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NUCLEAR BLANKET AND SHIELDING PROBLEMS IN DEMONSTRATION FUSION REACTORS

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Abstract

The main results of nuclear parametric calculations carried out for the conceptual design of the blanket and shielding of the FINTOR reactor are presented.

The effect on tritium breeding of blanket and reflector thickness and composition is investigated.

Different neutron shielding materials are analysed by means of a simplified calculation method (SABINE). A D-shaped blanket is proposed.

Introduction

The conceptual design of a minimum size experimental fusion reactor of the TOKAMAK–type (FINTOR) is in progress within the framework of a collaboration agreement between CNEN-Frascati, JRC-Ispra and the University of Naples.

A paper presented to this Conference gives the basic features of this reactor ¹⁾, the main operating characteristics being reported in Table I.

TABLE I : FINTOR CHARACTERISTICS

Power	250	MW(th)
Plasma Radius	2.0	m
Aspect Ratio	5.0	
First Wall Loading	2.5	m
Neutron Wall Radius	0.177	MW/m²
Magnetic Field	B_t^o = 5	Tesla
	B_t^{max} = 8	Tesla
Coolant	Helium	
Structural Material	Stainless steel	
Magnets	Nb – Ti + Cu	

The present paper outlines the main results of the nuclear calculations carried out to define the blanket and shielding characteristics of the reactor.

General Criteria

By taking as reference the main features of the reactor listed in Table I, and owing to the fact that FINTOR is an experimental reactor of low power, intented for feasibility demonstration and for flexible operation, the following main criteria can be established for the choice of materials and the dimensions of blanket and shielding:

- since the helium cooling is extended to both blanket and shielding, the blanket dimensions are limited only by tritium breeding requirements and not by heat recovery in the lithium; the energy losses are in any case low (less than 1 ‰);
- since technical feasibility is the primary goal of the reactor, the economic features are not as important as they would be in the case of a full-scale reactor;
- materials which can easily be loaded into and replaced in the modular blanket and shielding arrangement, have to be chosen in order to provide alternative solutions and in particular the operation of the reactor without tritium production. Furthermore, low density materials favour the modules handling for maintenance and repair;
- provision should be made for a less than full-time operation of the reactor; the material behaviour has therefore been investigated on the hypothesis of a 0.3 loading factor, so corresponding – for a 10-years operation – to about 3 year equivalent life.

On this basis, the following choices have been made:

- ST-316 stainless steel for the first wall and structural materials;
- natural metallic lithium for tritium breeding. The recently proposed [2] lithium-aluminium alloys, associated to the use of beryllium, offer however, a very attractive alternative solution and deserve some attention;
- aluminium oxide powder for the blanket reflector, even if carbon and stainless steel are comparable from a nuclear point of view;
- boron carbide powder for the neutron shielding and a lead layer in the outside part for gamma shielding. Water and hydrides have been excluded for safety and thermal stability or material resource reasons whereas tantalum has not been considered due to its prohibitive cost.

The limiting geometrical parameters are imposed by plasma balance conditions and by the maximum magnetic filed in the D-shaped toroidal magnet. This corresponds to the need for a D-shaped lithium blanket configuration, with zero-thickness in the inner part of the torus (see Fig. 1); then we can have available a shielding thickness of up to 1.1 m and a total thickness (blanket + shielding) of 1.7 m in the outer part.

Other limiting factors are:

- a tritium breeding higher than 1.20;
- a peak energy deposition in the magnet of the order of 1×10^{-4} W/cm^3 for the magnet helium cooling requirements;
- a fast (> 0.1 MeV) neutron flux, acceptable from the radiation damage point of view in the magnet copper stabilizer (2.3×10^{-5} displacements per atom per year).

Calculation Methods

A modular code system has been set up to calculate the main nuclear parameters in blanket, shielding and magnet. Fig. 2 shows a layout of the method as well as of the nuclear input data and outputs.

The neutron and gamma transport calculations are performed in one-dimensional geometry, in 100, 21 energy groups respectively. P-3 approximation is taken for the scattering anisotropy. The neutron energy group structure is that of GAM-II.

As far as nuclear data libraries are concerned, the effort is oriented towards a reference to ENDF/B-3. However, it has so far been necessary in some cases to take other nuclear data sources, namely:

- (n, γ) production cross sections from POPOP-4 for the isotopes which are not included in ENDF/B-3;
- the neutron KERMA-factors (apart from boron and lead) are still from AVKER [3], due to the fact that the code MACK [4] has only very recently been made available at Ispra;
- the atom displacement production rates for stainless steel have been calculated from DORAN [5] compilation, whereas for copper the UK-NDL Library was taken as the nuclear data source of ARTUS X [6]. The possibility of adapting RICE [7] (based on the ENDF/B-1 file) to ENDF/B-3 is being considered at Ispra.

This calculation scheme is quite expensive in computing time. A preliminary effort is planned to optimize the neutron energy group structure both for tritium production and magnet shielding calculations. For shielding parametric studies, the possibility of replacing the S-n approach with more simplified models has been investigated. The SABINE code [8], set up at JRC-Ispra in the framework of fission reactor shielding studies, seems attractive for this purpose. In the code, neutron and gamma transport is treated by the removal-diffusion model. 26-neutron energy groups are used, the nuclear data coming from the GGC-II library. The gamma flux is obtained as the product of the uncollided flux times a region dependent build-up factor interpolated from a table of values calculated by the BIGGI-3 [9]) transport code.

