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PROGRESS IN FUSION REACTORS BLANKET ANALYSIS AND EVALUATION AT CEA

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PROGRESS IN FUSION REACTORS BLANKET ANALYSIS AND EVALUATION AT CEA

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ABSTRACT

In the frame of the recent CEA studies aiming at

- the development, evaluation and comparison of solid breeder blanket concepts in view of thei adaptation to NET,
- the evaluation of specific questions related to the first wall design,

the present paper examines first the performances of a helium cooled toroïdal blanket design for NET, based on innovative Beryllium/Ceramics breeder rod elements. Neutronic and thermomechanical optimisation converges on a concept featured by a breeding capability in excess of 1.2, a reasonnable pumping power of 1 % and a narrow breeder temperature range (470 ± 30 °C of the breeder), the latter being largely independant of the power level. This design proves naturally adapted to ceramic breeder assigned to very strict working conditions, and provides for any change in the thermal and heat transfer characteristics over the blanket lifetime.

The final section of the paper is devoted to the evaluation of the heat load poloidal distribution and to the irradiation effects on first wall structural materials.

1. A HELIUM COOLED TOROIDAL BLANKET FOR NET, BASED ON COMPOSITE BERYLLIUM/CERAMICS BREEDER ROD ELEMENTS

1.1 Introduction

Previous CEA helium cooled solid blanket design studies had converged on a rods blanket concep of attractive performances (TBR > 1.45, breeder operating temperatures : 500 ± 20 °C), based on original composite Beryllium/LiAlO$_2$ (85/15 %) breeder rods, arranged in radial canisters [1] Optimised for DEMO relevant boundary conditions, this concept was found to be seriously impaired when adapted to devices with narrow breeding zones such as NET [2].

The present design, which takes advantage of the promising breeding properties of the same composite breeder elements but associates them with a multipasses toroïdal cooling scheme, is developped as a alternative to the radial canister concept for blankets of restricted thicknes

1.2 Design parameters and blanket architecture

The NET III B version of near term device [3] was adopted as boundary conditions for this study ; the associated design constraints are briefly summarized in table 1.

The general layout of the blanket sectors is illustrated on Figure 1 : it consists of rows of in series cooled tubular breeder modules arranged in the toroïdal direction, and containing composite Beryllium/Ceramics rods.

TABLE 1 [illegible]

[illegible]	[illegible]	[illegible] : 1.5/0.95 [illegible]
[illegible]	[illegible]	[illegible]
[illegible]	[illegible]	[illegible]
[illegible]	[illegible]	[illegible] [illegible] 0.35 + 0.08 m [illegible] 0.65 + 0.30 m
[illegible]	[illegible]	[illegible]
[illegible]	[illegible]	[illegible]
[illegible]	[illegible]	VERTICAL BLANKET REPLACEMENT

