UCLA-ENG--8607

DE86 012879

UCLA-ENG-8607
PPG-933

Neutronic Optimization of a $LiAlO_2$ Solid Breeder Blanket

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February 1986

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This work was supported by the U.S. Department of Energy, Office of Fusion Energy, Grant #DE-[redacted]03-80ER52061, with UCLA.
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1. INTRODUCTION

The development of blanket concepts based upon individual pods or canisters has been pursued since the early days of conceptual fusion designs [1,2]. Several perceived advantages have been considered for these designs. Results from a recent Blanket Comparison and Selection Study (BCSS) [3] have indicated that a combination of helium as a coolant and ceramic Li-bearing solid breeder can satisfy necessary neutronic and thermomechanical performance criteria. In a version of a GA blanket design [4], the Li_2O ceramic breeder has been proposed in the form of clad plates. The helium coolant temperature increases from the front to the back side of the blanket. Even though the energy deposition profile in the solid breeder decreases rapidly with depth, helium flow can be controlled to achieve thermal homogeneity within the blanket. However, several critical areas remain unaddressed in the previous studies:

(1) The plate cladding structure, with its thin dimensions, may not be tolerant to thermal and irradiation inelastic strains resulting in reduced lifetimes.

(2) "Box-type" structures are generally prone to stress concentrations at corners, leading to early failure.

(3) Even though Li_2O has a higher lithium atom density as compared to $LiAlO_2$, its swelling rate is about an order of magnitude larger.

(4) With a greatly reduced structure-to-breeder ratio, the breeding ratio was found to be marginal [4].

Recent efforts undertaken at CEA Saclay (France) have concentrated on Clad breeder elements. This has been dictated by the desirability of maintaining a reasonable breeder geometrical integrity and a well controlled working temperature. Their studies of Canister blankets with Beryllium multiplier and solid breeder indicated that better tritium breeding is achieved when Beryllium is mixed with the solid breeder [5]. However, the chemical compatibility problems, which are crucial for the viability of such a design, remain to be resolved.

In the present study, we adopt the following:

(1) A pressurized lobular configuration is used rather than the "canister" configuration. This allows for flat side plates as shown in figure (1). The configuration is well suited for tight-fitting locations, such as the in-board blanket. A high volume fraction can thus be achieved.

(2) Pressurized helium flows in the radial direction, achieving thermal homogeneity, as described in reference (6).

(3) The use of Beryllium in the front zone of the blanket is consistent with its thermo-physical properties, i.e. high conductivity. The lower conductivity breeder material is used deep inside the blanket, where the nuclear heat generation is lower.