Gamma flux for up to seven energy groups, separating the contributions of the different source regions, are produced. The code outputs provide nuclear heating and dose rates. The (n, γ) libraries now in the SABINE code are not adapted for CTR-calculations, due to the fact that no inelastic gamma sources are included. Work is in progress to replace this library by one produced by POPOP-4 [10]). The code has been extensively used on the shielding studies presented here. The reliability of these calculations has been checked by means of a comparison with ANISN. The results of this comparison for a typical magnet shielding arrangement, are shown in Fig. 3. It appears that SABINE overestimates the neutron attenuation up to the magnet boundary: this corresponds to an error in the total thickness of about 10%. For this configuration the computing time of the SABINE calculation is 4.4 minutes on IBM-370/165 (including γ-calculation) which has to be compared to about 1 hour in the ANISN calculation.

Parametric Study

Blanket Analysis

The following parameters have been investigated:

- blanket thickness,
- reflector material,
- reflector position,
- reflector thickness,
- stainless steel and void fraction in the blanket,
- blanket temperature.

The main results are presented in Figs. 4 to 8.
In Fig. 4 it appears that an increase of 30 cm of the thickness of the blanket corresponds to an increase of 10% of both contribution of Li-7 and Li-6 reactions and also of the tritium breeding. A saturation effects appears at 80 cm thickness.
Fig. 5 compares Al_2O_3 powder (density 3.2 g/cm^3) and stainless steel as reflector materials; stainless steel looks more favourable from the point of view of breeding as well as that of heat deposition and damage to the magnet. However, these effects are so slight that the choice between the two materials has to be made on the basis of other considerations (weight, activation, thermal conductivity, etc.). In the FINTOR design Al_2O_3 has so far been considered as reference blanket reflector.
Graphite looks quite similar to Al_2O_3 with a .5% lower tritium breeding.
The reflector position in the lithium blanket (Fig. 6) deos not particularly influence the tritium breeding, so it has been decided to place the reflector outside the blanket, which simplifies the mechanical and thermal design.
From Fig. 7 it appears that the saturation effect, due to the increase of the reflector thickness is sharp, so it was not considered useful to provide more than 15 cm for the Al_2O_3 region.

The fraction of stainless steel in the blanket strongly influences the breeding, the effect being more pronounced in Lithium-7 (Fig. 8). In the FINTOR design care has been taken to maintain this fraction below 5%.

Similar parametric calculations indicate a decrease of 5.5% on the breeding ratio, passing from 0 to 10% of the void fraction in the blanket, and a 0.3% decrease for a 100°C variation of the blanket temperature. For design calculations this effect can therefore be ignored.

Magnet Shielding Analysis

A number of candidate materials have been investigated (Figs. 9 and 10), namely: iron, borated graphite, organic liquid ($C_{18}H_{14}$), magnesium oxide, alumina, water, boron carbide and zirconium hydride.

Fig. 9 gives the fast neutron flux (> 0.1 MeV) distribution in blanket and shielding calculated by SABINE. By assuming that fast flux attenuation is a figure of merit for radiation damage in the magnet, B_4C and ZrH seem to be the most favourable materials. As mentioned in the introduction, helium-cooled B_4C powder has been finally chosen for the FINTOR shielding design. Alternative solutions proposed in similar studies by others[11, 12] are compared with the neutron fast flux attenuation in Fig. 10. The FINTOR solution appears to be less favourable than those which include water, whereas it looks better than the one using lead, which has been optimized from the point of view of heat deposition in the magnet.

Fintor Results

In Fig. 11 are schematized the blanket and magnet shielding arrangements which have been taken for FINTOR Reference Design calculations. A thin lead layer has been put outside the B_4C-shielding to reduce the gamma-energy deposition in the magnet. An increase of the lead thickness will further reduce heat deposition without a noticeable decrease of fast neutron flux in the magnet (~~see Fig. 10~~); the optimization of this parameter has not yet been made. Tables II and III are summarizing the tritium breeding and nuclear heating results.

TABLE II : TRITIUM BREEDING

Lithium-6 (n, α) T	0.729
Lithium-7 (n, α, n') T	0.468
Lithium Breeding Ratio	1.197

TABLE III : NUCLEAR HEATING IN MeV PER 1 FUSION NEUTRON

Zone	Neutron Heating	Gamma Heating	Total Heating
First Wall	0.31	0.80	1.11
Blanket	11.73	2.09	13.82
Reflector	0.55	0.88	1.43
Shielding	0.93	0.37	1.30
Magnet	–	6×10^{-4}	6×10^{-4}
Total	13.520	4.14	17.66

The energy amplification is about 26%. As mentioned in ref. 11, this value could be reduced by using a more consistent set of neutron KERMA factors. However, it can be observed that a considerable amount (7.4%) of the nuclear heating is deposited in the magnet shielding; this is probably due to the exothermic (n, α) B^{10} reaction (Q= + 2.79 MeV). Fig. 12 shows total neutron and gamma heat deposits. The energy deposited in the magnet has a peak of 1.65×10^{-4} W/cm^3 which requires a cooling electric power of 3.75 MWe, i.e. about 5% of the reactor power. This value would be unacceptable in a commercial reactor from an economic point of view but it is feasible and can be reduced, as already mentioned, by increasing the lead thickness.

