APA: U FREE PU PIN IN A HETEROGENEOUS ASSEMBLY TO IMPROVE PU LOADING IN A PWR — NEUTRONIC, THERMO-HYDRAULIC AND MANUFACTURING STUDIES

J. PORTA
Centre d'études nucléaires de Cadarache,
Saint-Paul-lez-Durance

A. PUILL
Centre d'études nucléaires de Saclay,
Gif sur Yvette

M. BAUER, P. MATHERON
Centre d'études nucléaires de Cadarache,
Saint-Paul-lez-Durance

France

Abstract

After having presented the specific context of France with respect to the fuel cycle and reprocessing, the problem of plutonium fuel utilization is posed.
If one of the solutions, a pressurized water reactor (PWR) with an increased moderation ratio seems possible, it entails making excessive changes to the reactor, the control systems, and the general architecture of the steam supply system.
Another solution consists in modifying the fuel itself so as to eliminate conversion on ^{238}U by using plutonium (Pu) in a neutronically inert matrix. However, the disadvantage of this type of fuel is that it has very low Doppler and draining coefficients and a very small delayed neutron fraction. To enable using these fuels, a heterogeneous assembly has to be defined, in which standard UO_2 rods provide the physical properties required to ensure acceptable safety coefficients.

Introduction

In view of the many changes that have occurred in the world nuclear landscape, the problem of managing our stocks of plutonium and irradiated fuels has become acute.

The following notions have been highlighted by the international trends expressed in numerous forums, particularly at the IAEA:

- Resistance to proliferation,
- Preservation of natural resources,
- Minimization of the volumes, quantities, and radiotoxicity of wastes arising from the nuclear industry.

All these points are underpinned by the crucial concept of public acceptance.

It is obvious, however, that a nuclear industry can only continue to exist if it produces a kWh that is competitive compared to other sources of energy.

In France, a plutonium reprocessing and recycling policy was set up very early to prepare fuel for the future breeder reactor system. However, a certain number of socio-economic factors led first to the checking of the development of the breeder system, then to its being placed in abeyance.

The reprocessing facilities that were constructed permit producing a high-quality mixed uranium-plutonium fuel (MOX), and were therefore oriented towards the production of a fuel intended for the cores of PWRs.

However, using plutonium in a conventional core poses numerous problems, such as, in particular, the great affinity of Pu isotopes for thermal neutrons, which results in a hardening of the neutron spectrum and considerably decreases the efficiency of control systems.

In addition, the moderator coefficient becomes very negative and this, in certain classes of accidents, such as cold hazards, steam line break at full power, or spurious secondary valve opening on shutdown at hot zero power, leads to reactivity consequences that are difficult to control. This is why the safety authorities have limited core loading of MOX fuel to 30 %.

Twenty reactors have been granted authorization to use MOX fuel, although only 17 currently do so, and this number will soon be increased to 28. However, given the fact that the French utility, EdF, has reprocessed only the amount of irradiated fuel needed for the fabrication of the MOX fuel authorized in these reactors, it is obvious that the stocks of plutonium and non-reprocessed fuels will continue to increase.

Various solutions have been proposed, notably highly moderated reactors: these would permit better thermalization of the neutrons, resulting in increased control system efficiency and a considerable decrease in the absolute value of the moderator coefficient, and would therefore allow 100 % MOX loading (1, 2).

However, this solution would entail making significant changes to the architecture of the core and control systems, and would lead to a considerable reduction of the power density of the core.

Another solution would mean modifying the fuel and using plutonium or MOX combined with neutronically inert material, so as to consume the plutonium while reducing or eliminating ^{238}U to ^{239}Pu conversion (3).

Unfortunately, it was demonstrated that homogeneous assemblies using this fuel made the core uncontrollable, because of a small βeff, a very low Doppler coefficient and, occasionally, a draining coefficient liable to be positive. The solution would be to supply the complement to these safety parameters by means of a heterogeneous load in which standard UO_2 fuel would provide the kinetic coefficients required for safety. This is the APA or Advanced Pu fuel Assembly (4,5,6).

