STATUS OF FRENCH R&D FOR ADVANCED LIGHT WATER REACTORS

J.L. NIGON
Centre d'études nucléaires de Cadarache,
Saint-Paul-lez-Durance, France

I. - INTRODUCTION

Presents PWRs lead to a significant reduction of electricity cost when compared to other sources. Then it seems reasonnable to keep the main features of PWRs when looking to improvements of investment cost, of operating and fuel costs, of flexibility and of safety.

Besides that we have to think about uranium conservation ; if nuclear starts again in many countries, -as we hope-, uranium market could get into crisis during the first half ot the 21st century, and uranium shortage could become a reality. Advanced PWRs are also aimed at fissile material saving.

Let us recall the french situation :

- Fuel reprocessing is decided ; it is presently in industrial operation, the capacity is 400 T/year ; two additionnal facilities are under construction which will lead to a total capacity of 2000 T/year ;

- Plutonium recycling in PWRs is decided ; it will start next year, in a mixed loading mode (30 % UO_2-PuO_2 - 70 % UO_2) ;

- The installed power of LMFBRs will remain low for some tens of years ;

- The total plutonium inventory allows recycling in water reactors before it is required for LMFBRs.

The three french partners CEA, EdF and FRAMATOME decided to lead a three year programme 1984-1987. FRAMATOME in fact had started a little bit earlier, namely in 1982 (first publication IN RNG -a french technical journal) and developed the RCVS, spectral shift convertible reactor core (for both Uranium and Plutonium fuel). The FRAMATOME's effort is estimated to about 40.10^6 FF per year.

CEA performs an R&D programme, the objectives of which are :

- to support FRAMATOME for RCVS design,
- to explore a wider range of parameters in order to estimate the feasibility and the interest of tight lattice PWR cores.

EdF-R&D Division - is associated to this feasibility study.

Simultaneously, EdF is defining the preliminary specifications of "REP 2000" (future standard for french PWRs in the year 2000 and following) ; the objectives of REP 2000 are :

- load follow capacity,
- cost effectiveness,
- operation flexibility.

- The FRAMATOME's RCVS and the CEA RSM feasibility study have to be considered in this context. The main objectives are

1) To improve performances, safety and to minimise the cost,

2) To save fissile materials according to a global strategy,

3) while minimum modifications of present PWR components will be accepted.

The R&D budget for PWRs (outside safety) of French CEA is around 450 10^6FF per year. Among this, 40 10^6FF/year are devoted to tight lattice core feasibility studies (period 1984 - 1987).

II. - MAJOR STRATEGIC AND ECONOMICAL CONCERNS OF RSM (UMPWR)

II.1. : RSM are compatible with the present french PWR components, except, of course, for the vessel head and vessel internals, core instrumentation and control system.

It coulf also be interesting to perform minor modifications of RCPPs, of accumulators and MPIS, and of pressurizer.

II.2. : The higher density of the core leads to either a higher volumetric specific power and a smaller core, or a lower mean linear heat flux, or both. This allows the designer to decrease the operating fuel temperature, then the initial conditions of an hypothetical accident are less severe.

If the core is slightly smaller, axial and radial blankets can be installed.

In addition to that, some specific options are proposed in order to improve safety :

- higher vessel, (integrated accumulators),
- upper plenum injection,
- new cladding material,
- suppression of soluble boron.

II.3. : Plutonium recycling will be easier in tight (or semi-tight) lattice core :

- uniform loading of fuel assemblies relaxes the range of Pu isotopic content which can be accepted,
- plutonium inventory is only slightly degraded (FIR : Fissile Inventory Ratio is in the range 0.8, in core without blankets, to 0.95 in core with blankets), and can be compared to the "natural" degradation during storage (resulting from Pu 241 decay),
- multiple recycling are realistic,
- finally, 100 % PuO_2-UO_2 assembly cores reduce the total number of Pu fueled plants, thus reducing the administrative problems.

II.4. : Introduction of Pu fueled RSM at the rate of 1000 MWe/Year from 2000 to 2040 would lead to 160 000 tons of natural uranium savings (in 2040) -(among a total of 790 000 tons)-, and to about 10 % savings of the total fuel cycle expenses till 2040 (about 200 10^9FF out of 2000 10^9FF).

II.5.: Basic data for fuel cycle cost evaluation are following :

- Natural uranium (supply & conversion)		700 F/Kg(in 1995)
- Enrichment		850 F/UTS
- Fabrication	UO_2	1 600 F/Kg
	MOX (3 times)	4 800 F/Kg
- Reprocessing	UO_2	6 260 F/Kg
	MOX (1.3. times)	8 100 F/Kg.