The response functions of interest for radiation damage evaluations are given in Figs. 13, 14, 15. In the first wall a maximum value of 7.2 d.p.a. is found after 3 years of equivalent life time (Fig. 13). For these conditions no problems either from swelling or from embrittlement are expected in ST 316 stainless steel. Fig. 14 shows atom displacements in the copper stabilizer. A maximum value of 4.2×10^{-5} d.p.a./year is found, which roughly corresponds, according to ref. 11, to a factor of two increase in the copper resistivity. From the magnet design point of view, this is the maximum acceptable value, so annealing of the copper during the life of the reactor should be envisaged.

It has to be noted that the high sensitivity of the copper resistivity to the atom displacements requires a very accurate evaluation of the d.p.a. in the magnet. Furthermore, this fact implies that, even in experimental reactors where the fast neutron fluence is low, a rather thick shield is needed.

Conclusions and Further Development

The results obtained so far permit one to draw some conclusions concerning the nuclear design of blanket and magnet shielding of an experimental, low power, TOKAMAK reactor; in particular:

- convenient breeding ratios can be obtained with 50 cm thick metallic lithium blankets, provided the fraction of structural material is kept low. However, the possibility of replacing liquid lithium by LiAl-alloys mixed with beryllium, should be investigated;
- various choices exist as far as reflector material and its position in the blanket are concerned;
- boron carbide with an external lead layer represents an attractive way to fulfill respectively the radiation damage and heat deposition requirements in the magnet, provided that a sufficient thickness is allowed all around the plasma.

Further developments will concern:

- a better description of the blanket configuration. In fact, as has already been pointed out, a D-shaped configuration, involving divertor slots, is proposed for the blanket in the FINTOR design. Monte Carlo or two-dimensional S-n calculations will be required;
- A more consistent use of nuclear data libraries completely based on the most recent ENDF/B files, together with a deeper analysis of copper displacement cross sections;
- the establishment of simplified and flexible design calculation methods, in particular the full adaptation of the SABINE-code to CTR conditions.

References

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2) Minimum Activity Blankets for Commercial and Experimental Power Reactors. J.R. Powell et al. Culham Workshop.

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5) Neutron Displacement Cross Sections for Stainless Steel and Tantalum based on a Lindhard Model. By D.G. Doran. *Nucl. Sci. and Eng.* 44, 130-144 (1972).

6) ARTUS X. Calcul des Quantités d'énergie cédées par le neutrons à un réseau atomique. By F. Gervaise. Rapport SERNA/S Nr. 13. CEA. Nov. 1971.

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8) SABINE. A One Dimensional Bulk Shielding Program by C. Ponti et al. EUR 3636 e.

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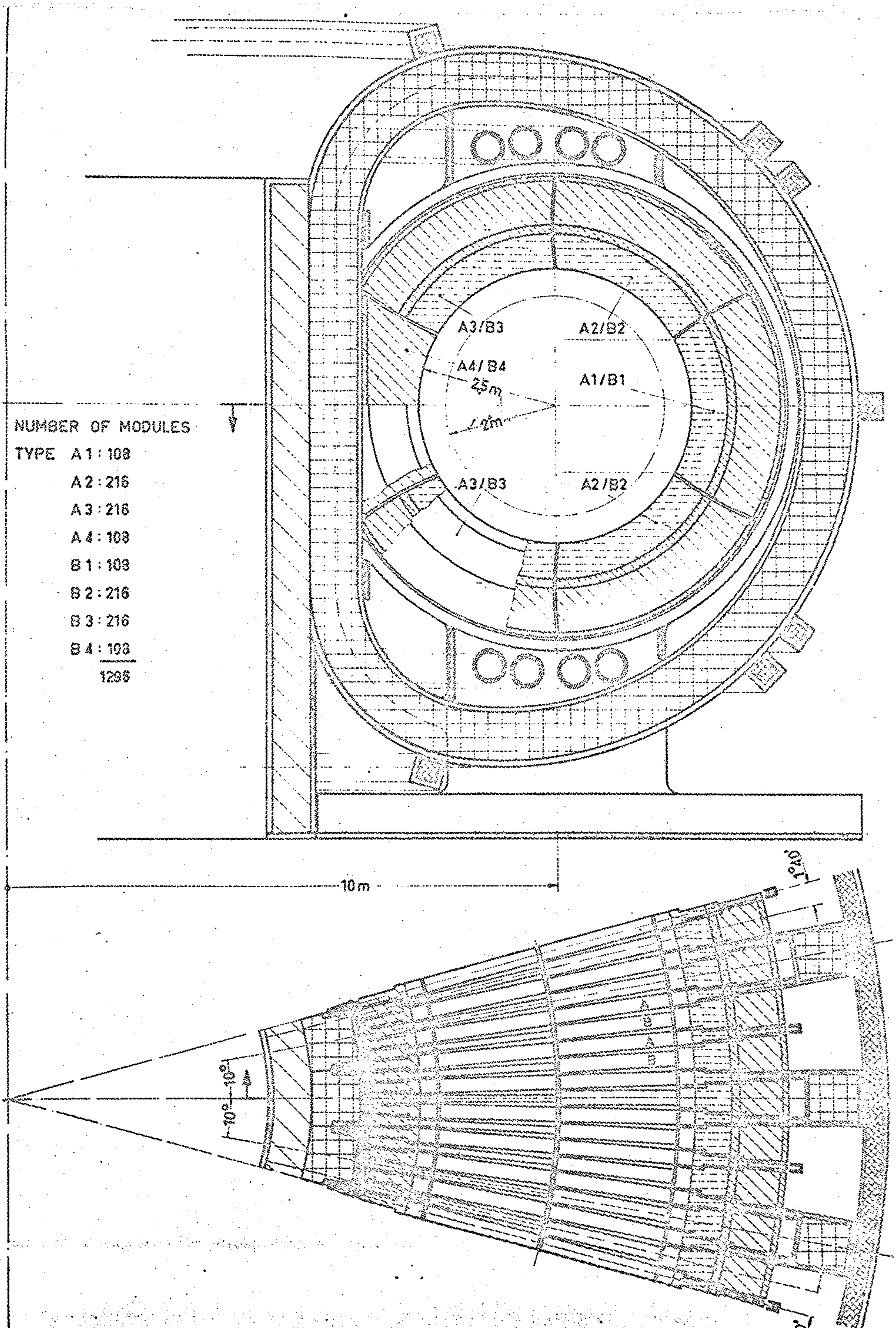
11) UWMAK–1. A Wisconsin Toroidal Fusion Reactor Design by the University of Wisconsin Fusion Feasibility Study Group. UW FDM–68. Nov. 20, 1973.