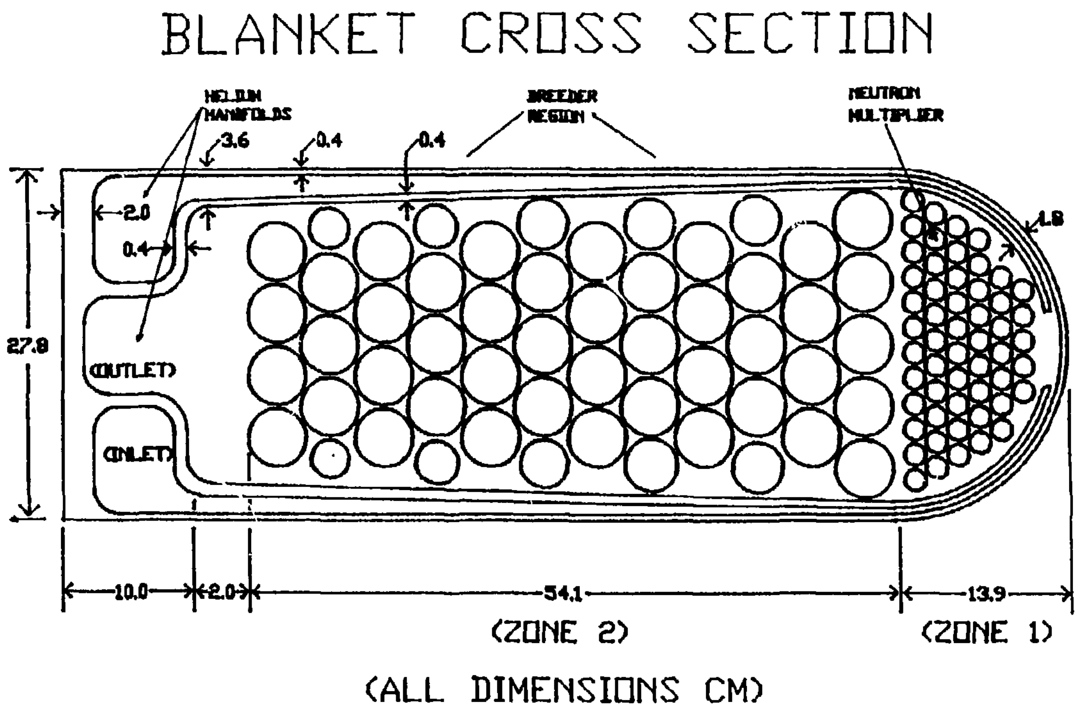

Fig. 1. Preliminary pressurized lobular blanket configuration

(4) Beryllium and Solid breeder pins are arranged such that helium cross-flow conditions are achieved. The small size of pins ensures minimum temperature assymmetries. This is shown to result in minimal bowing and deflections within the blanket [7].

(5) The side plates are tapered from the back to the front in order to accommodate the high internal pressure in a stand-alone hypothetical accident scenario (see reference [8]).

We present here the results of neutronic optimization calculations of a solid breeder blanket. The objective is to optimize the spatial material allocations in the blanket, consistent with a number of engineering constraints. The materials used in the present analysis are given below:

1. Structural Material: low activation material 9-C. This material, which is structurally equivalent to HT-9, has the following composition [9]:

Cr = 11.81%	C = 0.097%	V = 0.28%
W = 0.89%	Mn = 6.47%	Si = 0.11%
N = 0.003%	P < 0.005%	S < 0.005%
Fe = remainder		

   The 9-C structural material is used for the lobe shell, and also for caldding the solid breeder material.

2. Solid Breeder Material: $LiAlO_2$, with variable enrichment of Li, to be determined by the optimization study.

3. Neutron Multiplier: Beryllium

4. Coolant: Helium

5. Shield-1: Water and Ferritic Steel Fe-1422

6. Shield-2: Water, Ferritic Steel Fe-1422 and B_4C

2. SYSTEM CONFIGURATION AND MODELING

One dimensional neutron transport calculations have been typically based on a cylindrical approximation to the toroidal geometry. The ANISN neutron transport code [10], which utilizes the S-N method of solution to the transport problem, provides for approximate 1-D solutions for infinite slabs or cylinders. The actual blanket configuration, shown in figure (1), reveals more geometric details than can actually be modelled in 1-D calculations. For example, the solid breeder pins are also cladd with the 9-C ferritic material. A cross-section of a typical pin is shown in figure (2). To determine such details, many engineering factors have to be considered The final blanket configuration must result in acceptable neutronic and thermostructural performance. A basic consideration, however, is how to maximize the tritium breeding ratio by an appropriate spatial distribution of materials, without exceeding or conflicting with other engineering constraints.

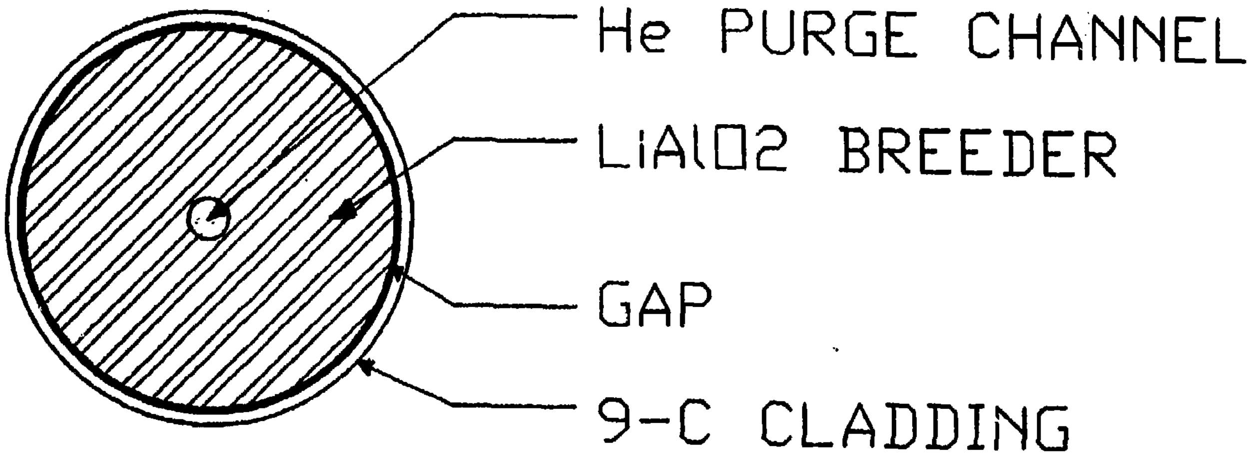

Fig. 2. Cross section of a typical breeder pin

A one dimensional model is obtained by treating the various zones as shells of infinite height. Materials are smeared within each zone according to their densities. This is consistent with a flat flux approximation, where self-shielding effects are neglected.