Note that, according to some experts, fabrication cost of MOX could reach 4 times the cost of UO_2, and reprocessing could be increased by 10 %.

Actualization cost is 8 % per year.

No value is attributed to Plutonium.

The average load is supposed to be 5100 h/y.

The out of core cycle length is 5 years (2 years before loading, 3 years after irradiation).

The reference reactor is a 1400 electrical MW RSM without blankets (the FIR is 0.8) using reprocessed uranium support.

The french reference PWR is a 45 000 MWd/t, UO_2 fueled (4 x 1/4), 1400 MWe, N4 type reactor

	REP UO_2 45 GWd/T 4 x 1/4	RCVS UO_2 45 GWd/T 4 x 1/4	RSM MOX 45 GWd/T 4 x 1/4	RSM MOX 55 GWd/T 5 x 1/5
Nat. U	1.91	1.66	-	-
Enrichment	1.50	1.21	-	-
Fabrication	0.63	0.66	2.13	1.87
Reprocessing	1.50	1.52	1.87	1.53
TOTAL	5.54	5.05	4.00	3.40
Equilibrium cycle cost	5.00	4.61	3.48	2.85

Fuel cycle cost in CF / KWh for 30 year lifetime and for equilibrium cycle.

According to the discharge burn-up (45 to 60 GWd/T), the total saving represents 110 to 150 MFF/y for a 1400 MWe plant.

III. - RANGE OF FEASIBILITY OF TIGHT LATTICE CORES

III.1. - Core Physics and Fuel Cycle

III.1.1. - Reduction of Uncertainties

In tight lattice cores, epithermal neutrons and resonances are of *major importance. Both basic data -microscopic cross sections- and core physics* methods had to be improved in order to reach better results in terms of

- multiplication factor,
- conversion ratio,
- void reactivity effect,
- control material absorption rate,
- isotopic composition evolution,
- fission product capture rate,
- local power distribution (water holes, RCC rods...)
- spectral distribution at fissile fertile interface.

Improvements in computer codes, in basic data library using the results of an important experimental programme lead to following results:

COMPARISON OF UNCERTAINTIES EVALUATED

IN 1983 AND IN 1987 FOR RSM

	1983	1987
multiplication factor (k)	1 ± 5000.10^{-5}	1.005 ± 300.10^{-5}
U_{238} capture rate (%)	± 15 %	+ 2 % ± 2 %
Void effect ($\frac{\delta k}{k}$)	$\pm 4000.10^{-5}$	$\pm 1000.10^{-5}$
F.P. capture rate (%)	?	± 10 %

III.1.2. - Experimental programme

a/ **ERASME** : large experiment (up to 1500 PuO_2 fuel rods) in the EOLE critical facility in order to determine important core parameters :

- multiplication factor,
- conversion ratio
- absorption rate of several RCC materials,
- power distribution with RCC and water holes, in three typical kexagonal grid :

ERASME.S Vm/Vu = 0.5 (equivalent to 0.6 in hot water)

ERASME.R Vm/Vu = 0.9 (equivalent to 1.1.)

ERASME.L Vm/Vu = 1.8 (equivalent to 2.1) (to be performed in 1987-88)

b/ **ICARE** : Irradiation in MELUSINE (pool type research reactor) of 263 MOX pins,with two experimental rods containing isolated isotopes in order to determine individual capture rates in a representative spectrum.

c/ **MORGANE** : Measurement of fission product gross capture by means of oscillations in an undermoderated core in the MINERVE facility. The several samples to be oscillated are in the range 20 - 60 GWd/T.

III.1.3. - Conclusion :

The comparaisons of calculated results and experimental results lead to small errors, of the same order of magnitude in tight lattice cores and in standard PWR cores. Thus it is now possible to undertake design studies as soon as EdF and FRAMATOME might wish to do it.

III.1.4. - Results of parametric studies :

Multiplication factor k_∞ versus Vm/Vu moderation ratio for several Pu content ("1st generation Pu", i.e. Plutonium issued from 33 GWd/T - UO_2 - PWR reprocessed fuel).