12) Magnet Shield Design for Fusion Reactors. J.T. Kriese, D. Steiner. ORNL–TM–4256. June 1973.



NUMBER OF MODULES

TYPE	
A 1	108
A 2	216
A 3	216
A 4	108
B 1	108
B 2	216
B 3	216
B 4	108
	1296

FIG. 1 SCHEMATIC VIEW OF 'FINTOR' WITHOUT DIVERTOR

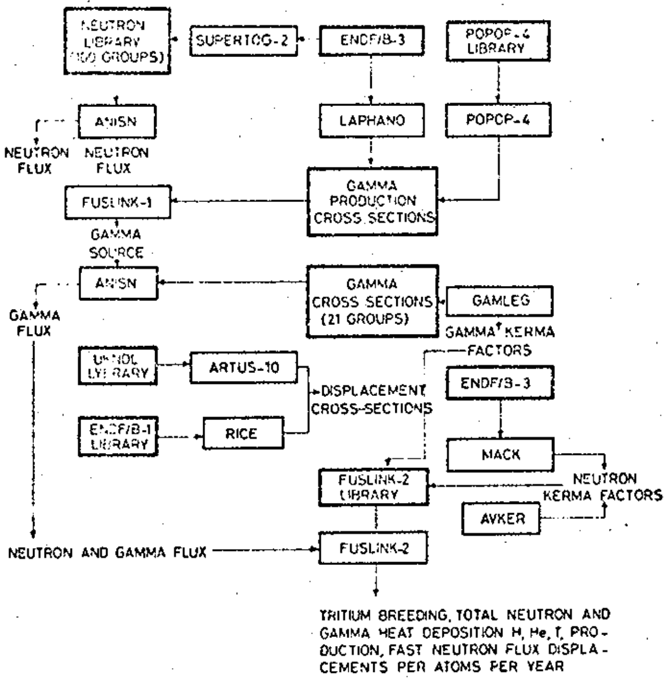


Fig. 2 - CALCULATION SCHEME

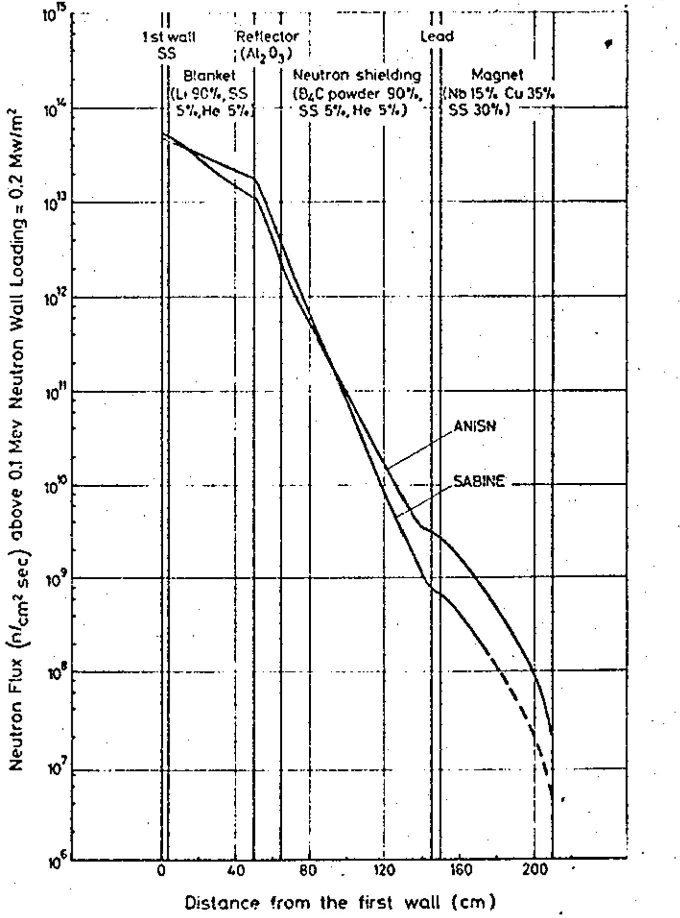


