GENERAL REMARKS ON FAST NEUTRON REACTOR PHYSICS

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Abstract

The main aspects of fast reactor physics, presented in these lecture notes, are restricted to LMFBR's. The emphasis is placed on the core neutronic balance and the burn-up problems. After a brief description of the power reactor main components and of the fast reactor chronology, the fundamental parameters of the one-group neutronic balance are briefly reviewed. Then the neutronic burn-up problems related to the Pu production and to the doubling time are considered.

TABLE 1

	THERMAL			FAST		
	235 U	239 Pu	233 U	235 U	239 Pu	233 U
$\eta - 1$	1.07	1.11	1.28	.88	1.33	1.27
ν	2.42	2.88	2.49	2.43	2.94	2.53
α	.17	.36	.09	.29	.26	.12
σ_f barn	582	743	528	1.81	1.76	2.79

Don't forget also that the depleted Uranium, the waste of enrichment facilities, can be used without any problem in fast breeders.

It must also be mentionned that the use of sodium leads to good temperature and pressure vapor conditions giving a high thermodynamic efficiency (40 to 45 %) : this is a really favorable environmental consequence.

Pu use and high specific powers are looked for : 0,6 to 0,8 MWth/kg Pu. Even with the high fissile density, that leads to high flux values, 3 to 7 10^{-15} n. m^{-2} s^{-1}. That is one of the core fondamental characteristic for the design.

The fuel is usually made of sintered mixed oxide cylindrical pellets, 4 to 7 mm in diameter. These pellets are placed in thight stainless steel cylindrical clads, 0,4 to 0,8 mm thick, that represent the first safety barrier for fission products. These pins are assembled in hexagonal stainless steel boxes that make the subassemblies : 169 to 325 pins per subassembly. The subassembly represents the elementary unit for core loading and unloading : it can contain between 10 and 20 Pu Kg.

1.1.2. Cooling

The high specific power per Pu mass leads to very important volumic powers, between 0,3 to 0,5 MWth per core liter. Then, it is necessary to have large exchange surfaces (pins) and to choose a cooling material with high thermal performances, small neutron absorption and slowing down, small activation to neutrons.

The sodium, commonly used now, has the best characteristics for FBR's. Sodium volumic percentages vary between 30 to 40 %, the sodium flows are of the order of 3 to 20 tons per second. The sodium temperatures are 380-400°C at core inlet and at the maximum 550°-600° in normal conditions. Then, there is no need to pressurize the circuits : that represents a very important advantage for construction and operation.

Due to the remarkable sodium thermal properties, high sodium temperatures at the core outlet can be reached leading to favorable vapor conditions for thermodynamic efficiencies.

Finally, the order of magnitude of the core volumic percentages are :

mixed oxide fuel	: 30 to 45 %
sodium	: 30 to 40 %
structural materials, cladding	: 20 to 25 %
gas	: 2 to 5 %

1. POWER REACTOR COMPONENTS

Although fast reactors are probably well-known for all of you, I would like to give a brief description of the main components of a power reactor that are important for physics.

1.1. Core (Fig. 1)

1.1.1. Fuel

As previously mentioned, the main characteristic is the high fissile material density. Mixed oxide fuel volumic percentages vary between 30 and 45 % for oxide fuel densities between 80 and 95 % of the theoritical value. According to the power and the design, enrichments vary between 30 %, low power (≃ 200 MWe), and 13 %, high power (≃ 2000 MWe).

The necessity to use enriched fuels represents a penalty to be reduced as far as possible. Then, the best

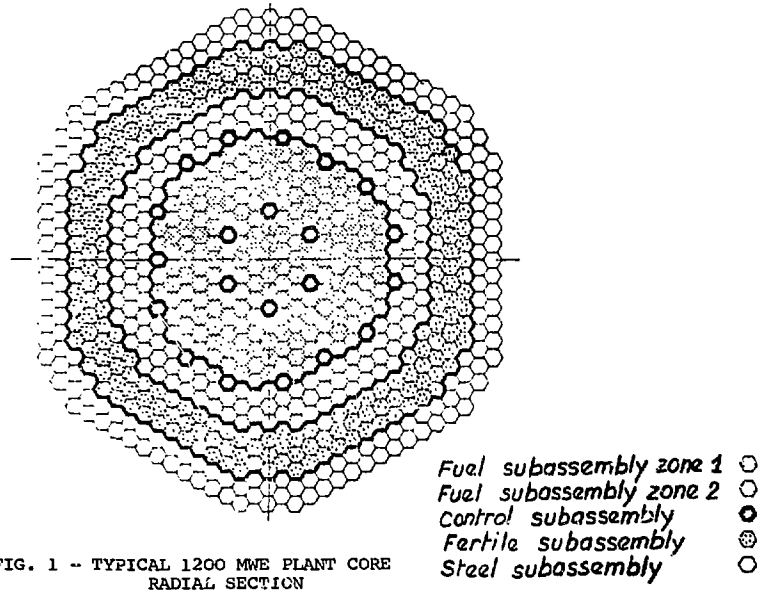


FIG. 1 - TYPICAL 1200 MWE PLANT CORE RADIAL SECTION

1.1.3. Core geometry and volume (fig. 2)

The core height is defined from economic conditions and is of the order of 1 m. Then the axial flatenning factor, ratio of the maximal lineic power to the average lineic power is about 1.3.

Radially, the power flatenning is improved by using two enrichment zones ; the enrichment ratio internal zone over external zone being of the order of 0,7 to 0,8.

Finally, for a maximal power of 450 Watt/cm, to produce 250 MWe or 1200 MWe 20 or 100 km of pins are needed. Then, core volumes correspond to 1000 or 10 000 l and fissile critical masses vary between 800 to 5000 kg according to the power.

1.2. Blankets (fig. 2)

The fissile core is surrounded axially and radially by fertile blankets made of depleted or natural oxide Uranium, sodium cooled, which have four main roles :

a. neutron reflectors

b. Pu production : it is a fundamental FBR's characteristic

c. power production : 5 to 12 % of the total power

d. neutronic shielding.

As we will see later on, due to the core compaction and the neutron average mean free path, the percentage of neutrons escaping out of the core is really high in fast breeders. The blankets represent the optimal use of these neutrons. Especially, you will see that fast reactors are only breeders because Pu atoms are produced in the blankets. Blanket thicknesses are defined from economic criteria and vary between 25 and 50 cm. The maximum UO_2 volumic percentage is looked for, but it must be recalled that the blanket power can doubie between beginning or end of life (Pu production).

Axially and radially, blankets are surrounded by shielding (for example stainless steel-sodium) which have three main roles :

- shielding against gamma and neutrons
- damage limitations on the structures
- component activation limitations (for pool-types reactor, mainly limitation of the secondary sodium activation at the intermediate heat exchanger level).