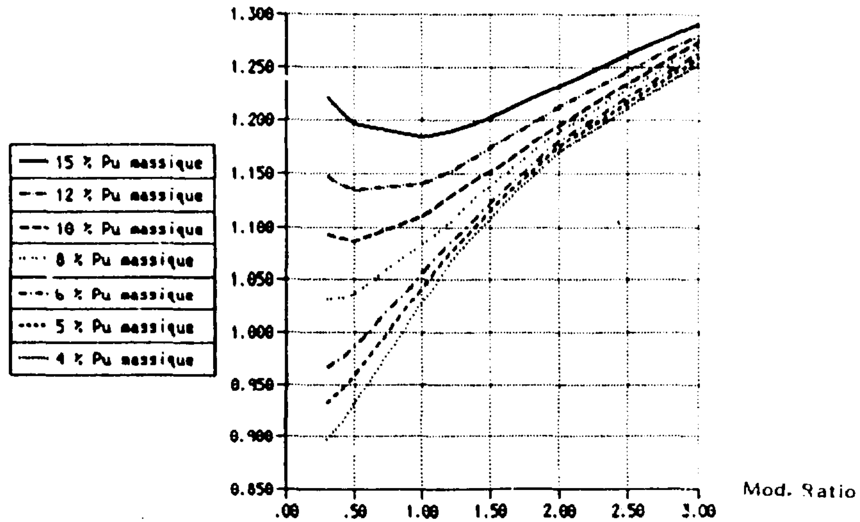

FIGURE 1 Multiplication Factor

Plutonium content versus target burn-up for various moderation ratio and for various plutonium isotopic composition have been evaluated with and without axial and radial blankets, with and without spectral shift rods.

All significant parameters have also been evaluated in many cases :

- RCC absorption rate,
- moderator coefficient,
- Doppler effect,

- Power gross effect,
- Reactivity decrease with burn-up
- Void effect.

Just have a look on void effect for RCVS core ($Vm/Vu \simeq 1.4$) :

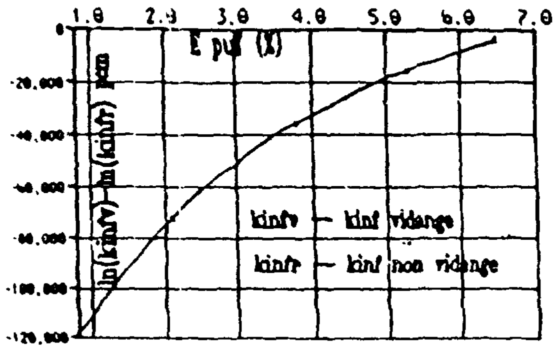

FIGURE 2 Void Effect

III.1.5. - Conversion features

Let us recall some useful definitions :

Conversion Ratio $= C = \dfrac{\text{(capture rate in fertile material)}}{\text{(fission rate)}}$

Pu inventory ratio $= \dfrac{\text{(Pu mass in new fuel assembly)}}{\text{(Pu mass in discharged fuel)}}$

Pu FIR = fissile inventory ratio, for fissile Pu isotopes,

Pu TIR = total Pu inventory ratio,

Pu Eq IR = Equivalent Pu (for RNR) inventory ration,

Net Pu consumption = NC = (mass of Pu in) - (mass of Pu out)

Specific consumption $= \dfrac{NC}{E} = \dfrac{\text{Net Consumption}}{\text{Energy Output}}$

Specific Fission Energy $= \dfrac{E}{FM} = \dfrac{\text{Energy Output}}{\text{Total mass of fissioned nuclides}}$

Fission Consumption : FC = Total mass of fissioned nuclides

Plutonium generation rate = Pu GR $= 1 - \dfrac{NC}{FC}$

(Note that $\dfrac{NC}{FC}$ = "normalized" specific consumption)

For a target discharge burn-up of 45 GWd/T, the Plutonium generation rate ranges from

0.80 (Vm/Vu = 1.6, no blanket),

to $\gtrsim$ 0.90 (Vm/Vu = 1.1, blankets and spectral shift).

III.2. - Thermalhydraulics and Safety

III.2.1. - Critical flux

An experimental program is under way in the Grenoble Nuclear Center (IRDI-DEDR-DRE-SETh)

- on the GRAZIELLA facility, in freon,
- on the OMEGA facility, in water.

From the two first series of tests on Graziella a critical flux correlation has been derived for tight lattices (Vm/Vu = 0.6) which is called "GSM 06".

Another correlation called G 1 will be derived using all results of the various series of test for both tight and semi tight lattices.