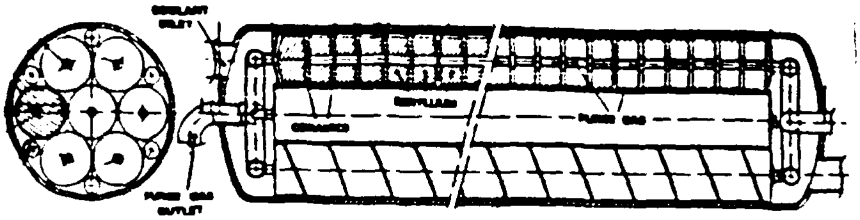


FIG.3 TUBULAR MODULE TYPICAL OF 2nd,3rd,4th ROWS

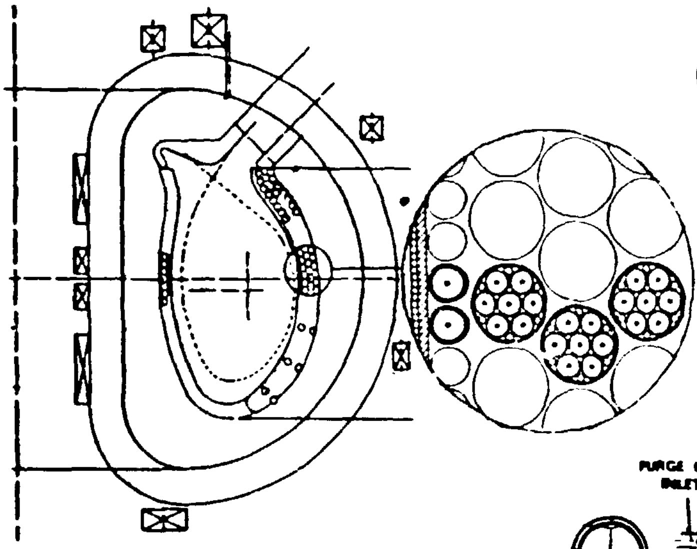


FIG.1 BLANKET GENERAL LAYOUT

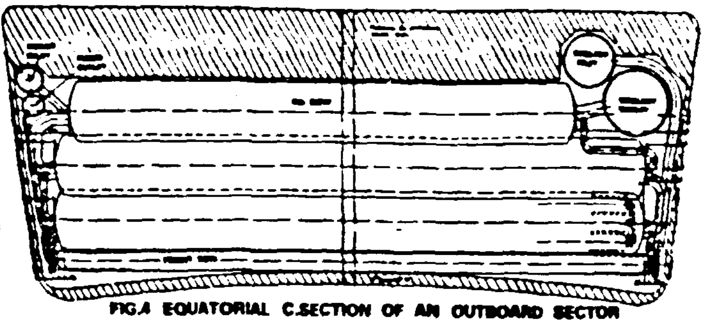


FIG.4 EQUATORIAL C.SECTION OF AN OUTBOARD SECTOR

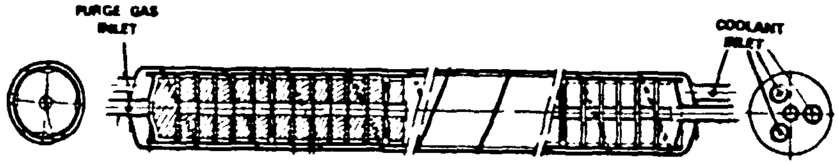


FIG.2 TUBULAR MODULE TYPICAL OF THE FRONT ROW

- The composite Beryllium/Ceramics breeder rods

. The breeder rods are composed of an alternate stack of Beryllium and γ $LiAlO_2$ hollow pellets separated by 0.2 mm steel sheets and gathered into the same clad equipped with wire wrap spacers (Figure 2). The thin steel sheets are intended to prevent oxidation of Be by the adjacent caramics. Steel whashers maintain the contact pressure between adjacent pellets and accommodate the anticipated Be swelling. A low pressure purge gas enters the central cavity of each rod, is distributed by sweeping the grooved surface of the LiAlO2 pellets, and finally collected in large grooves machined in the Be pellets outer surface.

. The neutronic optimisation of the breeder composition [4] leads to the respective volumetric proportions of 15 % γ $LiAlO_2$ (60 % enriched in ^6Li) and 85 % of Be, the latter acting simultaneously as moderator and neutron multiplier.

. This innovative breeder rod concept allows to make use of the excellent thermal conductivity of Beryllium to flatten the radial temperature gradient ΔT in the breeder material, making it possible to consider rods in excess of 7 cm in diameter, and hence to minimize the proportion of cladding material and the associated neutron parasitic captures, whereas pure ceramics rods with diameters in excess of 2.5 cm placed in the front region would not meet usual blanket operating conditions ($\Delta T \sim 200$ °C, $P_n \sim 1.0$ MW/m^2).

- Tubular modules arrangement and coolant routing scheme

The outboard blanket contains 1 row of small diameter tubes (92 mm) and 3 rows of larger diameter modules (190 mm), which are 2. to 2.5 meter long. The difference in module size and configuration of both row types is intended to accomodate the steep decrease of the heat deposition, as the distance from the plasma increases.

- Figure 2 shows the cross-section of a front row module. The pressurized coolant flows in an annulus between the SS 316 L module envelope and the cladding of a only breeder rod (80 mm in

- Figure 3 illustrates the cross section of a back row : each tubular module contains a bundle of 7 composite breeder rods 57 mm in diameter. 6 bare rods of Beryllium are inserted at the bundle periphery to improve both the heat transfer and the filling of the module. The Helium coolant flows in the gaps between the container and these 13 rods held in place by helical wire spacers.

- The available blanket thickness restricted to 1 of both types the number of breeder module rows on the inboard side.

- Figure 4 is an equatorial cross section of the outboard blanket. It illustrates the principle of the helium coolant and purge manifolding : Coolant and purge gas enter the tubular modules at opposite ends, sweep the breeder pins externally and internally in a countercurrent flow pattern and are finally collected at both opposite end and ducted towards the next breeder row. 2 adjacent tubes of the front row are connected in series with a single tube of each other breeder rows. The main headers, that run in the poloïdal direction, are integrated into the back shielding and supporting structure of the blanket. The first wall is assembled with the back structure so as to contain the breeder modules in a actively pumped closed box that acts as a secondary vacuum chamber.

