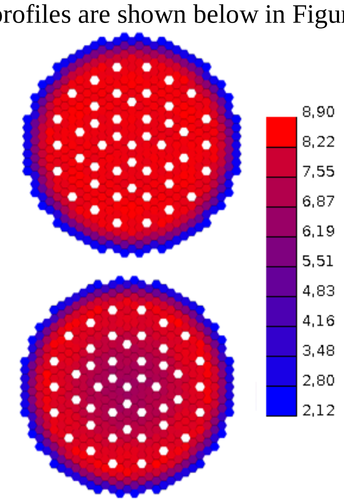


Figure 6 : Comparison of the power distribution in the core for the reference case (above) and a case with 5 % Pu9 in the blankets (below) at EOL

It can be observed that the addition of fissile material in the core leads to a small power redistribution towards the outer part of the core. However, the power in the peak assembly is not affected by this redistribution. The higher the amount and quality of fissile material, the higher the redistributed fraction of the power. Nevertheless, for the cases studied here, this fraction of redistributed power remains below 8 %.

III.F Comparison with the addition of moderating material

A short comparison can finally be done to compare the addition of 5 % of fissile material (here, ^{239}Pu) with the addition of moderating material. Two have been selected here: MgO, which is readily available and usable for sodium cooled reactors, and ZrH_2, which use may be more of an issue due to possible dissociation in case of accidental transients. 5 vol% of fuel in the assembly were replaced by the same amount of moderating material here.

As it can be seen below in Table 4, MgO is not efficient enough to compensate for the decrease in loaded fuel volume fraction. Consequently, its use will not be further considered here.. On the other hand, with ZrH2, it can be seen that the total Am consumption is close to the one of the fissile-loaded case. However, it can also be seen that the curium production in the blankets and in the core is 12 % higher than the reference configuration, which will have negative impacts on the cooling time and neutron source.
Indeed, the cooling time necessary to reach 7.5 kW per average sub-assembly for the ZrH2 moderated case is close to 6300 days, that is more than 5 years longer than the case with fissile material, for a similar specific consumption of americium. This is explained by the degraded spectrum in the moderated blankets which leads to a higher capture rate on americium isotopes.

Table 4 : Comparison of the performances of two moderated cases with 5 vol% of moderator and one with 5 at% of 239Pu.

		5 at% ^{239}Pu	5 vol% MgO	5 vol% ZrH2
Core	Np	0,54	0,55	0,54
	Am	2,66	2,72	2,76
	Cm	0,69	0,74	0,73
Total Core		3,89	4,01	4,03
Blankets	Np	0,14	0,13	0,1
	Am	-7,91	-6,11	-8,01
	Cm	1,5	1,24	1,72
Total	Np	0,68	0,68	0,64
	Am	-5,25	-3,39	-5,25
	Cm	2,19	1,98	2,45
Total		-2,38	-0,73	-2,16

In terms of inventory, as it is shown in Table 5, the final inventories in plutonium are similar while the moderated case exhibits a lower total americium mass due to the lower initial mass in the blankets. Overall, the two cases are similar as the increase in the cooling time for the moderated case is compensated by the decrease in the loaded mass.

Table 5 : Comparison of the inventories at BOL and EOL for the moderated and fissile loaded cases

	^{239}Pu		ZrH2	
kg	BOL	EOL	BOL	EOL
Pu core	11786	11831	12037	12127
Pu blankets	631	1557	0	1256
Am core	93	331	94	344
Am blankets	2413	1277	2100	957
Total pu	12418	13389	12037	13383
Total Am	2506	1608	2194	1301

Finally, in terms of thermal-hydraulics, it appears that the use of moderating material in the blankets leads to a higher power production at the beginning of irradiation compared to a fissile-loaded approach with similar increase in the power production at EOL, which leads to a smaller undercooling at BOL and thus a lower ΔT with the neighboring assemblies. This can be seen below in Table 6.