Fig. 3-FAST NEUTRON FLUX (> 0.1 MEV) COMPARISON ANISN SABINE

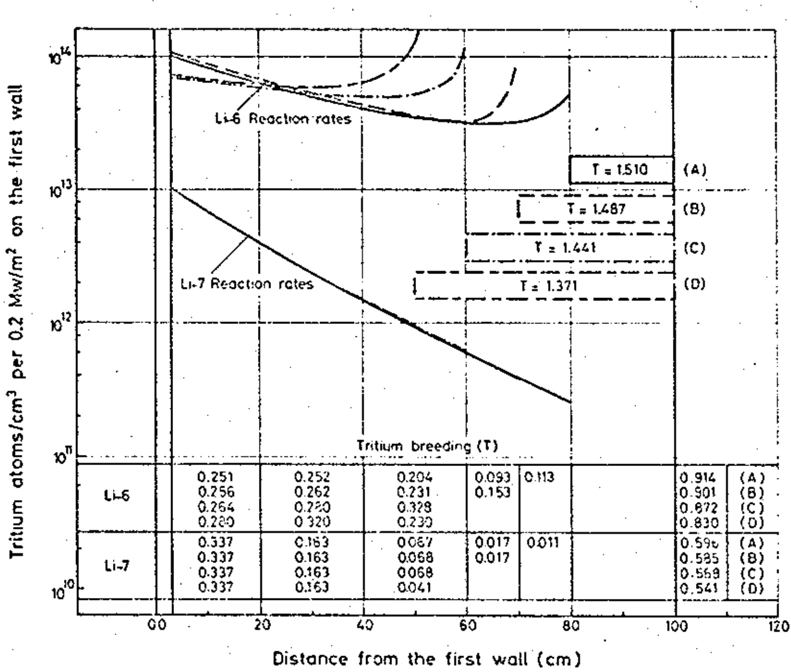


Fig. 4 - TRITIUM BREEDING AS A FUNCTION OF THE BLANKET THICKNESS

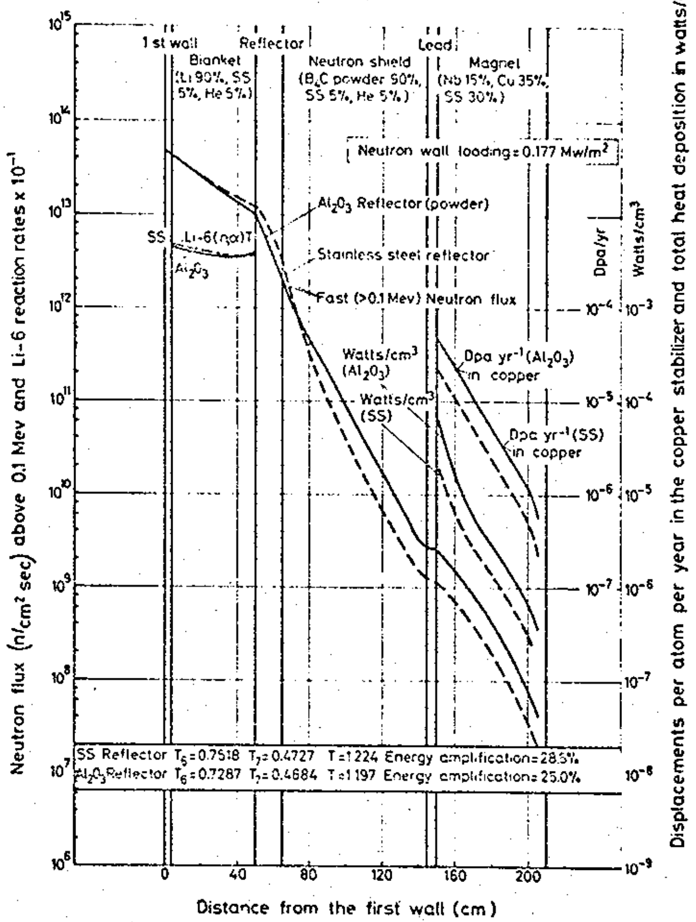


Fig.5 - COMPARISON BETWEEN Al₂O₃ AND STAINLESS STEEL REFLECTOR

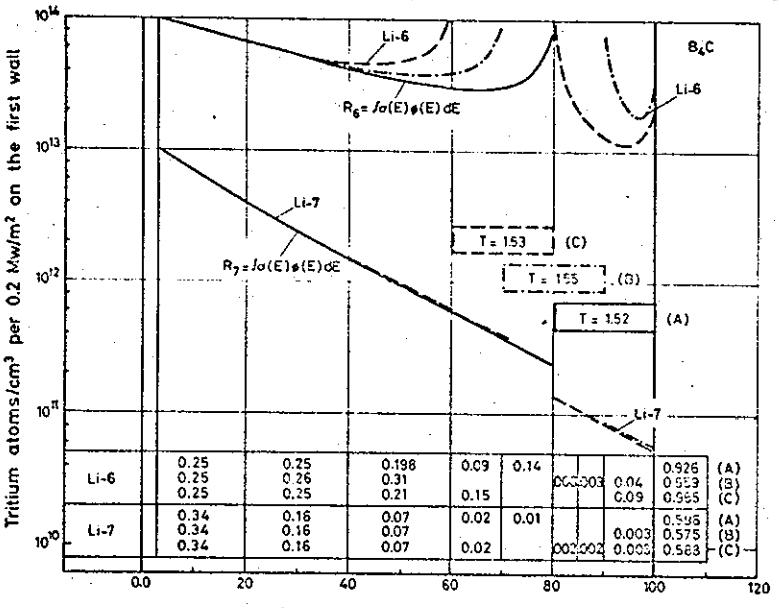