The neutron transport and gamma production cross-sections were obtained as a coupled set of 46 neutron groups and 21 gamma groups produced by the AMPX modular code system [11] from the nuclear data in ENDF/B-IV. The cross sections were weighted with a 1/E spectrum for $E_n > 0.345$ eV, and with a Maxwellian distribution for $E_n \leq 0.345$ eV. The gamma interaction cross-sections were uniformly weighted. Due to cost limits, the 4 neutron group cross-sections were collapsed to a smaller number of groups.

In the next section, we describe the methodology adopted in this optimization study, outlining the basic principles of the technique. This is followed by the results of optimization studies, starting with modeling an idealistic system without any constraints. This is then followed by modeling a system that is more realistic with appropriate engineering constraints.

3. THE SWAN OPTIMIZATION CODE

A. General

SWAN [12] is a code developed for the analysis and optimization of the nucleonic characteristics of CTR blankets. Any nuclear system, that is described by the inhomogeneous linear transport equation, can be analyzed by the SWAN code.

SWAN is composed of two modules, SWIF and ANISN. ANISN [10] is a one dimensional discrete ordinates transport code. SWIF is a code developed for perturbation calculations and optimization studies.

The type of optimization problems that can be handled by SWAN can be characterized as follows:

Given the external source distribution

$$S(x) = S(\underline{r},E,\underline{\Omega}) \qquad (1)$$

and the atomic density distribution $N_i(\underline{r})$ for all I materials the density of which are variables of the optimization, find the material density distribution that will extremize the functional $F_e(N_1,N_2, \ldots, N_I)$ subject to:

1. the constraints imposed by the density limits

$$N_i^{min}(\underline{r}) \leq N_i(\underline{r}) \leq N_i^{max}(\underline{r}) \leq N_i^{o}(\underline{r}) \qquad (2)$$

where these densities are assumed constant in each zone.

2. the constraint on the total volume fraction available:

$$\sum_{i=1}^{I} \frac{N_i(\underline{r})}{N_i^o(\underline{r})} = \text{const} \leq 1.0 \qquad (3)$$

and

3. preservation of the value of the constraints [denoted by the functional $F_c(N_1,N_2,\ldots,N_I)$] imposed on the problem.

The characteristic to be extremized (F_c) can be of either one of two general categoris: a weight-type characteristic or a nucleonic characteristic. A weight-type characteristic is any characteristic expressible in the form

$$F_w(N_1,N_2,\ldots,N_I) = \sum_{i=1}^{I} \int d\underline{r} C_{w,i}(\underline{r}) \; N_i(\underline{r}) \quad , \qquad (4)$$

to be referred to as a weight functional. A nucleonic characteristic is one expressible in the form of a bilinear functional:

$$F_b(N_1,N_2,\ldots,N_I) = \int d\underline{r} \left[<\phi, S_b^+> + <\phi_b^+, S> - <\phi_b^+, H\phi> \right] \qquad (5)$$

(using the notation <,> for $\iint dE \, d\underline{\Omega}$) where the flux $\phi = \phi(\underline{r},E,\underline{\Omega})$ and the adjoint $\phi_b^+ = \phi_b^+(\underline{r},E,\underline{\Omega})$ are the solution of, respectively, the linear Boltzmann equation and its adjoint

$$H\phi(\underline{r},E,\underline{\Omega}) = S(\underline{r},E,\underline{\Omega}) \text{ and } H^+\phi_b^+(\underline{r},E,\underline{\Omega}) = S_b^+(\underline{r},E,\underline{\Omega}). \qquad (6)$$

Consequently the functional has the value

$$F_b = \int d\underline{r} <\phi, S_b^+> = \int d\underline{r} <\phi_b^+, S> \quad . \qquad (7)$$

The adjoint source term, S_b^+, is to be selected so as to give the adjoint function, and consequently the functional, an adequate physical meaning.