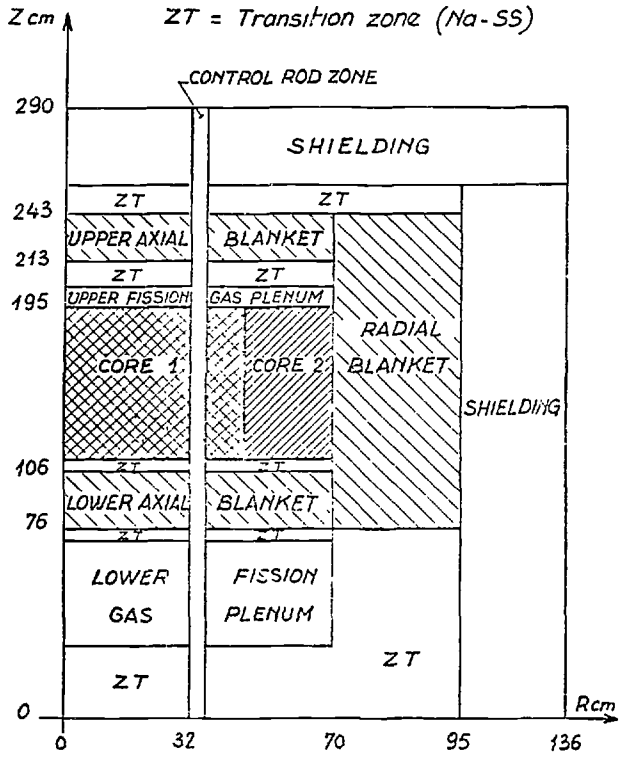


FIG. 2 - PHENIX TYPE PLANT-CORE
AXIAL SECTION

1.3. Control rod (fig. 1)

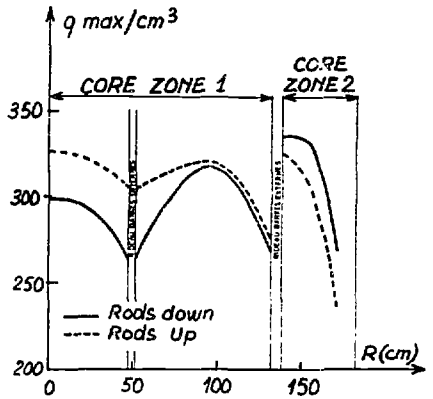
The reactor control is obtained by control rod subassemblies replacing normal subassemblies. Boron carbide enriched or natural is commonly used in pins stainless steel cladded, sodium cooled. Then, control rods have three fonctions :

- safety : $\simeq$ 3 % in keff
- compensation : loss of reactivity due to neutronic burn-up, temperature and power rises ($\simeq$ 3 and 2 % in keff depending on the design)
- operation.

At the end of the cycle, the absorber part is out of the core, then sodium holes remain in the core.

The optimisation of the whole control rod system, taking into account the global antireactivity needed ($\simeq$ 8 to 10 % in keff), is performed on the composition, the number and the positions of the control rods in the core : it is rather design dependent.

However, for large power reactors, the most important physics problem is the influence of control rods on radial power distributions and the interactions between the control rods. The control rod positions are commonly used to improve the radial power flatenning and to control the radial power distribution during burn-up (hot spot control - fig. 3).



RADIAL POWER DISTRIBUTION IN THE CORE
FIG. 3 WITH TWO CONTROL ROD RINGS

2. FAST REACTOR CHRONOLOGY

2.1. The first fast neutron chain reactions were obtained on highly enriched fuel masses for military purpose. JEZEBEL or GOVIDA in the States used about 10 kg of fissile fuel.

2.2. Moving towards the civil applications, the faisability of fast neutron reactors was demonstrated rather early by two experiments, CLEMENTINE (25 KWth, 1946) and especially EBR 1 that produced the first world nuclear electricity (1951).

2.3. The main development work was orientated in the fifties and sixties on technology and fuel research. Irradiation reactors were built, for example BR 5 (1958, 5 MWth), Bor 60 (1969, 60 MWth), DFR (1961, 72 MWth), EBR II (1963, 62 MWth), RAPSODIE in FRANCE (1967, 24 MWth) and its FORTISSIMO version (1971, 40 MWth).

2.4. The industrial prototypes appeared in the late sixties-early seventies using the previous technology and reaching power levels between 500 and 100 MWth, for example BN 350 (1971, 1000MWth), PFR (1974, 270 MWe), PHENIX (1973, 250 MWe) in FRANCE that reached the 4 billion KWh production level in february this year. Several other prototypes are now being built in various countries.

2.5. According to the satisfactory results obtained from prototype operation, several large power size stations are now being constructed for example BN 600 in USSR (600 MWe) and SUPERPHENIX 1 (1200 MWe) for which the industrial operation is planned in 1983.

3. NEUTRONIC CHARACTERISTICS

This short presentation of the fast breeder neutronics characteristics assumes that the general neutronics theory is well-known.

General problems related to nuclear data

a) In the neutron energy range useful for fast breeders, 0,5 Kev to 10 Mev, all the microscopic cross sections for the various isotopes are of the same order of magnitude (fig. 4 to 7). The competition between the various reactions and the various isotopes is then rather strong. This explains also the large amount of informations required to calculate a fast breeder core.