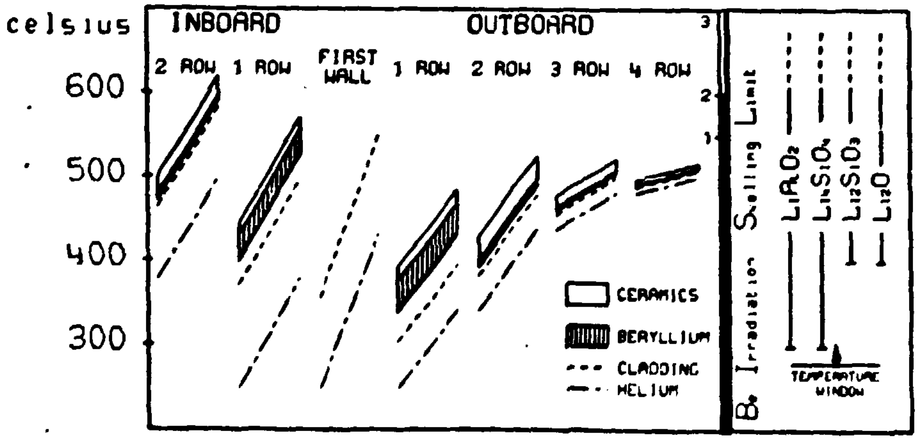
1.3 Blanket performances

1.3.1 Thermal hydraulic performances

- Figure 5, drawn from 2D thermal calculations assuming Helium working conditions of 6 MPa and [250-500]°C, summarizes operating temperatures throughout the blanket at nominal power, while Figure 6 presents for various wall loadings the proportion of the breeder working below a given temperature.

- These results prove the efficient breeder temperature control afforded by the adopted design which combines the advantages of a quasi radial cooling direction, where the coolant temperature rise compensates for the decrease of the heat generation within the bulk of the blanket, and the excellent thermal conductivity of Beryllium that, when mixed with the breeder as proposed, flattens radial temperature gradients in 5 to 8 cm in diameter breeder pellets.

FIG. 5 BLANKET OPERATING TEMPERATURES AT PN = 0.95 MW/M**2



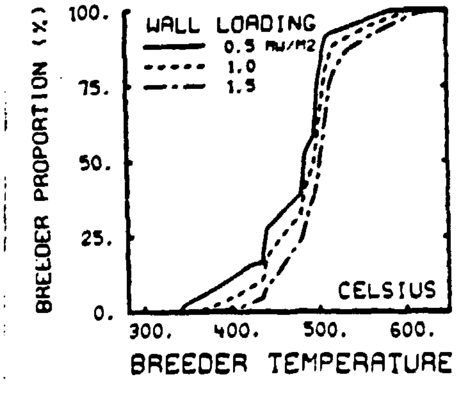


FIG. 6 PROPORTION OF THE BREEDER WORKING BELOW A GIVEN TEMPERATURE, FOR 3 NEUTRON WALL LOADINGS.

-The proposed design is privilegedly adapted to ceramics breeders assigned to strict working conditions, since only about 5 % of the breeder is operated at temperatures below 400 °C, the lower limit for an efficient tritium release from most breeder candidates, and this over a wide range of wall loadings. It has the potential to accommodate blanket power swings and poloïdal variations of the wall loading, or to compensate for any variation in thermal and heat

→ The beryllium temperature is moderate and essentially stays below the swelling threshhold predicted by the BEESTON model [5] as a function of the fast neutron fluence and of the subsequent helium production.

- A total pressure drop of 0.06 MPa along the cooling lines (tubular modules + poloïdal headers), calculated for a uniform neutron wall loading of 1.5 MW/m^2 (corresponding to an average value of 0.95 MW/m^2 with a peak factor of 1.58) requires a reasonnable but not negligible pumping power of 1 % the removed thermal power.