An Effectiveness Function may be defined for a bilinear function as:

$$e_{b,i}(\underline{r}) = <\phi, \frac{\delta S_b^+}{\delta N_i}> - <\phi_b^+, \frac{\delta H}{\delta N_i} \phi> \qquad (8)$$

and for a weight function as:

$$e_{w,i}(\underline{r}) = C_{w,i}(\underline{r}) \quad . \qquad (9)$$

The Substitution Effectiveness Function (SEF) is defined as follows:

$$Q_i(\underline{r}) = e_i(\underline{r}) - e_I(\underline{r}) \frac{N_I^o(\underline{r})}{N_i^o(\underline{r})} \qquad (10)$$

so that

$$\delta F = \sum_{i=1}^{I} \int d\underline{r}\ e_i(\underline{r})\ \delta N_i = \sum_{i=1}^{I-1} \int d\underline{r}\ Q_i(\underline{r})\ \delta N_i \quad . \tag{11}$$

Equations (10) and (11) hold for both bilinear and weight functionals.

The material densities for the n^{th} iteration are obtained from the parameters in the (n-1) iteration from the relation:

$$n_i^n(\underline{r}) = N_i^{n-1}(\underline{r}) + A_i^n\ Q_{e,i}^{n-1}(\underline{r}) + B_i^n\ Q_{c,i}^{n-1}(\underline{r}) \quad . \tag{12}$$

The procedure for calculating the A_i^n and B^n are described in detail in Reference [12]

4. OPTIMIZATION RESULTS

4.1 Preliminary Calculations

Preliminary calculations were first performed with slab geometry, as shown in figure (3). In this case, we assumed 60% enrichment of the Li^6 in $LiAlO_2$ at 95% of the theoretical density. Volume fractions were taken as 0.4 for $LiAlO_2$, 0.18 for 9-C and 0.48 for He. The first shield was assumed to be 95% Fe-1422 and 5% H_2O, while the second shield was assumed to contain the following materials:

Fe-1422 = 4.5%, H_2O = 5%, B_4C = 50%.

Reference calculations were performed using the S16-P3 approximation.

Several runs were performed with plane and cylindrical geometry options in the code. It was found that the results for both the TBR and various reaction rates are within 0.01%. Due to the cost of optimization procedures, a smaller set of groups was used. A small library of 9 neutron groups and 3 gamma groups was created by collapsing the basic library of 46 n groups and 21 γ groups, using flux weighting from preliminary blanket calculations. It was found that the TBR is within 10%. Moreover, the relative spatial allocation of materials was relatively unaffected.

4.2 Stage-I Optimization: Maximum TBR

The objective of this phase of the study is to allocate various materials in fixed geometry, in order to obtain the maximum tritium breeding ratio. The choice of materials is limited to $LiAlO_2$, 9-C, He and Be within the blanket. The composition of the two shields was not changed, however. The arrangement of materials is shown in Fig. (4).

Starting with a uniform distribution of materials in the blanket, the SWAN code calculates the effectiveness of each material at each spatial point. New material distributions are then calculated using the method of steepest descent. No constraint is imposed on the distributions except that the volume fractions must be positive and must add up to unity. The new material distributions are then used to calculate a new value of tritium