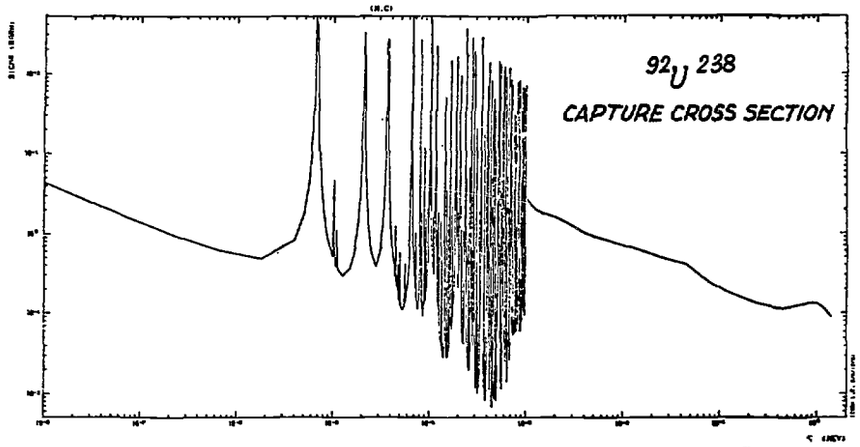


FIG. 4

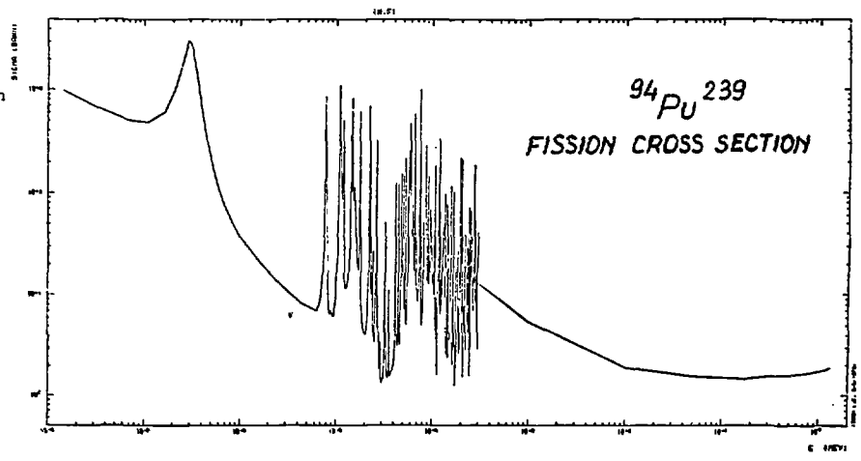


FIG. 5

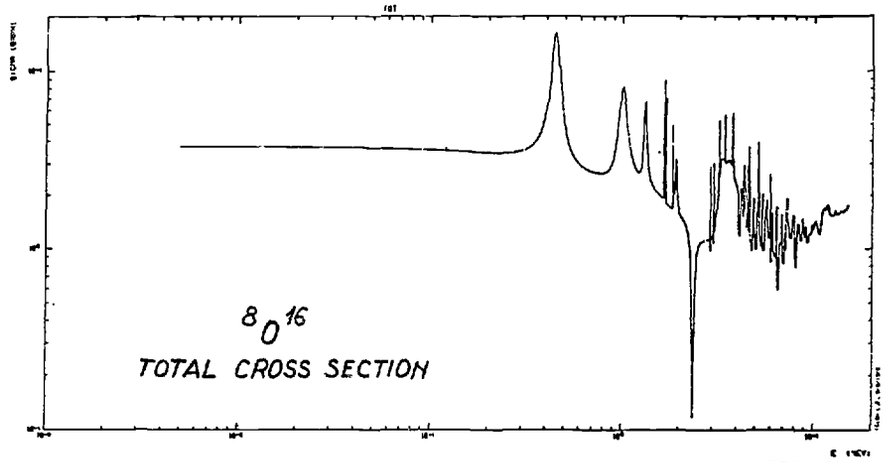


FIG. 6

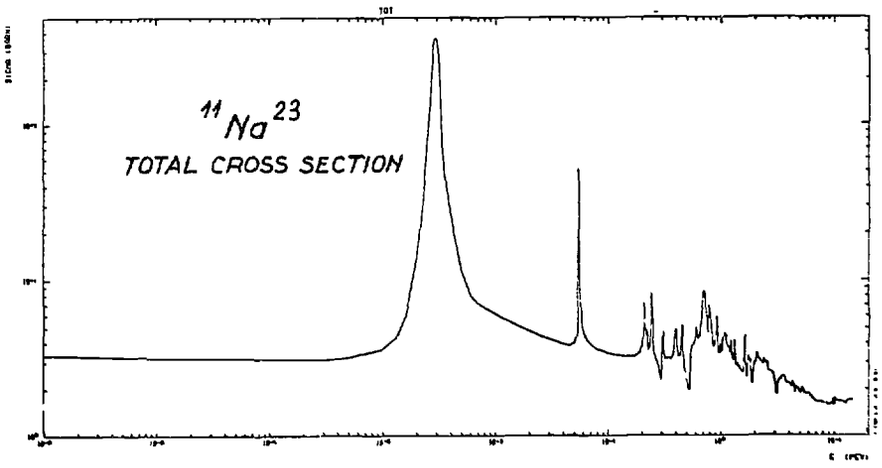


FIG. 7

b) The isotopes called "light" as oxygen, sodium, iron, chromium, nickel, contribute to the elastic slowing down : they present very broad diffusion resonance (fig. 6-7)

	Energy (KeV)	Width (KeV)	Average spacing (KeV)
Na	2.85 ± 0,015	0.41 ± 0.0[illegible]	30 ± 6
Oxygen	440 ± 4	45 ± 5	300 ± 50

c) The isotopes called "heavy" as Plutonium isotopes or 238 U present also resonances in the low energy range, below 60 Kev. These heavy isotopes absorption resonances differ largely from the light isotope diffusion resonances as it is shown for the main resonance parameters of 238 U or 239 Pu : (fig. 4-5)

	239 Pu	238 U
Fission width :	34 or 2100 mev	
Capture width :	40 mev	24 mev
Spacing :	3,2 or 9,6 ev	23 ev

d) The elastic slowing down is mainly due ($\geqslant$ 90 %) to light elements in the order, oxygen, sodium, iron (table 2). The elastic slowing down cross section represents approximatively 90 to 98 % of the total macroscopic cross sections, for a standard composition, below 100 Kev, where the inelastic slowing down remains small :

ENERGY		TOTAL	ABSORPTION	ELASTIC	INELASTIC
100 Kev	$\sum$ cm^{-1}	0.3	.0035	.293	.044
	%	100 %	1,2%	97,5%	1,3 %
1.8 Mev	$\sum$ cm^{-1}	0.18	.0056	.138	.037
	%	100 %	3,1%	76,2%	20,7 %

e) At the opposite, in the high energy range, the important heavy isotope volumic percentage leads to a large inelastic slowing down, mainly by 238 Uranium (50 to 90 % between 300 Kev and 1 Mev).

f) The neutron mean free path is large compared to the fuel and cell dimensions, decreasing from 5 cm at high energy to 2 cm at 1 Kev : then, the heterogeneity problems remain simpler than in thermal reactors.

g) The threshold fission cross sections for example on 238 U or 240 Pu, play a major role in the neutron balance due to the fuel composition. Threshold reactions (n, 2n), (n,p), (n, α) can not be neglected either for heavy isotopes or for structural materials although the energy thresholds are above 2 Mev.

Table 2

AVERAGE ENERGY LOSS
BY ELASTIC COLLISION (IN KEV)

E	1 MeV	100 KeV	1 KeV
Oxygen	115	12	0,12
Sodium	60	8	0,08
Iron	27	3	0,04
238 U	5	0,8	0,00

3.1. Energy neutron balance

3.1.1. Neutron spectrum

Due to the competition between the various reactions, the energetic spectrum of the neutron flux plays a really important role in FBR physics : it varies continuously versus the reactor type (fig.8).

All the neutrons are born from fission with a maxwellian spectrum (average energy 2 Mev) : there is no more neutron coming directly from fission below 100 Kev ($\leqslant$ 1,5 %). At high energy, the competition exists mainly between the production on one side, the leakage and the inelastic slowing down on the other side.

Down to 100 Kev, all the reactions compete including the absorption and the elastic slowing down. Below 100 Kev, the absorption role increases and the slowing down is purely elastic. Finally, in the low energy range, the largest majority of the neutrons are absorbed, over the 100 ev energy, by the 238 U capture resonances.

Finally, the greater the core power, the softer the spectrum. An increase of the reactor power leads to a core volume increase, a leakage decrease, then an enrichment reduction giving a lower relative role to the production compared to the other reactions.

The neutron spectrum is also strongly influenced by the large elastic diffusion resonance (see fig.8).

3.1.2. Slowing down

The variation of the slowing down densities versus energy for the elastic and inelastic reaction rates is given in figure 9.

For the integrated total slowing down densities, the percentages for both reactions are : elastic 58 %, inelastic 42 %, but the energy distributions of slowing down densities are rather different :

first moment : elastic 350 Kev
inelastic : 1,5 Mev

To characterize the neutron spectrum by only one parameter, the r parameter, defined by R. NAUDET as the ratio of the average number of neutron produced per fission $\nu \Sigma_F$ to the integral of the slowing down density $\xi \Sigma_s$ is commonly used in France. It can be easily shown that 1/r represents the averaged lethargy interval covered by a neutron during its life : it is the average lethargy shift between the fission neutron spectrum and the disparition neutron spectrum. The r parameter gives a good representation of the spectrum hardness, taking into account not only the fuel but the whole cell. For power plants, r varies between 0,5-0,4 for 250 MWe reactors and 0,4-0,3 for 1200 MWe.