Fig.6-TRITIUM BREEDING AS A FUNCTION OF THE POSITION OF THE ALUMINIA REFLECTOR

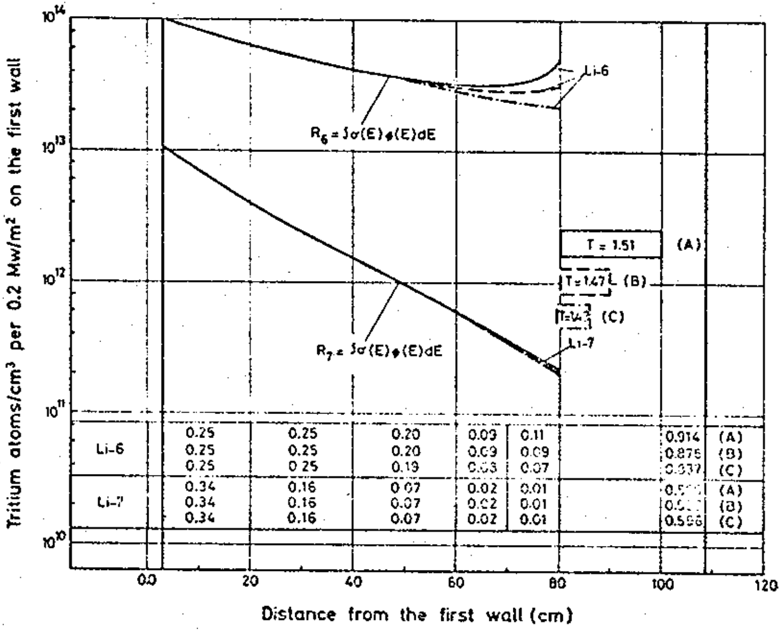


Fig. 7 -TRITIUM BREEDING AS A FUNCTION OF THE THICKNESS OF THE Al_2O_3 REFLECTOR

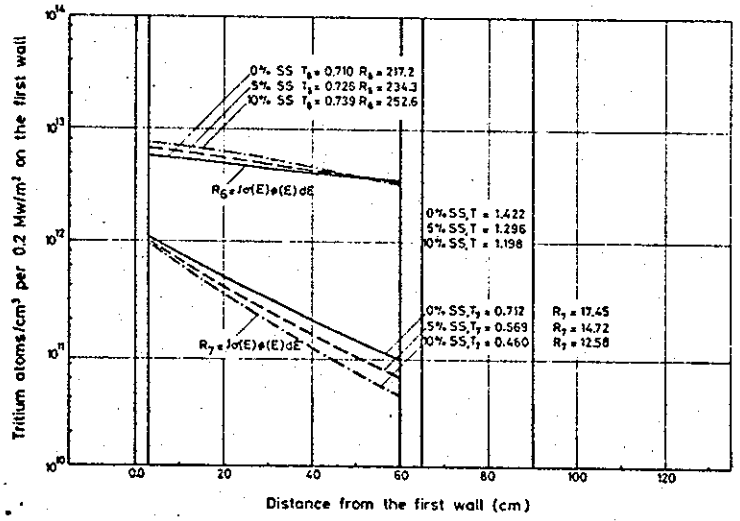


Fig. 8 - TRITIUM BREEDING AS A FUNCTION OF THE STAINLESS STEEL CONTENT IN THE LITHIUM BLANKET

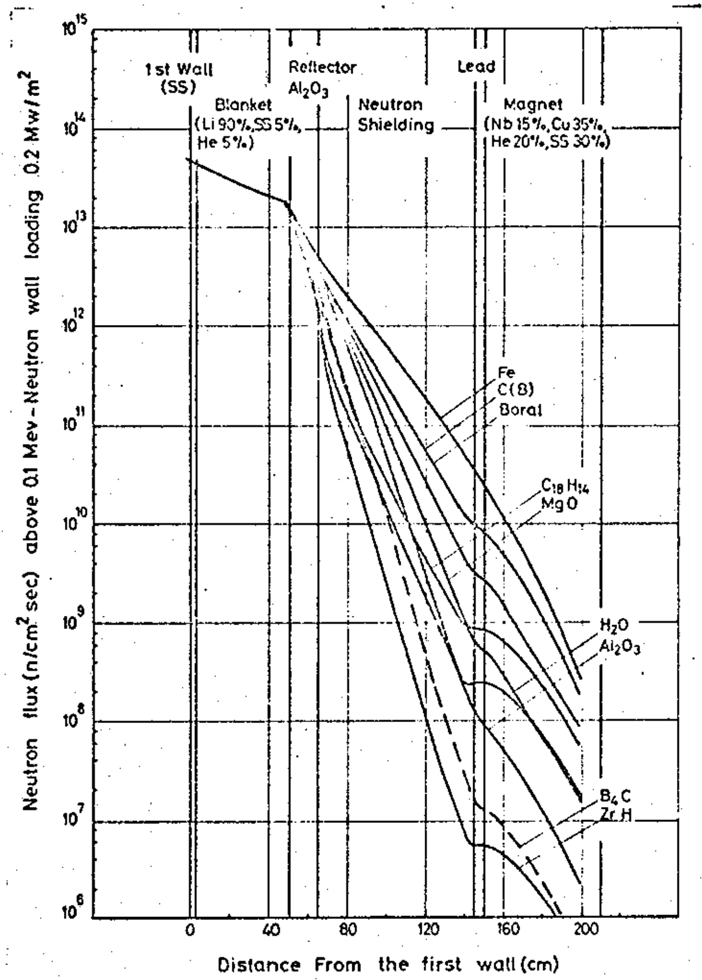