Table 6 : Comparison of power evolution in the blankets between a fissile-loaded and moderated case

		^{239}Pu	ZrH2
1 fuel neighbor	Power BOL (MW)	1,33	1,37
	Power EOL (MW)	2,26	2,16
	Debit EOL (kg/s)	11,90	11,38
	ΔT BOL (K)	88	95
2 fuel neighbors	Power BOL (MW)	1,91	2,04
	Power EOL (MW)	3,12	3,08
	Debit EOL (kg/s)	16,4	16,2
	ΔT BOL (K)	92	99
3 fuel neighbors	Power BOL (MW)	2,37	2,5
	Power EOL (MW)	3,88	3,78
	Debit EOL (kg/s)	20,4	19,9
	ΔT BOL (K)	92	99

Overall, it appears that the addition of 5 at% Pu 239 is similar to the addition of 5 vol% of ZrH2 in terms of minor actinides consumption, and that the higher fuel cycle impacts of the moderated cases are counterbalanced by the lower inventory. The moderated case exhibits a slightly better behavior in terms of thermal hydraulics behavior.
It is also possible to compare the approach using ZrH2 with the fissile-loaded approach at equal performances. The reference taken is the consumption of americium in regular blankets which is equal to -6.61 kg/TWeh. The results are shown below in Table 7. The estimated inventory is calculated using Eq. (4). It is an approximation which takes into account the cooling time to 7.5 kW and a 2 years manufacturing time to evaluate the total inventory in the fuel cycle. It allows easy comparison between two equivalent strategies in terms of performances.

$$I = Am(BOL) + Am(EOL) * \frac{(Cooling\ time + 2years)}{Irradiation\ time} \quad (4)$$

Table 7: Comparison of the moderated and fissile loaded cases at similar performances

kg/TWeh	Reference value		5 at % 239Pu	5 vol% ZrH2
Blankets	Np	0,15	0,12	0,10
	Am	-6,61	-6,61	-6,61
	Cm	1,34	1,26	1,39
Cooling time to 7,5 kW (days)		3339	2193	3907
Am at BOL (kg)		2507	2060	1984
Am at EOL (kg)		1800	1351	1293
Estimated fuel cycle inventory (kg)		3863	2695	2967

It can be observed in Table 7 that, at equal consumption, the cooling time associated with the fissile-loaded approach is reduced due to the faster spectrum and the lower curium production compared to a moderated approach. As the initial loaded mass and consumption are very close, the final inventory in the fuel cycle is lower in the fissile loaded case as the irradiated targets are less active.
It can be concluded from this that, for the performances considered, using fissile material is a better option in terms of fuel cycle impacts than using moderated material. The extension of this work to additional cases is planned.

III.E. General Overview

An interesting way to visualize the various impacts of the addition of fissile material in the minor actinides bearing blankets is to use radar plots. This is done in Figure 7 where five estimators have been selected:

- Transmutation performances, which are expressed compared to the reference case
- Neutron source after 5 years of cooling, which is a measure of the shielding required for handling the assembly
- Cooling time compared to a regular assembly
- Americium inventory in the reactor (core + blankets) at the end of irradiation
- Temperature difference with the neighboring assemblies

This visualization allows the reader to compare effectively the various approaches and to observe the opposition between the fissile-loaded approach and the moderated approach. The former increases the transmutation performances without modifying the assembly design while only slightly impacting the neutron source and decay heat. The latter increases the transmutation performances while strongly increasing the decay heat and neutron source, but compensate this by a lower inventory in the blankets and thus during cooling. It also allows the reader to compare the various fissile materials available for such approach, and notably to verify that no material is the best for all the considered parameters.

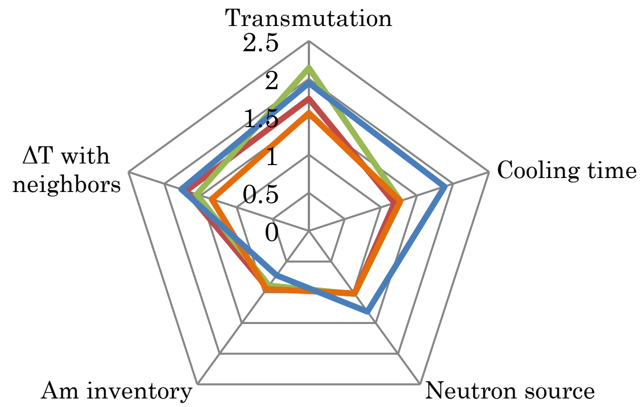


Figure 7 : Radar plot showing the various impacts of the fissile material studied here. The red curve corresponds to 235U, the green one to Pu239, the orange one to Pu 2035 and the blue one to the moderated case using ZrH_2.