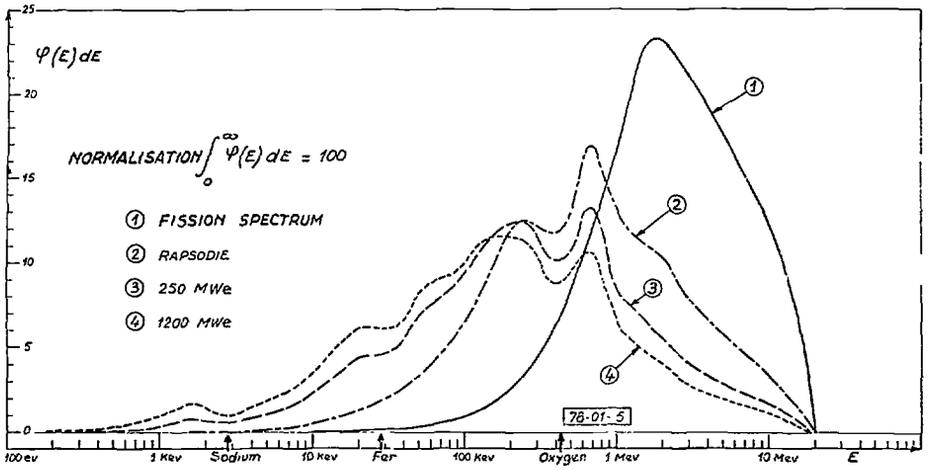


FIG. 8. NEUTRON FLUX ENERGETIC SPECTRUM IN A FAST REACTOR

3.1.3. Energy distribution of the neutron balance components (fig. 10)

The neutron balance can be easily split up into four components :

production = capture + fission + leakage.

The energy distribution of these four components is represented fig. 10 for a 1200 MWe typical reactor. The probability energy threshold 50 % varies strongly for the various terms :

production = 1,3 Mev leakage : 230 Kev
capture : 18 Kev
fission : 150 Kev

3.2. Integral neutron balance

The knowledge of the integral neutron balance, or average reaction rate neutron balance, represents the main role of reactor physics work. If the analysis of the energy neutron balance is important for the detailed understanding of the phenomena, at the level of the definition and of the optimisation of the plan characteristics, only integral quantities are needed for the design.

3.2.1. Global analysis of the integral balance

The decomposition of the neutron integral balance for a typical 1200 MWe reactor into the usual balance components is given table 3 for 100 neutrons produced per fission.

This table shows firstly the importance of the leakage (22 %). The analysis of the isotope contributions demonstrates the first role of 239 Pu for the production and the absorption per fission and the first role of 238 U for the capture rate (239 Pu production).

The contributions of the other isotopes remain however important : Pu isotopes for fission and capture, structural materials for capture.

3.2.2. Average microscopic cross sections per isotope

The various balance components are calculated from average microscopic cross sections : an order of magnitude of these data is given table 4 for a typical 1200 MWe reactor (enrichment : 15 %).

Outside the 235 U - 239 Pu comparison previously commented for FBR's, the favorable ν and fission cross sections for 241 Pu, the positive ($\nu\sigma_f - \sigma_a$) parameter for 240 Pu and the low microscopic capture cross section for 238 U must be noted. The important 238 U role in the capture balance comes from its concentration. The structural material cross sections show the difficulty of high concentration in stainless steel of nickel, molybdenum or manganese. Finally, sodium and oxygen captures are negligeable in the balance.