The test series characteristics are summarized here under :

Vm/Vu	channel type	Nb of rods	Heating length
0.6	typical cell	19	2 m
0.6	guide tube	19	2 m
0.6	inter assy gap	24	2 m
1.1.	typical cell	19	4.2 m
1.1.	guide tube	19	4.2 m

All series will be performed on Graziella ; some of them will be performed on Omega in 1987 and 1988.

III.2.2. - Reflood :

Preliminary experimental studies have been performed on the ECCHO-B facility in Grenoble Nuclear Center (IRDI-DEDR-DRE-SETh), using two 37 rod bundles, one for tight lattice (Vm/Vu = 0.6), the other for semi tight lattice (Vm/Vu = 1.1)

The range of parameters was

- pressure	0.1	-	0.2	MPa
- heat flux	1.8	-	3.5	W/cm^2
- initial wall T°	300	-	800	°C
- inlet subcooling	30	-	60	°C
- mass flow rate	3.6	-	1.2	g/cm^2.s

Water was injected at the bottom of the test section for both test series. In addition to these tests, upper injection was performed, as well as combined injection, on the second test section (Vm/Vu = 1.1).

Main results are :

- the quench front progresses very slowly in the tight lattice bundle, and even stops in some cases ;

- the quench front progresses in all cases in the semi tight lattice bundle up to the top, and, although the quench velocity is lower than in a PWR (Vm/Vu = 2) bundle, no major difficulty is expected.

Moreover, the CATHARE code fits the experimental results in the semi tight lattice bundle remarkably well, as shown in the following figure : (Fig. 3)

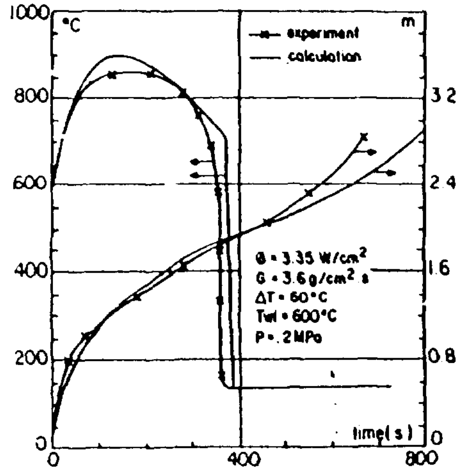

FIGURE 3

III.2.3. - Flow blockage

Even in semi tight lattice core, according to the rod geometry : external diameter = 9.5mm and to the grid pitch = 12.23 mm, ballooning may block the channel :

$\frac{\Delta \ell}{\ell}$ = 4.3 is enough for 100 % blockage.

However, new clad material such as Zy4 quenched in β phase might be convenient.

A first series of tests on the EDGAR facility (Saclay Nuclear Center ; IRDI-DMECN-DTECh-SRMA) led to a maximum $\frac{\Delta \ell}{\ell}$ at clad rupture of about 50-55 % ; in such condition one may expect mean values of $\frac{\Delta \ell}{\ell}$ 35 % and no major blockage problems.

Further tests might be undertaken in order to develop a modelling of thermal and mechanical behaviour of this material.

IV. - EXAMPLES OF SOME RESULTS

IV.1. - Tight lattice core (Vm/Vu = 0.6)

. The core was composed of

187 fissile fuel assemblies

54 fertile fuel assemblies (radial blanket)

61 rod clusters

The target burn-up was 45 000 MWd/T (3 x 15 000).

The required Pu content is 8.9 % fissile content (12.5 % total Pu content).

The Pu total inventory ratio is 0.87,
0.82 in the fissile core, 0.05 in radial & axial blankets.

The limits of some safety concerns are tresspassed :

a/ Void effet is largely positive

. global effect is estimated in the range
2700 to 4400 10^{-5} (BOC - EOC)

. local effect is estimated in the range
+ 6 000 to + 7 300 10^{-5}

b/ DNB margins would require primary pumps delivering much higher flow rate than present PWR pumps.

c/ Reflood is proved to be very difficult.

IV.2. - Semi tight lattice core (Vm/Vu = 1.1 - 1.4)

Several cores have been investigated,

- with and without blankets,

- with and without spectral shift clusters,

- for several burn-up and loading modes.