1.3.2 Neutronic performances

Nominal performances and comparison with the radial canister concept

- The tritium breeding ratio afforded by the current section of this blanket, was estimated with a 1D-cylindrical representation of both inboard and outboard blanket segments simultaneously ; $LiAlO_2$, 60 % enriched in ^6Li is selected for the calculation. In these conditions, the estimated breeding capability with full coverage, amounts to 1.21 with respective contributions of 0.95 for the outboard and 0.26 for the inboard segments. Optimizing the shielding design, is expected to somewhat relax the constraint upon the thickness of the breeding zone and to thus improve the breeding performances by a few per cents.
- The error on TBR associated with ENDF/B4 nuclear data uncertainties calculated with use of sensitivity profiles and covariance matrices, amounts to 1.4 %, with respective contributions of ^6Li elastic : 1.3 %, ^7Li elastic : 1 %, Be elastic : 0.2 %, Be (n,α) : 1.2 % and Be (n,2n) : 3.7 %.
- This toroïdal blanket arrangement proves better adapted to breeding zones of restricted thickness, compared (table 2) with the radial canister concept [1,2] previously developped by CEA for DEMO and adapted to NET. Based on the same Be/Ceramics breeder elements, both designs exhibit a breeding potential significantly improved, compared with conventional Helium cooled rod blankets (TBR ∿ 1.04 for $LiAlO_2$/Be) ferritic steel concept considered in the BCSS [6]).

	RADIAL CANISTER CONCEPT [1,2]		PRESENT TOROIDAL CONCEPT
	DEMO	NET	
FIRST WALL	1 cm GRAPHITE + 0.3 cm SS	1.2 cm SS + 0.3 cm SS	1.5 cm GRAPHITE + 1.5 cm SS
BREEDING BLANKET	45 - 50 % { 85 % BERYLLIUM, 15 % $LiAlO_2$ (60 % ^6Li, 80 % TD) } 15 % SS STRUCTURE		
	0.7 m THICK BREEDER ZONE	0.55 m INBOARD + 0.85 m OUTBOARD	
TBR WITH 100 % COVERAGE	1.52	1.08 (0 + 1.08)	1.21 (0.26 + 0.95)
	3 D MONTE CARLO 100 % NEUTRON REFLEXION	1 D ANISN INBOARD + OUTBOARD	

TABLE 2 - COMPARISON OF NEUTRONIC PERFORMANCES OF THE RADIAL CANISTER CONCEPT [1,2] WITH THE PRESENT TOROIDAL DESIGN.

Sensitivity of neutronic performances to design options

- Breeder pellet size : Comparison between 1D, 2D and 3D calculations demonstrates that for the proposed dimensions, 1D homogeneous methods give TBR results with an accuracy of 5 % [4].
- Steel content : The TBR proves strongly sensitive to the steel volume fraction of the blanket with rates of decrease of - 0.75 % per % de S.S. and - 1.4 % per mm of S.S. in the first wall.
- Effet of depletion : The low breeder content (8 %) results in a average depletion rate of - 5 $\%/MWy/m^2$ and a peak value of - 6.9 $\%/MWy/m^2$, implying a TBR degradation rate of - 0.4 $\%/MWy/m^2$. The subsequent burn up effects on the ceramics mechanical integrity could be alleviated, if necessary, by a slight departure from the optimum Beryllium and ceramics volume fraction, this, with little consequence on the TBR (table 3).

BERYLLIUM/$LiAlO_2$ (60% ^6Li, 80% TD)	$LiAlO_2$ INVENTORY	$\frac{\Delta TBR}{TBR}$ (%)
0.85/0.15	1	—
0.8 /0.2	4/3	- 1.2
0.75/0.25	5/3	- 4.2
0.7 /0.3	2	- 8.8

TABLE 3 - INCIDENCE ON THE TBR OF A SLIGHT DEPARTURE FROM OPTIMUM Be/$LiAlO_2$

- Dependence on NET configuration : Performances of the toroïdal arrangement appear to be seriously impaired by an augmentation of the blanket segmentation from 24 to 32 or 48 toric sectors, thus increasing the relative fraction of the toroïdal lengh needed for piping connections between the modules, and leading to a TBR decrease by 6 % and 14 % respectively.

1.3.3 Tritium recovery

A low pressure (1 bar) Helium is used to collect the tritium. Its 100 ppm H_2 charge is intended to ease tritium release from the ceramics as suggested in [7]. This purge gas enters the blanket by the rear row modules and leaves by the front row , flowing countercurrent with the cooling helium. This routine scheme minimizes the tritium permeation to the coolant, which rate is kept below 0.1 gT/d by flowing the purge gas at a rate corresponding to an exit partial pressure of 0.2 Pa of HT.