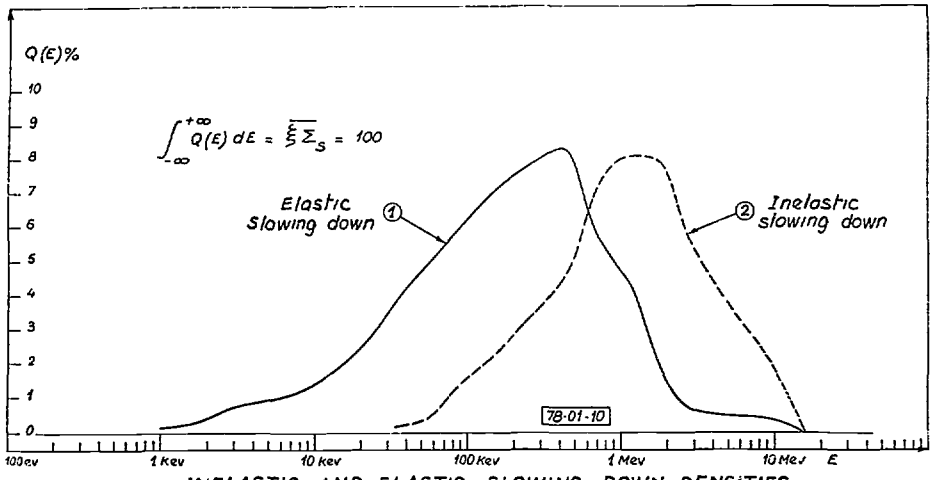


FIG. 9. INELASTIC AND ELASTIC SLOWING DOWN DENSITIES
SLOWING DOWN DENSITY TOTAL NORMALISATION

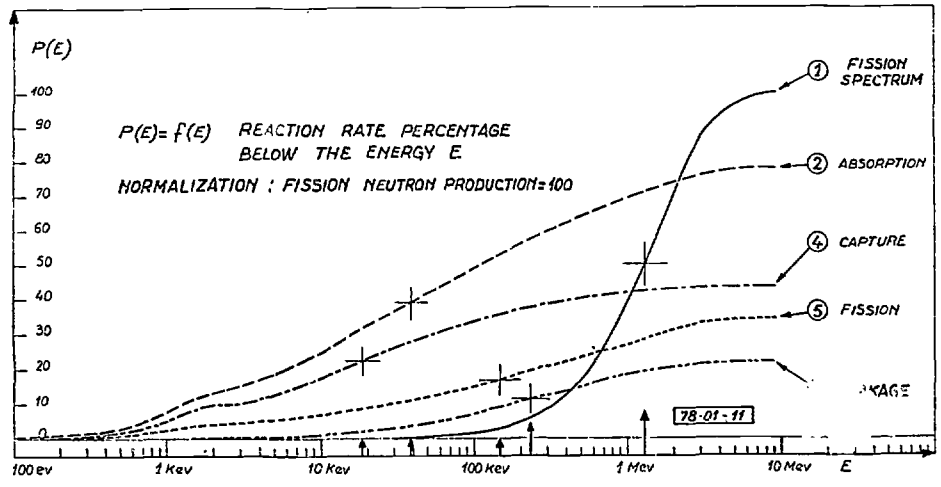


FIG. 10. ENERGY DISTRIBUTIONS OF THE NEUTRON BALANCE MAIN COMPONENTS

TABLE 3 - DECOMPOSITION OF THE CORE INTEGRAL NEUTRON BALANCE FOR A 1200 MWE TYPICAL REACTOR

COMPONENT	ISOTOPE TYPE	ISOTOPE
PRODUCTION 100 neutrons	FISSILE 83.7	239 Pu : 77.5 241 Pu : 4.2 235 U : 2.0
	FERTILE 16.3	238 U : 12.0 240 Pu : 4.2 242 Pu : 0.1
LEAKAGE 22		
ABSORPTION 78	FISSILE 37.2	FISSION 28.7: 239 Pu : 26.5; 241 Pu : 1.4; 235 U : 0.8
		CAPTURE 8.5: 239 Pu : 8.0; 241 Pu : 0.3; 235 U : 0.2
capture : 44 fission : 34	FERTILE 35.8	FISSION 5.6: 238 U : 4.2; 240 Pu : 1.4; 242 Pu : 0.0
		CAPTURE 30.2: 238 U : 28.1; 240 Pu : 2.0; 242 Pu : 0.1
	PARASITIC 5.0	Fe : 1.9 Cr : 0.6 Ni : 0.9 Mo 1.0 Mn 0.6

3.2.3. Cell parameters

a) Among the various parameters that characterize globally the cell neutron balance , the material buckling B^2_m is the most important one :

$$B^2_m = \frac{\nu \Sigma_F - \Sigma_A}{D}$$

with D the leakage coefficient.

The material buckling is proportionnal to the difference between the neutron production and the neutron absorption, i.e. to the leakage and it is very simply related to the critical dimensions of the bare medium.

The variation of the material buckling is given fig. 11 versus the r parameter for three independent parameter changes :

- the enrichment : 12 to 30 % with constant fuel volumic percentage
- the fuel volumic percentage : 24 to 44 % with constant enrichment
- the stainless steel volumic percentage : 17 to 32 % compensated on the sodium volumic percentage , with enrichment and fuel volumic percentage constant.

The major importance of the enrichment E parameter can immediately be noted.The second important parameter is the fuel percentage. The relative sensitivities are :

$\frac{dB^2}{B^2}$ %	Enrichment %	Fuel %	Stainless steel %
250 MWe	1,5	1,3	- 0,1
1200 MWe	2,7	2,1	- 0,1

TABLE 4 - AVERAGE MICROSCOPIC CROSS SECTIONS IN BARNS

	$\overline{\sigma}_f$	$\overline{\nu}$	$\overline{\nu\sigma}_f$	$\overline{\sigma}_c$	$\overline{\sigma}_a$	$\overline{\nu\sigma_f - \sigma_a}$	$\overline{\eta}$
235 U	1.951	2.445	4.771	.584	2.535	2.236	1.882
238 U	. 040	2.900	.116	.271	.311	-.195	.373
239 Pu	1.810	2.922	5.289	.547	2.357	2.932	2.244
240 Pu	.371	2.976	1.104	.523	.894	.210	1.235
241 Pu	2.640	2.968	7.836	.568	3.208	4.628	2.443
242 Pu	.252	3.087	.778	.622	.874	-.096	.890
Fe				.0077	.0077	-.0077	
Cr				.0077	.0077	-.0077	
Ni				.0188	.0188	-.0188	
Mo				.157	.157	-.157	
Mn				.063	.063	-.063	
Si				.0023	.0023	-.0023	
O				.0009	.0009	-.0009	
C				$.9\ 10^{-5}$	$.9\ 10^{-5}$	$-.9\ 10^{-5}$	
Na				.0022	.0022	-.0022	

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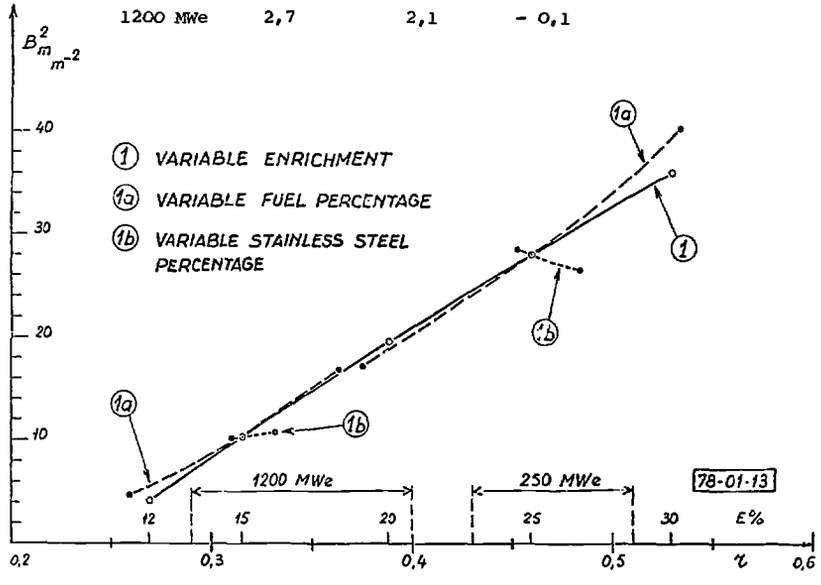


MATERIAL BUCKLING VARIATION VERSUS THE ENRICHMENT AND THE VOLUMIC PERCENTAGES

FIG. 11.

In a real design, should the fuel volumic percentage increase, the enrichment decreases to maintain the criticality : there is a strong compensation between both effects.

It can be also noticed that for the full power range studied, the material buckling, then the leakage varies 3 to 4 times.

b) The variations of the leakage coefficient D for the same parameter changes are relatively small compared to that of the material buckling, for example :

Enrichment	: 15 %	25 %
D	: 1,4	1,5

c) The migration area $M^2 = \frac{D}{\Sigma_A}$ is naturally higher in fast breeder compared to thermal reactors : M^2 = 250 to 280 cm^2.

d) <u>Ratio production over absorption</u> : $K^* = \frac{\nu\Sigma_F}{\Sigma_A}$

That parameter defines the surcriticality potentiality of the lattice calculated on the fondamental mode.

At the criticality level, the cell balance can be expressed by the well-known formula :

$$Keff = 1 = \frac{K^*}{1 + M^2 Bg^2}$$

It is clear that the K* variation will be of the same type that the leakage variation, especially K* increases with r :

Enrichment	15 %	30 %
K*	1,25	1,85

From the previous expression of the cell balance, it is rather simply to calculate the reactivity variation due to the cell characteristics variation (material buckling) or due to the reactor geometry (leakage or geometrical buckling) :

$$\frac{d\,Keff}{Keff} = \left[\frac{K^* - 1}{K^*}\right]\left[\frac{d\,B^2m}{B^2m} - \frac{d\,B^2g}{B^2g}\right]$$

This relation allows to calculate the reactivity effect due to a 1 % critical mass variation obtained either by enrichment variation (before fuel fabrication) or by radius variation (after fabrication), for example for a 1200 MWe reactor:

	Enrichment	Radius
$\frac{dk}{k}$ %	0,6	0,3

The interest to define accurately the reactivity uncertainty before fabrication appears clearly.