The APA Assembly

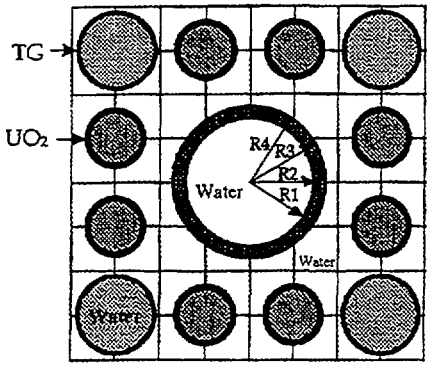
Externally, the APA assembly is strictly identical to a conventional assembly and can replace a standard assembly in a load.

It comprises 120 standard UO_2 rods, 24 guide tubes, one instrumentation tube and 36 APA rods (Figure 1). The latter are big, very thin annular rods, 25.2 mm in diameter and 1.26 mm

thick, with inner and outer cladding to permit inside circulation of water. The material used is a 32.6 % PuO_2, 63.9 % cerium (CeO_2), and 3.5 % Er_2O_3 ceramic-ceramic composite (CERCER) (Table 1).

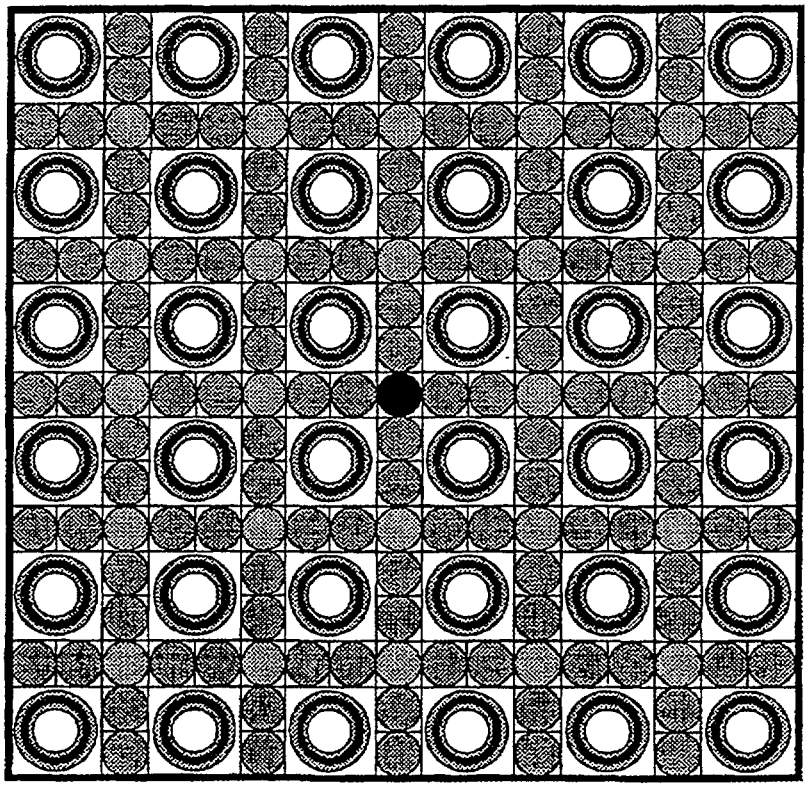
The advantage of the slight thickness of the ceramic is to increase the "apparent" thermal conductivity of the rod.

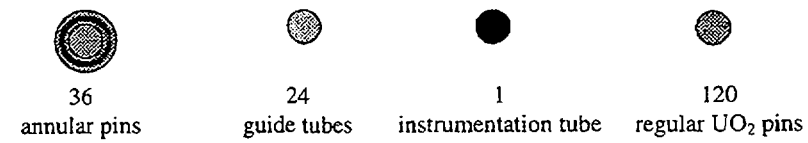
In addition, the inner water provides both an increase in the local moderating ratio (~ 6) and significant internal cooling.



R4 (mm)	=	11.5	
R3 (mm)	=	11.0	} 1.261 mm
R2 (mm)	=	9.739	
R1 (mm)	=	9.239	
$VpuO_2/Vcomb$	=	0.23	