1.4 Conclusion

The present work, performed within the framework of the European Fusion Technology Programme, examines the potential interest for NET of a blanket design which combines the attractive thermal performances of helium cooling in a quasi radial (toroïdal multipasses) direction, with the promising breeding capability of composite Beryllium/$LiAlO_2$ breeder elements. This toroïdal blanket design is proposed as an alternative to the radial canister design, - based on the same breeder elements and previously developped by CEA for the DEMO-,to permit a more credible adaptation to NET. The optimisation of the neutronic and thermomechanical performances converges on a concept featured by a breeding capability in excess of 1.2, a reasonnable pumping power of 1 % of the thermal power and a narrow breeder temperature range (470 ± 30 °C for 80 % of the breeder) and largely independent of the power level. This unique feacture is well adapted to ceramics assigned to very strict and narrow working conditions, and provides for any change in the thermal and heat transfer characteristics over the blanket lifetime. In return, critical issues relate to the great number of tubes and connections, which questions the system reliability, and to the use of beryllium in internally purged composite breeder rods. An experimental programme to ascertain beryllium resistance to radiation damages and to corrosion by adjacent materials or atmospheres is a requisite for the thorough investigation of this blanket concept and for the assessment of its viability.

2. EVALUATION OF IRRADIATION EFFECTS ON FIRST WALL MATERIALS

Effects of irradiation on structural materials are of importance for the design of the first wall, since materials activation generates residual afterheat, which has to be taken into account when dimensionning the F.W. cooling circuit, and induces large γ doses long after shut down, while changes in components concentrations can alter mechanical performances of metallic alloys.

The activities of a 304 SS first wall protecting a NET III blanket sector equipped with either the hereabove presented CEA helium cooled solid breeder concept, or the NET $Li_{17}Pb_{83}$ concept, were calculated and found of comparable level (figure 7). For this purpose, a new library of neutron cross-sections has been generated from UKCTR III A updated with ENDF/B5.

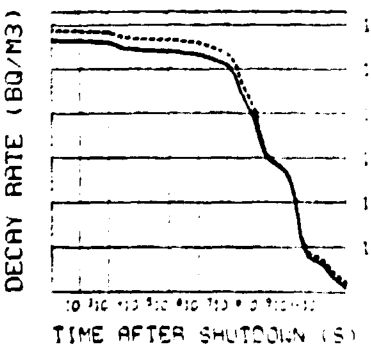


FIG.7 DECAY RATE AFTER [illegible]

N_i63, M_o93, N_i59 give rise to a high long term (> 1 century) activity level, which is an incentive to develop low activity versions of conventionnal stainless steels in which N_i and M_o are replaced by another component such as Mn.

3. POLOÏDAL DISTRIBUTION OF VOLUME AND SURFACE HEAT LOAD

A lot of factors, among them geometry of the plasma chamber, shape and location of the plasma in this chamber, induce non uniformity in the heat load distribution along a poloïdal contour of the first wall or blanket.

3D Monte Carlo calculations (TRIPOLI 2 code [8]) performed on a toric sector of NET II A equipped with the Li_{17} Pb_{83} NET blanket concept reveal (Figure 8) poloïdal peak factors on the first wall of 1.6 and 1.35 for the surface (assuming α energy and 14 MeV neutron current are identically distributed) and volume (neutrons and γ energy) heat deposition respectively. These results are in good agreement with those of previous calculations carried out on the same configuration with different computational tools [3].

Heat extraction circuits and shieldings should be dimensioned in consequence, in order to limit hot spots temperatures and doses. This work is also a first qualification stepof the codes intended to be used in shield design studies for NET.

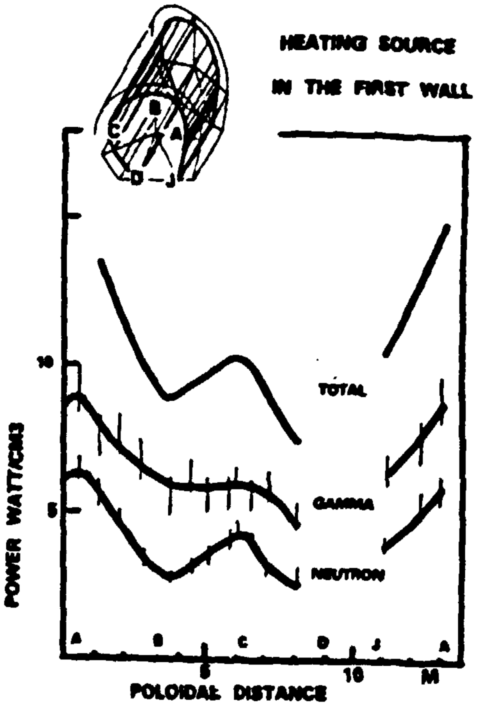


FIG.8 DISTRIBUTION OF SURFACE AND VOLUME HEAT LOADS ALONG A POLOIDAL CONTOUR OF THE NET II A FIRST WALL.

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