3.2.4. <u>K* parameter analysis</u>

The leakage represents a global property of the cell, but the ratio production over absorption can be decomposed between the various cell components. The well known four factors formula can also be used in fast reactor physics with the following definitions :

$$K^* = \varepsilon \cdot \eta \cdot p \cdot f$$

$$\varepsilon = \frac{\nu\Sigma_F\ (\text{fertile}) + \nu\Sigma_F\ (\text{fissile})}{\nu\Sigma_F\ \text{fissile}}$$

$$p = \frac{\Sigma_A\ (\text{fissile})}{\Sigma_A\ (\text{fissile}) + \Sigma_A\ (\text{fertile})}$$

$$\eta = \frac{\nu\Sigma_F\ (\text{fissile})}{\Sigma_A\ (\text{fissile})}$$

$$f = \frac{\Sigma_A(\text{fertile}) + \Sigma_A(\text{fissile})}{\Sigma_A(\text{fertile}) + \Sigma_A\ (\text{fissile}) + \Sigma_A(\text{parasitic})}$$

The table 5 presents this decomposition for a 1200 MWe typical core. The main conclusions previously mentionned can also be done on this analysis.

Finally, the previous remarks put in evidence the main problems to solve in fast reactor physics calculationnal methods :

- large number of nuclear data ;
- division of the calculations into two well separated problems as for thermal reactors ;
  - . the cell reaction balance problem for which a detailed energy analysis and an accurate leakage treatment are needed,
  - . the global spatial calculation for which the energy treatment can be largely simplified but the geometry description detailed.

TABLE 5 - ANALYSE OF K*

ON THE FOUR FACTORS FORMULA

FACTOR	VALUE	ISOTOPE
ε	1.194	ε -1 238 U : .1431 240 Pu: .0507 242 Pu: .0007
η	2.243	$\bar{\eta}$ (239 Pu): 2.245
P	.511	$\frac{1}{p}$ - 1 238 U: .8687 240 Pu: .0919 242 Pu: .0018
f	.936	$\frac{1}{f}$ - 1 Fe : .0245 Cr : .0069 Ni : .0104 Mo : .0123 Mn : .0063 O : .0032 Na : .0041
$K^* = \varepsilon \eta p f$	1.282	

4. NEUTRONIC BURN-UP

The fuel cycle problem is especially important for FBR's ; on an economic basis, the main advantage of that reactors relies on their low fuel cycle cost that allows to accept an eventual higher investment cost than light water reactors. The fuel cycle knowledge in-pile or out of pile represents a fondamental problem for FBR's physics.

4.1. Breeding gain and doubling time

4.1.1. 239 Pu equivalent weight for Pu isotopes

Due to the different isotopic compositions used in FBR's, it is necessary to define an equivalence between these isotopes for the design. The commonly used equivalence, defined by A. BAKER, is a reactivity equivalence in a 239 Pu - 238 U scale where the 239 Pu weight is 1 and the 238 U one is 0.

The weight Wi of one isotope is defined by the relation :

$$Wi = \frac{\sigma^+_i - \sigma_{238U}}{\sigma^+_{239Pu} - \sigma^+_{238U}} \qquad \text{with } \sigma^+_i = \nu\sigma_{Fi} - \sigma_{ai}$$

Then, the equivalent enrichment E is expressed versus the Pu total enrichment in atoms E' and the equivalent weight Wi for Pu isotopic composition ξ_i by the relation :

$$\varepsilon = E' \sum_i \xi_i w_i$$

Figure 12 represents the Wi variations versus the r parameter for enrichment changes. The values of the different isotopes weights are present table 6 for a 1200 MWe typical core.

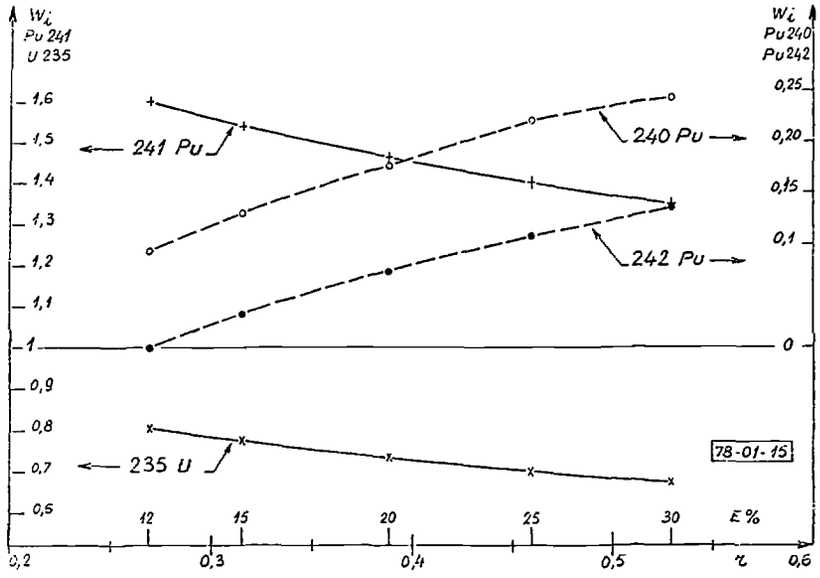


FIG. 12. VARIATION OF THE WEIGHT W_i IN EQUIVALENT 239 Pu VERSUS THE τ PARAMETER (ENRICHMENT VARIATION)

Table 6 - Weight W_i in 239 equivalent Pu of one atom of the various isotopes

	250 MWe	1200 MWe
235 U	.701	.777
238 U	0.	0.
239 Pu	1.	1.
240 Pu	.222	.130
241 Pu	1.400	1.542
242 Pu	.108	.032
r	. 459	. 315

4.1.2. Plutonium production

a) Breeding gain

The plutonium production in FBR's is characterized by the breeding gain GRG defined as the ratio of the net 239 Pu equivalent balance (production minus consumption) over the whole fissions :

GRG : [(production - absorption) of 239 Pu equivalent] /Fissions

$$GRG = \frac{\sum_{i=1,I} (C_i w_{i+1} - A_i w_i)}{\sum_{i=1,I} F_i}$$

with i = 1, I corresponding to the wole heavy isotopes

c_i, a_i, f_i capture, absorption and fission rates for the isotopei integrated on the whole reactor

W_i 239 Pu equivalent weight of the isotope i.

If the numerator integration is performed either on the core or on the blankets, one deals respectively with the internal breeding gain GRI or external breeding gain GRE with the obvious relation :

GRG = GRI + GRE

The orders of magnitude of the breeding gains are given in the following table for two typical reactors :

	250 MWe	1200 MWe
GRI	- 0,46	- 0,16
GRE	+ 0,58	+ 0,40
GRG	+ 0,12	+ 0,24

It must be mentionned again that fast reactors are only breeders due to the blankets.

b) Pu production

Knowing that one gram of 239 Pu produces 1 MW day, it is rather simple to calculate, for a given reactor power W_{th} and a given time, the Pu net balance in a reactor with a given breeding

gain GRG. Table 7 presents, over a period of one year, the mass of Pu burnt and produced in a 1200 MWe typical reactor with a 75 % load factor and three radial blanket subassembly rows.

One can notice the decrease in the radial blankets Pu production versus the distance to the core centre due to the flux attenuation. The difference between the upper and lower axial blankets comes from the control rod absorbers in the upper axial.

c) Doubling time

The Pu excess available in FBR's cycle is characterized by the doubling time.