Local moderation ratio = 5.96 (annular pin)
Global moderation ratio = 3.47





36	24	1	120
annular pins	guide tubes	instrumentation tube	regular UO_2 pins

Figure 1 - The APA assembly

Table 1 : fuel composition

UO_2 d = 10.21 g/cm^3 120 standard rods						
Uranium Percent mass	^{235}U 2.2	^{238}U 97.8	Fuel assembly $\frac{Pu+^{241}Am}{U+Pu+^{241}Am} = 0.1037$			
APA d = 7.85 g/cm3 36 annular rods						
Material Percent mass	PuO_2 32.6	CeO_2 63.9	Er_2O_3 3.5			
Plutonium Percent mass	^{238}Pu 3.3	^{239}Pu 41.7	^{240}Pu 28.7	^{241}Pu 14.5	^{242}Pu 10.7	^{241}Am 1.1
Cerium Percent mass	^{140}Ce 88.8	^{142}Ce 11.2				
Erbium Percent mass	^{162}Er 0.1	^{164}Er 1.5	^{166}Er 22.9	^{167}Er 22.9	^{168}Er 27.2	^{170}Er 15.1

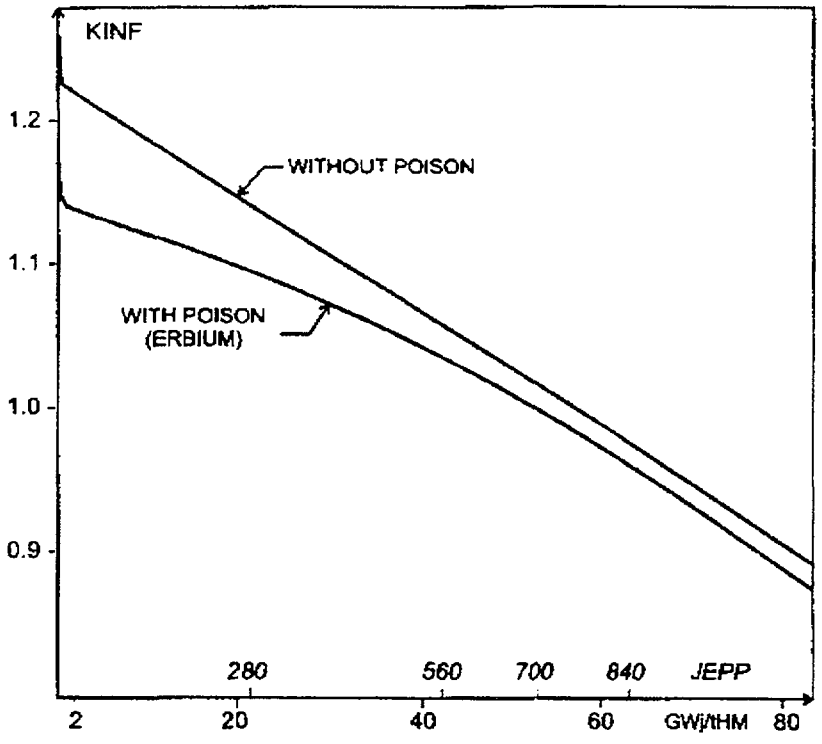


Figure 2 : reactivity swing with and without poison

Figure 2 shows evolution calculations for constant boron (600 ppm) made using the APOLLO 2 computer code. A considerable increase in erbium reactivity control can be observed, even if the residual penalty remains significant*. The Xe effect at saturation is about 1760 percent milli k (pcm) (10^{-5} $\Delta k/k$).

* This is not a major economic factor insofar as the cost of Pu is already included in the cost of uranium reprocessing.

Table 2 : averaged and maximum integrated power levels

	BOL	EOL 1120 EFPD
Annular rod (36)		
Linear power density (W/cm)		
Average	1010	664
Maximum	1084	681
Power density (W/cm^3)		
Average	1230	809
Maximum	1320	829
Surface heat flux (W/cm^2)		
Avearge	77	51
Maximum	83	52
UO_2 standard rods (120)		
Linear power density (W/cm)		
Average	87	191
Maximum	91	196
Power density (W/cm^3)		
Average	162	356
Maximum	170	366
Surface heat flux (W/cm^2)		
Average	33	76
Maximum	35	76

Temperature Assessment

As the FLICA IV code was modified to take internal and external cooling of the rod into account, it can be observed (Table 2) that power densities are very high in the APA rods, but that, as a result of their extensive contact surface, the temperatures of both the fuel and the cladding remain very cold (Table 3).