IV. CONCLUSION AND PERSPECTIVES

The loading of fissile materials in heterogeneous targets has been studied here at a solution to compensate for the low transmutation kinetics at the core periphery. Various fissile material have been studied, notably plutonium 239 and uranium 235. It was found that adding up to 5 % of fissile material in the blankets increased significantly the transmutation efficiency with limited impacts on the core behavior or fuel cycle evolution. Uranium 235 was found to be the optimal nuclide regarding thermal-hydraulic behavior of the core while ^{239}Pu yielded the best results in terms of transmutation performances. This approach was also compared to the opposite approach of moderated targets, either with MgO or ZrH_2. It was found that it was superior to the former as adding moderating material decreases the fuel volume fraction in the target and the added moderation is not enough to compensate for the decrease in reaction rate. The results concerning the use of ZrH_2 were found to be similar to the one involving 5 at% of ^{239}Pu. It was shown that, for a level of performances similar to the reference case (no fissile, no moderated material), the fissile approach yielded better results. As ZrH_2 addition in the targets has not been technologically demonstrated so far, the use of ^{239}Pu or other fissile material appears as a potentially interesting solution to improve the transmutation process using readily available technology.

In a second time, it is planned to compare those two approaches more in depth, notably at the scenario level in order to evaluate the impact on the overall fuel cycle of the additional requirements in fissile material.

REFERENCES

[1] M. Salvatores, "Nuclear fuel cycle strategies including partitioning and transmutation," *Nuclear Engineering and Design,* no. 235, pp. 805-816, 2005.

[2] C. Chabert, C. Coquelet-Pascal, A. Saturnin, G. Mathonniere, B. Boullis, D. Warin, L. V. D. Durpel, M. Caron-Charles and C. Garzenne, "Technical and economic assessment of different options for minor actinides transmutation : the French case," in *GLOBAL2011*, Tokyo, 2011.

[3] NEA, "Minor actinides burning in thermal reactors," Paris, 2013.

[4] NEA, "Homogeneous versus heterogeneous recycling of transuranics in fast nuclear reactors," NEA, Paris, 2012.

[5] L. Buiron and e. al, "Heterogeneous minor actinides transmutation on a UO2 blanket and on (U,Pu)O2 fuel in sodium-cooled fast reactor. Assessment of core performances," in *GLOBAL*, Paris, 2009.

[6] P.Sciora, L.Buiron, G.Rimpault and F.Varaine, "A break even oxide fuel core for an innovative French sodium-cooled fast reactor : neutronic studies results," in *GLOBAL*, Paris, 2009.

[7] C. De Saint Jean, "Americium once-through of moderated targets in a CAPRA core," in *Seminar Int. CAPRA conf*, Karlsruhe, 1998.

[8] C. de Saint Jean, J. Tommasi, F. Varaine, N. Schmidt and D. Plancq, "Americium and curium heterogeneous transmutation in moderated S/ A in the framework of CNE scenarios studies," in *GLOBAL2001*, Paris, 2001.

[9] C.Chabert and a. et, "CONSIDERATIONS ON INDUSTRIAL FEASIBILITY OF SCENARIOS WITH THE PROGRESSIVE DEPLOYMENT OF PU MULTIRECYCLING IN SFRs IN THE FRENCH NUCLEAR POWER FLEET," in *GLOBAL2015*, Paris, 2015.

[10] M. Saito, "Advanced core concepts with enhanced proliferation resistance by transmutation of minor actinides," in *GLOBAL05*, Tsukuba, Ibaraki (Japan), 2005.

[11] H. SAGARA, M. SAITO, Y. PERYOGA, A. EZOUBTCHENKO and A. TAKIVAYEV, "Denaturing of plutonium by transmutation of minor actinides for enhancement of proliferation resistance," *Journal of nuclear science and technology,* vol. 42, no. 2, pp. 161-168, 2005.

[12] C. Coquelet-Pascal, M. Meyer, R. Girieud, M. Tiphine, R. Eschbach, C. Chabert, C. Garzenne, P. Barbrault, B. Gannaz, L. V. D. Durpel, D. Favet, M. Caron-Charles, B. Carlier and J.-C. Lefèvre, "Scenarios for Fast Reactors Deployment with Plutonium Recycling," in *Fast Reactors*, Paris, 2013.

[13] G. Rimpault, "The ERANOS code and data system for fast reactor neutronic analyses," in *PHYSOR*, Seoul, 2002.

[14] NEA/OECD, "The JEFF-3.1 Nuclear Data Library," NEA, Paris, 2006.

[15] A. Tsilanizara, C. Diop, B. Nimal, M. Detoc, L.Luneville, M. Chiron, T. Huynh, I. Bresard, M. Eid, J. Klein, B. Roque, P. Marimbeau, C. Garzenne, J. Parize and C. Vergne, "DARWIN: An Evolution Code System for a Large Range," *Journal of Nuclear Science and Technology*, vol. Supplement 1, pp. 845-849, 2000.