The linear doubling time TDL is defined as the time necessary for one reactor to produce in excess the Pu inventory needed to operate a new identical reactor. Then, TDL is proportionnal to the Pu cycle inventory MC for one reactor and inversely proportionnal to the breeding gain :

$$TDL \propto \frac{MC}{GRG}$$

The Pu cycle inventory MC depends not only of the in pile inventory but also of the out of pile inventory (transport, storage, reprocessing, refabrication,...).

The combined doubling time TDC is defined as the time needed for several fast reactors to produce the Pu inventory needed to operate a new one :

TDC = Log 2 TDL

For a typical equilibrum FBR's situation and 1200 MWe typical plants (GRG = 0,24) the values are respectively :

TDL $\simeq$ 30 years

TDC $\simeq$ 20 years

It must be noted that the doubling time is extremely sensitive to breeding gain values :

GRG = 0,40, TDC = 12 years.

4.2. Variations of cell characteristics versus burn-up

a) There are three mains problems related to the neutronic burn-up during the operating cycle :

- variation of 239 Pu equivalent enrichments and of the fuel heavy isotope volumic percentages
- variation of the isotopic compositions
- fission product effect.

Due to the high burn-up aimed at (100.000 MWD/T or 12 % fission burn-up), these problems remain rather complex, but fortunately there is no strong spectrum effect.

For the core burn-up, the internal breeding gain plays a major role and can be easily related to the 239 Pu equivalent enrichment variation.

Table 7

Plutonium balance in a typical 1200 MWe reactor in kilogramme for a one year operation time

	Burn-up	Production	Net balance
core zone 1	540	490	- 50
CORE core zone 2	360	280	- 80
core	900	770	- 130
BLANKETS			
Upper axial	-	60	
Lower axial	-	120	
Axial	-	180	+ 180
Radial zone 1	-	90	
Radial zone 2	-	40	
Radial zone 3	-	20	
Radial	-	150	+ 150
Blankets	-	330	330
REACTOR	900	1100	200

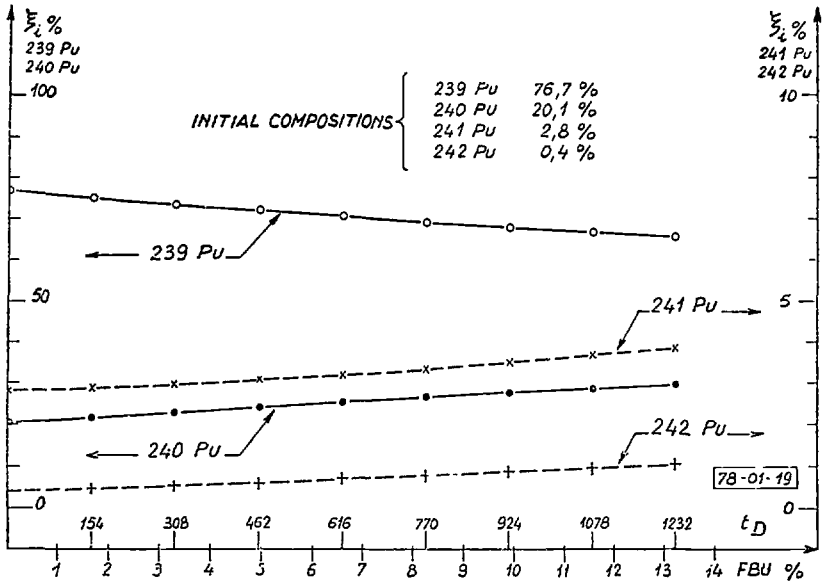


FIG. 13. *Pu ISOTOPIC COMPOSITION VERSUS BURN-UP (FBU IN %)*

The Pu isotopic compositions vary with burn-up : the 239 Pu composition decreases, the 240 Pu accumulates (fig. 13).

Taking the hypothesis, in some years, of the Pu autorecycling in FBR's, the equilibrum Pu isotopic compositions will be :

	239 Pu	240 Pu	241 Pu	242 Pu
Pu core recycling alone :	58 %	34 %	5 %	3 %
Pu core + blanket recycling :	70 %	25 %	3 %	2 %

This demonstrates an important FBR's characteristic : the Pu containing higher isotopes as 242 Pu that represent a big penalty for thermal reactors is replaced by better Plutonium specially in the blankets by 238 U capture.

Among the fission products isotopes, there is no strong poison as in thermal reactors : the average FP cross section is about 0,5 barn/fission.

b) Finally, at the reactor level, the reactivity loss per cycle ρ due to enrichment, isotopic composition variations and fission product effects depends on the spectrum and the reactor power. It can be divided into two terms, the fuel effect and the FP effect :

	250 MWe	1200 MWe
ρ fuel	75 %	25 %
ρ fission products	25 %	75 %
ρ % for 1 year	6 % in keff	3 % in keff

Due mainly to the less negative internal breeding gain in a 1200 MWe typical reactor, the heavy isotope reactivity effect remains largely smaller than in a 250 MWe reactor : the absolute global reactivity loss remains also smaller.

The power distributions vary also during burn-up due to the two enrichments zones and to the flux distribution. The characteristics of the plant are chosen to limit the power flatness variation during the cycle, especially radially, around the optimal value. The remaining power flatness variation is controlled usually by absorber control rods movements.

4.3. Consequences for the out of pile cycle

The fuel compositions in heavy atoms and fission products corresponding to the in-pile neutronic burn-up define a large part of the out of pile fuel cycle problems : handling, storage, transport, reprocessing, fabrication, waste.

It will only be possible here to mention the main aspects :

The residual power comes mainly from fission products ($\geq$ 80 %) and for a minor part from heavy isotopes ($\leq$ 20 %), especially from the α emission of curium isotopes for cooling times longer than 5 days.

The neutronic emission of the unloaded subassemblies, important for the shielding problems in the transportation and handling and for the operation is mainly due to the spontaneous fissions of the Cm isotopes (70 %) and to the (α, n) reaction on the oxygen.

For the fuel recycling and refabrication, several heavy isotopes have a specific influence : 241 Am strong γ producer, 238 Pu for the α emission, (α, n) reaction, 236 Pu and 232 U leading to hard gamma producers.

Finally, for the long-term-waste problems related to the long period α transactinides (241 Am, 243 Am, 242 Cm, 244 Cm), fast breeders present a good advantage compared to thermal reactors. All these isotopes have significant fission cross sections in a fast reactor spectrum. Taking into account the flux level, they can be largely transformed by fission in shorter period wastes (fission products).

5. CONCLUSION - FUTURE PLANS

This short presentation of fast breeder physics can only deal with some major points of that large problem. It would have been necessary to talk about cinetics (short prompt life time, β eff), dynamics (reactivity feed back, doppler effect, temperature and power coefficients), sodium void coefficients, physics aspects related to safety, ...

5.1. However, the major aspects covered here-above and the variations presented for the main cell parameters versus the design core data help to understand the FBR's specific characteristics and to clarify the physical parameters the most important for design optimisation, operation and safety.

