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PRESENT STATUTS OF THE NEUTRONIC STUDIES OF FAST REACTOR FERTILE REGION AT CEA

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I - INTRODUCTION -

An increasing effort has been devoted at CEA to a better assessment of the neutronics characteristics of the fertile regions of fast reactors (both external, radial and axial, blankets and internal fertile regions of the heterogeneous concept).

In this paper, we present some recent developments in this field related to :

- blanket properties influence on the prediction of the critical mass ;
- cross-section definition for blanket regions ;
- diffusion/transport methods for neutron propagation in blankets ;
- experiments planning to check calculational methods.

II - CRITICAL MASS AND BLANKET PROPERTIES -

The influence of the blanket properties on the critical mass assessment has been verified in the analysis of several critical experiments performed on MASURCA. In particular, it has been possible to measure the blanket "reflector saving" by means of reaction rate distribution measurements. Radial and axial flux curvature were derived by a standard procedure /1/, and then it was possible to derive the blanket "reflector saving", using the critical size of the assembly. In the following table, a number of results are shown, in terms of the deviation between 2D (R,Z) calculation and experiment, for a number of critical assemblies of the R,Z programme in MASURCA /2/, and some SNEAK assemblies studied in the framework of the KFK/CEA cooperation :

	ASSEMBLY	E-C/C
MASURCA	R1(U235 core, 22% enrichment)	- 0.9 ± 1. %
	R3(U235 core, 15% enrichment)	- 2.4 ± 1. %
	Z1(Pu239 core, 18% enrichment)	+ 0.3 ± 1. %
	ZONA1(Pu239 core, 18% enrichment)	+ 1.0 ± 2. %
SNEAK	9A(U235 core, 20% enrichment)	+ 2.5 ± 1. %
	9B(Pu239 core, 16% enrichment)	+ 2.0 ± 1. %

All the blankets were UO_2/Na blankets with approximately 50/50 v/o. All calculations used the CARNAVAL IV 25 group cross-section set. When transposed to a large power reactor, the discrepancies observed and the associated uncertainties, result in bias factors on the critical mass, with associated uncertainties. The order of magnitude of these bias factors : $\approx [250 \pm 200]\ 10^{-5}\ \frac{\Delta K}{K}$ confirms that the CARNAVAL IV system is well adjusted for the fast power plant design.

Reaction rate distributions calculations (U238 and U235 fission, U238 capture), were also compared to experiments. Some typical results are shown un the following table, where (E-C)/C values refer to the first 10 cm in the UO_2/Na blanket :

ASSEMBLY	F8	F5	C8
R1	- 2 ± 1.5 %	- 2 ± 1.5 %	- 4 ± 2 %
R3	- 8 ± 1.5 %	- 4 ± 1.5 %	- 6 ± 2 %
ZONA 1	- 3 ± 1.5 %	- 2 ± 1.5 %	- 4 ± 2 %

Further improvement is expected with a more refined method to treat core/blanket interfaces.

III - BLANKET CROSS-SECTION GENERATION -

The main improvements related to the blanket cross-section preparation have been the following for the CARNAVAL IV system :

- heterogeneity effects are taken explicitely into account in the calculation ;
- the calculation of the neutron moderation makes use of the leakage spectrum of the core to obtain a correct weighting spectrum /3/ ;
- leakage effects on the self-shielding is accounted for by means of a simple prescription, i.e. introduction in the background calculation of a $D(\frac{\pi}{a})^2$ term where "a" is the blanket thickness. Refined space dependent calculations have verified the accuracy of this approach.

For that concerns the resonance mismatch at the core/blanket interface, work is in progress to eliminate self-shielding discontinuity at this interface.

Fine group space dependent calculations have been performed /4/ to test space dependent effects in the internal fertile regions of the heterogeneous concept.

IV - DIFFUSION/TRANSPORT CALCULATIONS OF NEUTRON PROPAGATION -

Both the strong degradation of the neutron spectrum through an external blanket of 30÷50 cm thickness approximatively, and the interface problems, both core/blanket and blanket/shield, are difficult to be treated with diffusion theory in many cases of interest.

Morever, complete transport calculations, based on S_N methods or other, can still be too heavy to be carried out in a standard way.

Therefore, there has been increasing interest in the use of improved approximations of the transport equation, the so call "anisotropic" diffusion methods. At CADARACHE, one original method has been developped /5/, and is being tested. Numerical comparison of the relative performances of diffusion, transport and "anisotropic" diffusion indicate an improved agreement between the last two methods, when reaction rates distributions are compared at a core/blanket interface. This is shown in the following table, related to slab geometry and to typical fast core and blanket compositions :

	Distance from interface (cm)	U 238 fission rate	
		(DIFF-TRANSP)/TRANSP	(ANIS.DIF-TRANSP)/TRANSP
Core	- 3	- 2.91	- 0.28
	- 1.5	- 2.38	- 1.32
	0	3.74	- 2.07
Blanket	+ 1.5	8.42	- 1.20
	+ 3	11.49	- 0.60
	+ 4.5	11.73	- 0.37

More tests are under way for a wider range of cases of interest.