Fig.9-FAST NEUTRON FLUX (>0.1 Mev) FOR DIFFERENT NEUTRON SHIELDING MATERIALS

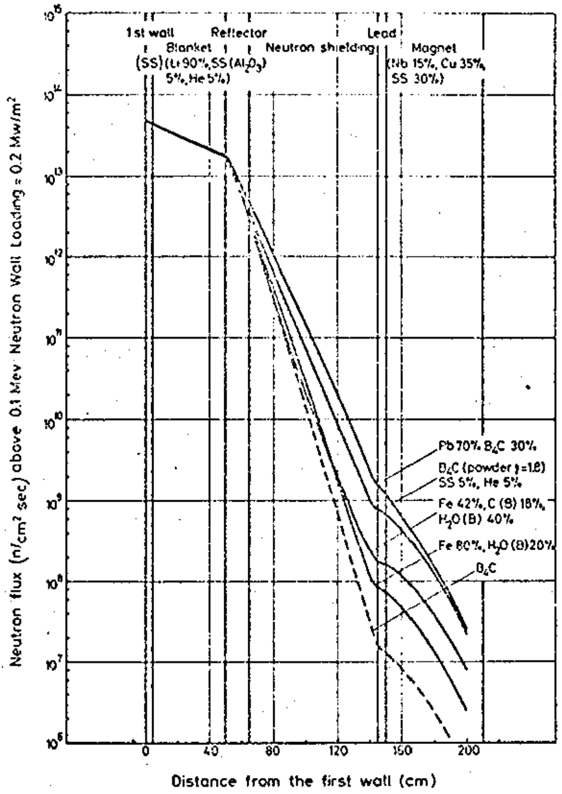


Fig.10-FAST NEUTRON FLUX (> 0.1 MEV) FOR DIFFERENT NEUTRON SHIELDING MATERIALS

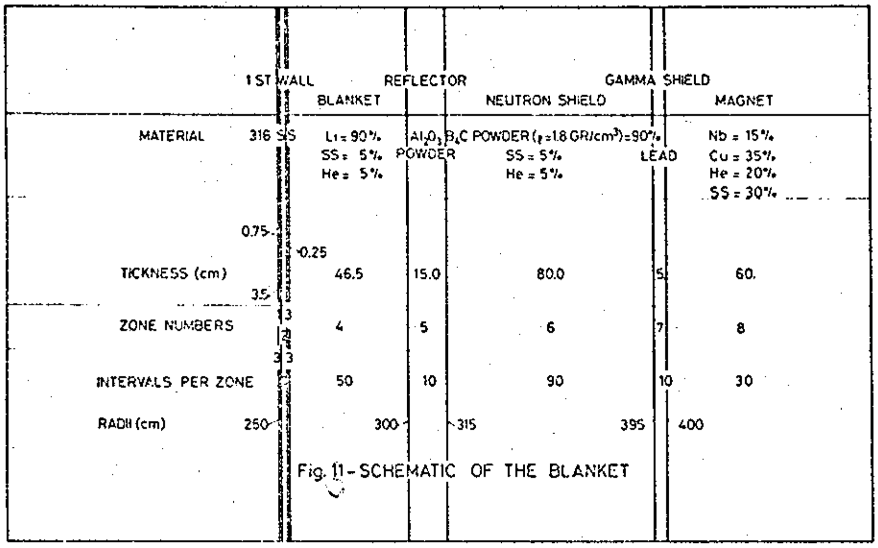


Fig. 11 - SCHEMATIC OF THE BLANKET

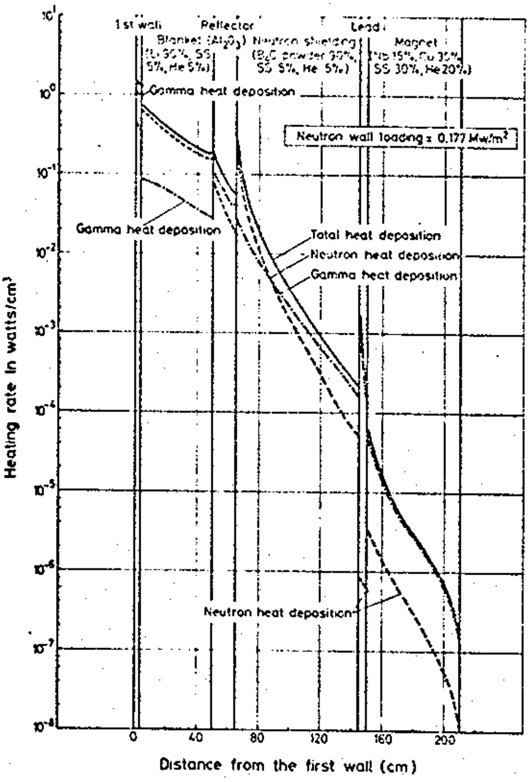


Fig. 12 - NEUTRON, GAMMA AND TOTAL HEATING IN THE BLANKET, SHIELDING AND MAGNET