Table 3 : Temperature distribution - power density 1320W/cm^3

	Internal Clad		Fuel			External clad		
Internal Water	Coolant Side	Fuel Side	Internal Rod Side	Maximum	External Rod Side	Fuel Side	Coolant Side	External Water
$T_{W,i}$	$T_{C,1}$	$T_{C,2}$	$T_{F,2}$	T_{FM}	$T_{F,3}$	$T_{C,3}$	$T_{C,4}$	$T_{W,e}$
299°C	322°C	345°C	470°C	544°C	513°C	363°C	337°C	309°C

Table 4 : Heavy Nuclei Balance (kg) for 1 TWh

	UO2 (3.7%)	APA
235U	-84.74	-22.18
238U	-94.41	-46.94
Total plutonium	**+32.67**	-76.52
Minor Actinides	**+2.73**	**+9.04**
Plutonium + Americium	**+35.40**	**-67.40**
SWU	14816	3054

Material Aspects

The 120 UO_2 rods are enriched 2.2 % with ^{235}U. The APA rods contain plutonium known as 2G, which comes from MOX reprocessing in PWRs, discharged at 33 GWd/t, cooled three years and stored two years after reprocessing for a cycle of four times 280 EFPD in a 900 MWe PWR. Plutonium consumption is 48.6 % with about 80 % consumption for ^{239}Pu and 36 % consumption for ^{241}Pu, offset, however, by the production of + 36% of ^{242}Pu. Table 4 shows for an equivalent produced energy, the mass balances of a PWR using UO_2 fuel and one using APA fuel.

Kinetic Coefficients

Table 5 also gives a comparison of the kinetic coefficients of conventional loading and APA loading.

The role of plutonium can be clearly observed to affect the efficiency of the boron, the moderator coefficient and the βeff. However, the values obtained remain within an acceptable range.

Fabrication Materials

The first fabrication tests using classic powder metallurgy processes gave excellent results and rings meeting specifications were manufactured in the laboratory.

However, the slight thickness involved required very precise fabrication in view of the fact that grinding on 1.2 mm thick rings is extremely costly and results in numerous rejections. Vibrocompacting techniques should be able to be implemented to improve fabrication (7, 8).

Possible Scenario

Considering the situation in France after the year 2000, twenty-eight (63 GWe) PWRs, recycling 30 % 1G MOX (4 batches), 12 MOX and 18 UO_2, and four 1450 MWe APWRs using APA assemblies would be enough to absorb the second generation of Pu (2G). These new reactors (APWRs) would replace 6 decomisionned 900 MWe.

Table 5 : Comparison of the physical parameters

Fuel assembly parameters	APA	UO_2 (3.7%) PWR
Boron worth (pcm/ppm)		
BOL, full power, xenon = 0, C_B = 550 ppm	- 4.9	- 8.9
C_B = 1450 ppm	- 4.7	
Moderator temperature coefficient (pcm/°C)		
BOL, T = 296°C, C_B = 1500 ppm	- 17.3	+ 6.4
C_B = 600 ppm	- 31.9	- 15.4
EOL, T = 296°C, C_B = 10 ppm	- 36.1	
Doppler coefficient (pcm/°C)		
BOL, 650 to 286°C, C_B = 1500 ppm	- 2.0	- 2.6
Overall draining (pcm)		
BOL, water at 1500 ppm to without water	- 13 630	- 70 000
Core parameters		
Moderator temperature coefficient (pcm/°C)		
BOL, 1450 ppm	- 24.9	- 21.0
EOL, 10 ppm	- 57.8	- 62.0
Effective β (pcm)		
BOL	370	595
EOL	400	522

Table 6 : Medium Term operating Scenario for the French Reactors

Type of PWR [Net Power MW (electric)]	Fuel	Number of Units	Type of Management (Annual Cycles)	Enricment with ^{235}U [Total Pu Content (%)]	Tonnage in HM/yr unit (t-1) (Number of Fuel Assemblies)	Unloading Burnup (GWd/tonne HM)
APWR (1450)	APA	4	1/4	1.0 [10.4%]	14.763 t U - 1.708 t Pu (60)	72
CPY (920)	UO_2-MOX	28	1/4	3.7 [6.5%]	12.923 t U - 5.538 t UPu (28) (12)	43
P4 (1300)	UO_2	20	1/3	3.1%	34.387 tonne U (64)	33
N4 (1450)	UO_2	4	1/3	3.25%	36.525 tonne U (68)	33