V - EXPERIMENT PLANNING FOR BLANKET STUDIES -

To test method improvement and basic data, an experimental programme has been planned. According to a standard procedure at CEA, a parametric series of experiments has been planned, the main parameters being :

- core spectrum into the blanket ;
- core spectrum from blanket into the shield ;
- UO_2 / structural material ratio in the blanket ;
- SS / Na ratio in the blanket.

The programme will be performed at the fast source reactor TAPIRO (CASACCIA), in the framework of the CEA-CNEN Fast Reactor Physics Agreement.

A number of situation of interest for large fast power reactors will be covered, with the variation of the main parameters chosen.

As an example, Figures 1 and 2 show the capture and fission rate distributions of U 238 in the radial and axial (over inner core) blankets of SUPER-PHENIX 1, compared to the same distributions in some of the foreseen configurations. Details on these configurations can be found in the following table :

Configuration	Blanket composition (v/o)
C1	UO_2/Na 50/50 ; Na/SS 4/1
C2	UO_2/Na 30/50 ; Na/SS 4/1
C3	UO_2/Na 50/30 ; Na/SS 2/1
C4	End of cycle composition

All these configurations have the same source spectrum (typical of the outer core of SUPER-PHENIX) and a Na/SS (50/50) shield. The "end of cycle" configuration will have a ∿ 3% fissile fuel build-up simulation. Further configurations will have both different source spectra (a softer spectrum, typical of the inner core of SUPER-PHENIX, and a harder spectrum, related to the higher enrichment of the heterogeneous core concept), and different shield compositions.

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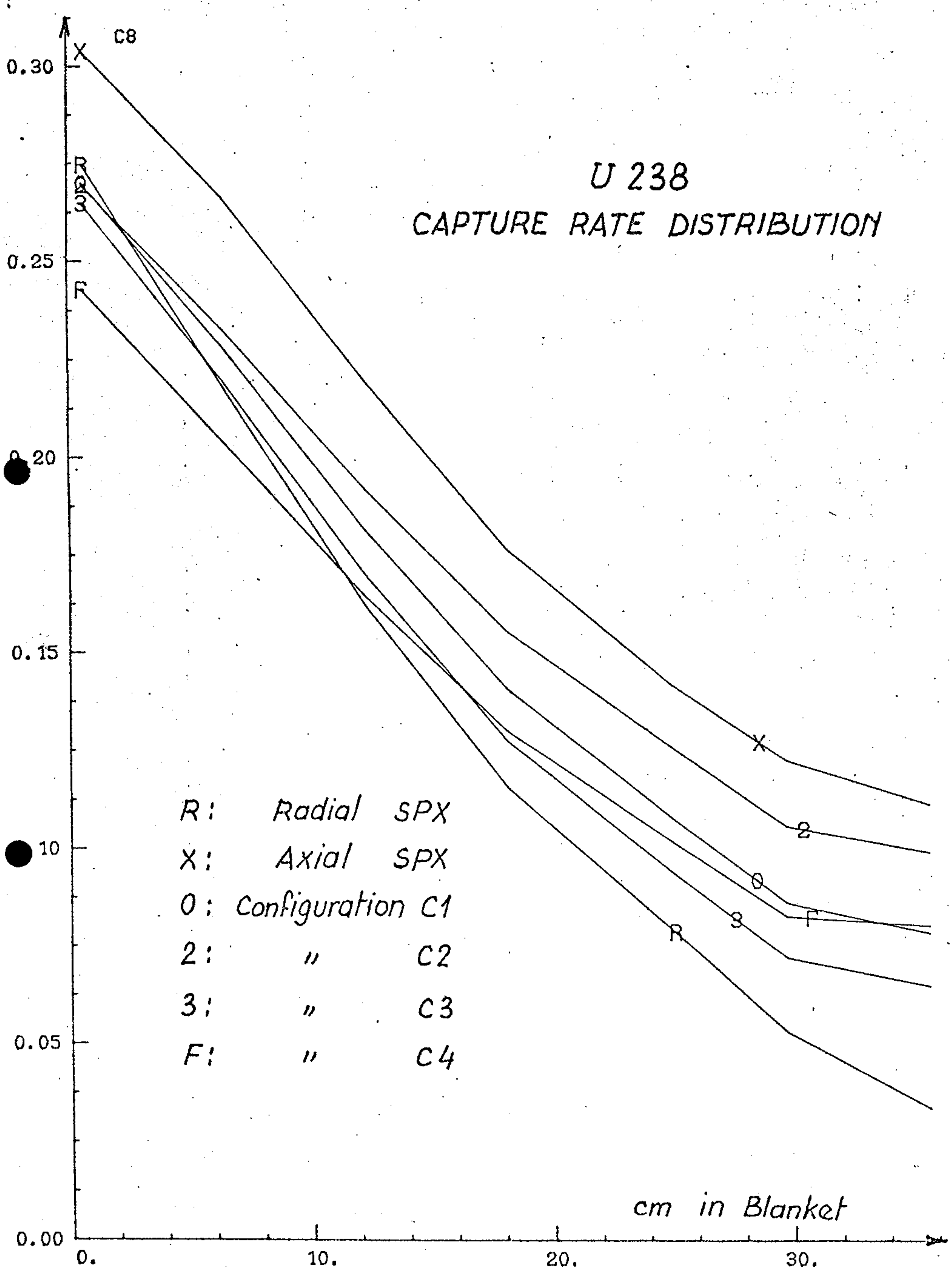

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