To reach the accuracies requested by the design on these characteristics, fast reactor physics use first nuclear physics and mathematical physics, second, and mainly, integral experimental results issued from zero power facilities, for example MASURCA in FRANCE (fig. 14) or from operating power reactors, for example PHENIX (fig. 15). The results of that parametric integral experimental programs plays the leading role in the improvement of calculationnal physical systems used for the design studies.

5.2. Reactors physics aim also to improve the present core performances, especially to increase the breeding characteristics that represents the main FBR's advantage.

In that direction, a new core concept, including blanket zones inside the fissile core zones is being studied. The so-called heterogeneous concept is developped along the following lines.

To increase the breeding gain in the classical homogeneous concept, the first idea consists in decreasing the enrichment in an homogeneous way. However that solution leads to such an increase in the critical mass, that finally the

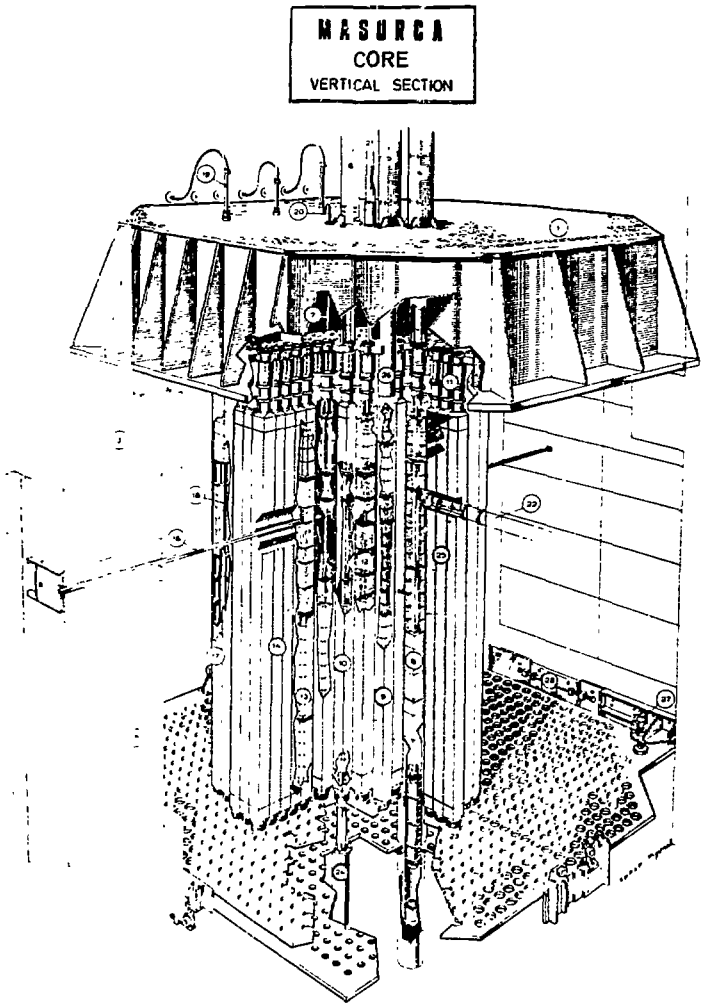
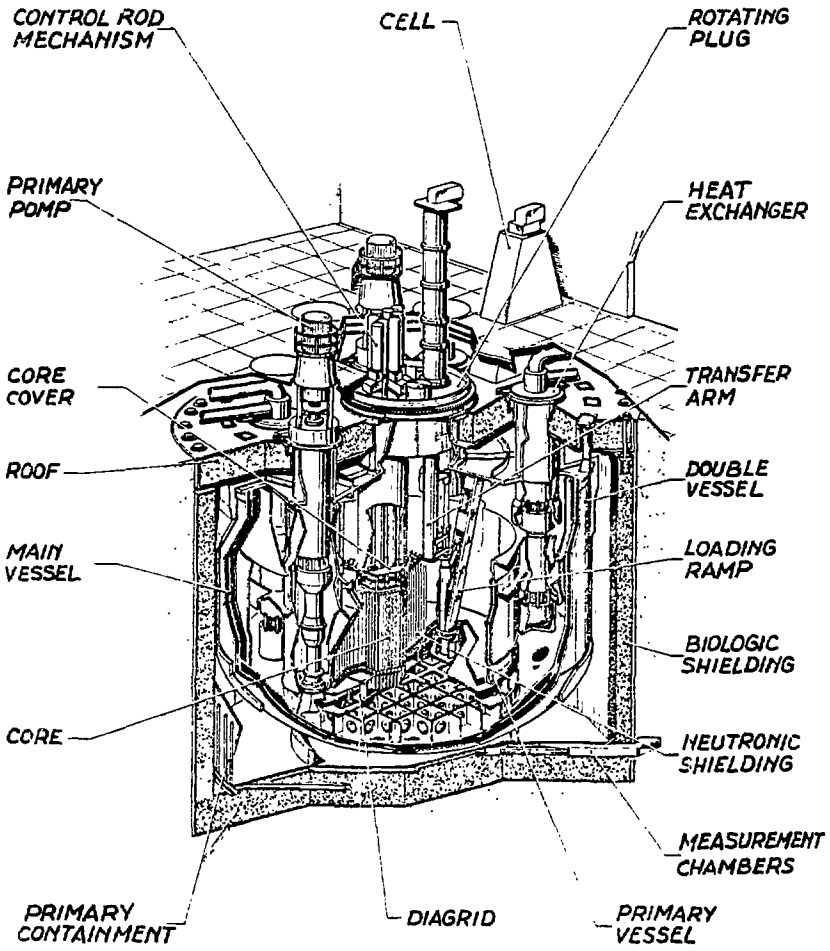


FIG. 14.



PHENIX 250 MWe : BLOCK REACTOR

FIG. 15.

doubling time (see § 4.1.), the only important parameter for breeding, increases largely. But, if an heterogeneous distribution of fertile zones inside the fissile ones is performed, there exist optimal solutions leading to a minimum ratio of the enrichment to the global breeding gain, the enrichment of the fissile zone being of the order of 30 %. From a physical point of view, the main effect comes from spectrum differences in the internal blankets and in the fissile zones compared to the homogeneous spectrum :

- -the average 238 U capture cross section increases
- -the average 239 Pu η parameter increases.

If the geometrical distribution of the internal blankets is optimized to improve the radial power flatness, for a given total power, the core volume then the doubling time can be minimized. The following table compares relatively two 1200 MWe reactors on the homogeneous and heterogeneous basis :

	Homogeneous	Heterogeneous
In-pile critical mass tons Pu	3,2	3,7
GRG	0,2	0,35
Combined doubling time years	22	13

Outside the doubling time improvement, the heterogeneous concept presents also other potential advantages :

- decrease of the reactivity loss per cycle due to the GRI improvement, leading to decrease the number of control rods
- decrease of the flux level due to the high fissile zone enrichment, then possibility for the same fluence to have higher burn-up
- improvement of the sodium void coefficient for safety.

5.3. If the performances of such an heterogeneous concept are confirmed by all the experimental programs carried out now, it is reasonnable to think that fast breeders reactors using the well-known mixed oxide fuel will potentially be ready to cover, around the end of this century, a larger part of the new electricity needs.

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