Table 7 : Annual balance of HM in the reactors

Loading	Plutonium Production
1. Uranium metal: 1254.7 tonne 2. U-Pu metal: 155.1 or10.08 tonne of first generation Pu at 70% of fissile Pu (176 tonne of MOX) 3. Plutonium metal: 6.83 tonne of second generation at 55% of fissile Pu	1. 12.73 tonne of first generation 2. 7.05 tonne of second generation 3. 3.00 tonne of "third" generation
Theorically, when in equilibrium, there remains (no losses): 12.73 - 10.08 = 2.65 tonne of plutonium of first generation (70% fissile Pu) 7.05 - 6.83 = 0.22 tonne of plutonium of second generation (55% fissile Pu) ⇒ Total mixing 5.87 tonne 3.00 tonne of plutonium of "third" generation (70% fissile Pu)	
(48% fissile Pu)	

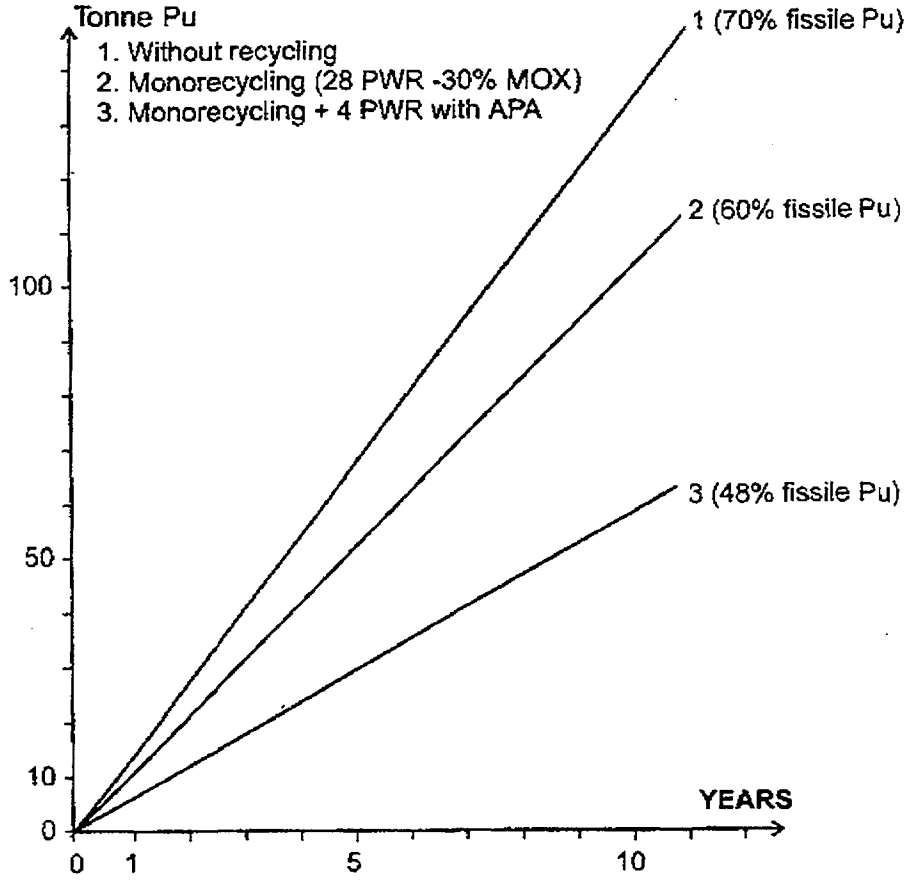


Figure 3 : Change of the Pu inventory (63 GWe)

The results of this scenario are reported in Table 6. It should be noted that the Pu from APA cycling is very degraded (28 % fissile Pu). The combination of these three restricted qualities (Table 7) results in 48 % fissile Pu. This quality can be used in another APA cycle provided 235U enrichment is increased to something up to 2.5 % in the UO_2 rods (instead of 2.2 % in the example selected here);

The results of the various scenarios, once-through, single cycling, single cycling + APA are reported in Figure 3. The advantage of an APA solution is apparent.

Conclusion

Obviously, a great many studies have yet to be performed in all domains -mechanics, neutronics, thermal-hydraulics, safety, etc.- to enable completely defining an industrial assembly. However, the concept presented here has the generic advantage of featuring a significant degree of freedom with respect to uranium and plutonium for adjusting control and kinetic coefficients to acceptable values, while preserving high plutonium utilization potential. This type of solution, possible in the relatively short term, could, in addition, provide end of cycle Pu for CAPRA type reactors.

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