Let us have a look on results dealing with one Vm/Vu = 1.4 core for three cases :

Case nr	1	2	3
Blankets	yes	no	no
Fertile Ass.	54	0	0
Fissile Ass.	199	253	253
Relodfraction	1/4	.4	1/5
Target burn-up (GWd/T)	45	45	55

The results of equilibrium cycle evaluation are :

Case nr	1	2	3
fresh fuel fissile Pu content %	6.0	5.7	6.5
discharge B U	45.5	44.4	54.8
cycle duration (MWd/t)	12 300	11 500	11 700
(days)	358	335	339

These Pu contents are consistent with negative void effects (local and global). The feasibility of first cores loading has been investigated, but not optimized.

The hot assembly factor for each case at beginning of equilibrium cycle is :

Case nr	1	2	3
Assy factor	1.262	1.242	1.254

The local hot pin factor inside one assembly is 1.10, near the water holes if the rod cluster is out.

The balance of reactivity for control rod design has been evaluated in two cases :

nr 1 - core with blanket, 61 RCC

nr 3 - core without blanket, 85 RCC

In both cases, the core may be controlled at intermediate -zero power - 150°C - stage without soluble boron, but soluble boron has to be used to stop the reactor at 20°C.

The DNB margins have also been evaluated for the 3 cases. Of course, since the number of fissile assemblies is different from one case to the other, the mean linear heat flux was different, of the order of 150 W/cm in cases 2 and 3, and of the order of 180 W/cm in case 1.

The flow decrease transient (successive to a pump trip) has been calculated.

There is no difficulty in cases 2 and 3, but in case 1, DNB margins are tresspassed.

The mean linear heat flux has to be decreased down to about 160 W/cm, or the pump characteristics have to be slightly improved (or both !). However, an increase of about 10 % of the flow rate seems to be possible without designing a totally new pump, RCP used in N4 type plants might be modified to reach this goal.

The Large Break LOCA (200 % Cold Leg Break) have been calculated ; it leads to lower clad temperatures at first peak, and never higher clad temperature than in PWRs.

V. - CONCLUSION

It has been proved that

* *In the range Vm/Vu $\geqslant$ 1.1, there is significant interest in water reactors using Pu* core, and, in addition to that, FRAMATOME suggests a convertible concept using either UO_2 or UO_2-Pu O_2 fuel.

* the positive void effect limit is above 7 % Pu fissile content, wich is consistent with Vm/Vu $\geqslant$ 1.1 and discharge burn-up of 60 GWd/T.

* the DNB margins for cores with $Vm/Vu \geq 1.1$ lead to linear heat fluxes of the order of 160 W/cm, wich are consistent with the target power, even if some minor improvment of RCPS is required.

* the LOCA doesn't seem to be a limit for semi tight lattice cores, the relativily low quench velocity being compensated by more favorable initial conditions.

* flow blockage should not be a problem, although it has still to be checked.

* Basic knowledges, computation models and codes are available to undertake a detailed design in conditions similar to those of present PWR design.

No decision is taken in France ; future work will depend of international situation, and of the specifications of REP 2000 that EdF will finalize end of 87, beginning of 88.

LA RECHERCHE ET LE DEVELOPPEMENT SUR LES REP FRANÇAIS

Y. DENNIELOU
Electricité de France,
Villeurbanne, France

Généralités

L'amélioration de la sûreté et de la disponibilité des centrales nucléaires doit être le souci constant des producteurs d'énergie, et pour le second point d'autant plus qu'ils disposent d'un parc nucléaire dont la contribution est importante dans la production d'électricité.

L'expérience acquise en France, qui est dotée d'un parc de l'ordre de 50 unités nucléaires en exploitation (en majeure partie constitué de réacteurs à eau pressurisée) et totalisant une puissance de près de 45 GWe, a conduit EDF à engager un certain nombre d'actions de recherche et développement dont le montant est estimé à près de 300 MF pour une année, uniquement pour l'amélioration des réacteurs à eau pressurisée.

Une part importante de ces actions de recherche et de développement se fait d'ailleurs en collaboration avec les organismes de recherche du C.E.A. et avec le constructeur FRAMATOME.

La politique de Standardisation qui est une des causes du succès dans la réalisation du programme nucléaire français conduit EDF, pour ce qui concerne le futur, à respecter les mêmes principes que ceux qui ont présidé à l'évolution des différents programmes, soit : procéder par palier, en améliorant les conceptions, mais sans changement notable, de façon à tirer profit de l'expérience acquise, tant dans le contrôle des coûts que dans les leçons tirées des incidents constatés.

La recherche et le développement sur les R.E.P. français

EDF non seulement est maître d'ouvrage et exploitant, mais a aussi un rôle d'architecte industriel des centrales qu'elle programme.