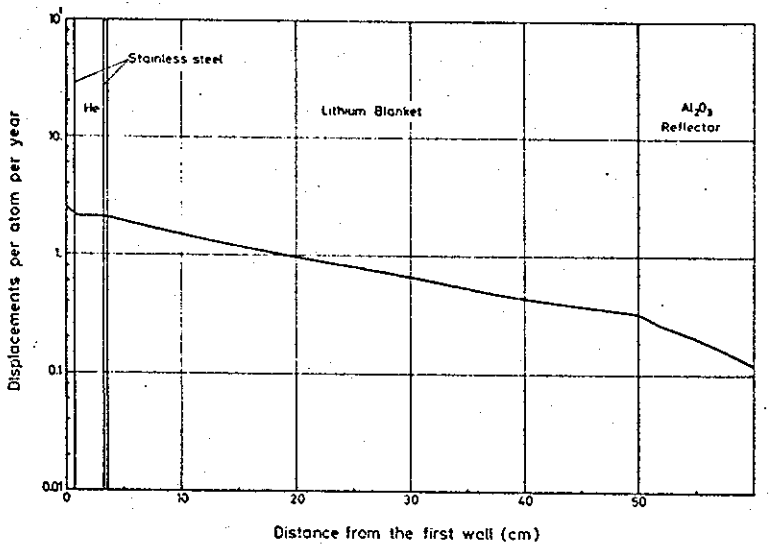


Fig.13 - CALCULATED DISPLACEMENT RATE FOR STAINLESS STEEL (NEUTRON WALL LOADING 0.177 Mw/m²

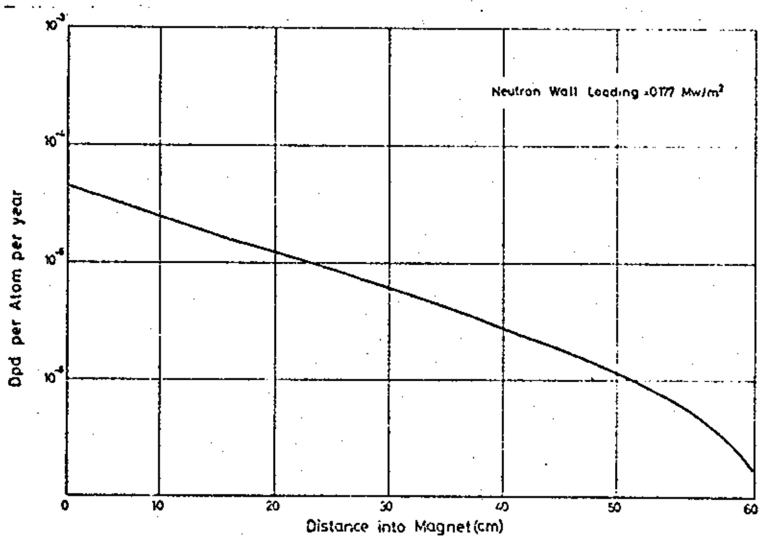


Fig.14-DISPLACEMENT PER ATOM PER YEAR IN THE COPPER STABILIZER

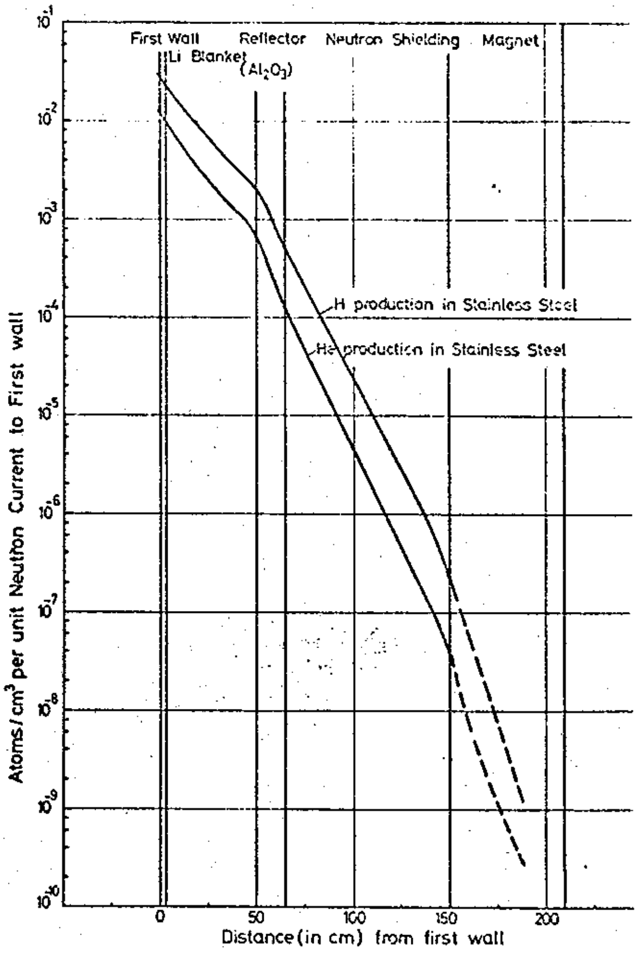


Fig.15-H, He, PRODUCTION IN BLANKET AND SHIELD