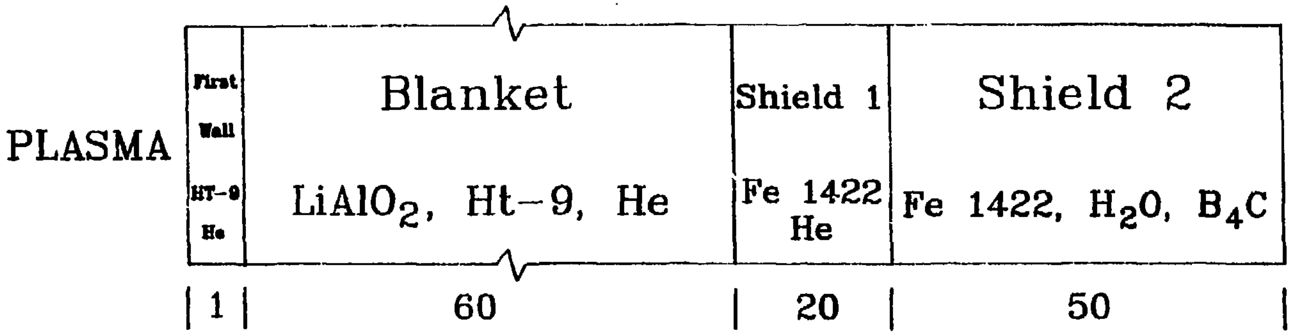

Fig. 3. One-dimensional blanket model for preliminary calculations

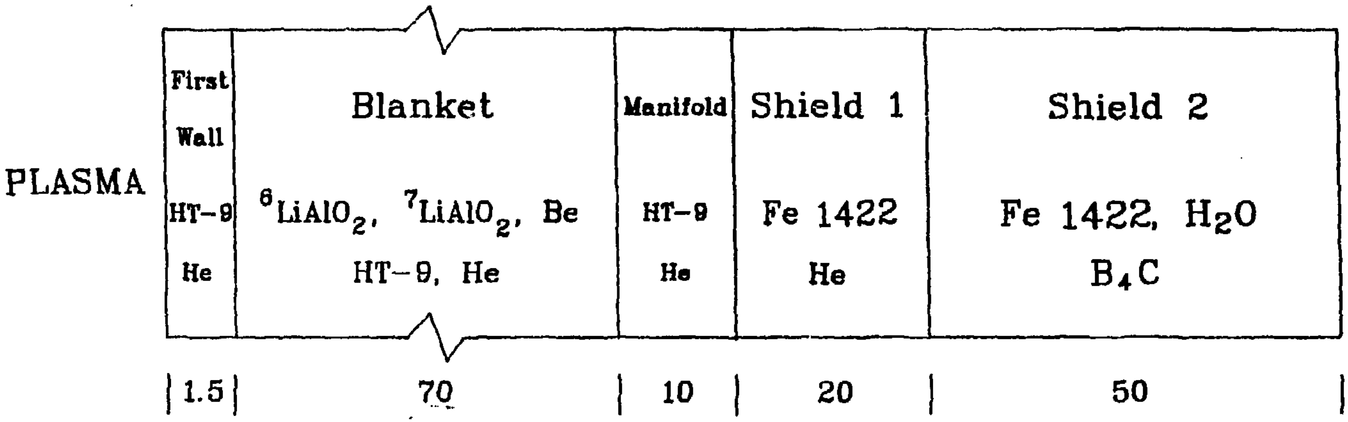

Fig. 4. One-dimensional blanket model for maximum TBR

breeding ratio, and generate new effectiveness functions. This iterative scheme is repeated until convergence is achieved.

First, it was found that the TBR was increased from 0.94 to 1.54, which is an increase of 64% due to the optimized allocations. Figure (5) indicates that the TBR is very sensitive to the location of Be. The material distributions, as shown in Figure (5), display similar trends as the effectiveness functions. Be is shown to occupy the front zone of the blanket, with little contributions from other materials. Except for the front section, where Be is dominant, all other materials maintain a uniform distribution with the following ratios: (a) Breeder to structure ratio $\simeq 4$, (b) Li^6 enrichment of 60%. Nuclear heating, like most other reaction rates, was found to be high in the front, and decreases rapidly with depth. In $LiAlO_2$, the heating rate was found to be high (126 w/cm^3).

A comparative study was then performed for a water-cooled system. All other materials, as well as the geometry, were adopted from the previous helium case. Figures (6) and (7) show the TBR effectiveness, and material allocations, respectively. The effectiveness is shown for the first iteration, while the allocations are given after convergence was achieved (21^{st} iteration). It can be observed that the material distribution is similar to the helium-cooled blanket case. However, the Be volume fraction reached a maximum of 1.0, while it was only 0.78 for the Helium case. Most of the other features of the results are very similar to the helium case. After 21 iterations, the TBR was found to increase from 0.94 to 1.7, an increase of 79%. The heating of $LiAlO_2$ was found to be exceptionally high in the first zone (870 w/cm^3), dropping to 180 w/cm^3 in the second interval.

4.3 Stage-II: Two Zone Constrained Blanket

Even though very high TBR values were obtained in the previous case (1.54 for He and 1.7 for H_2O), the spatial allocation of materials was not subject to geometrical or engineering constraints. We then attempted to impose realistic constraints on the optimization. A two zone blanket was first subjected to analysis and optimization, where behind the first wall a Beryllium zone was introduced. This zone was composed of 6 rows of 1.71 cm unclad Be rods (0.3 cm rod spacing). The composition of this zone was not changed during subsequent optimization. The back zone of the blanket was a mixture of $LiAlO_2$, 9-C, and Helium, with a fixed Li^6 enrichment of 60%. The structure to breeder ratio was varied, and its effect on the TBR was studied. With a collapsed library of 12 neutron groups, the TBR was found to be reasonably insensitive to this ratio, as shown in Figure (8). Calculations with the full 46 group library resulted in a TBR of 1.42, which is 8% lower than the 12 group results due to spectral differences.

The nuclear heating was found to be high in the front of the second zone, with a cell average value of 65 w/cm^3. However, averaging over separate materials showed even higher values reaching 98.8 w/cm^3 in $LiAlO_2$ alone. These high heating rates have led to temperatures well above the 1100°C limit set for $LiAlO_2$. This, in turn, led to the next and final optimization step; and that is to include temperature limits as well as density constraints.

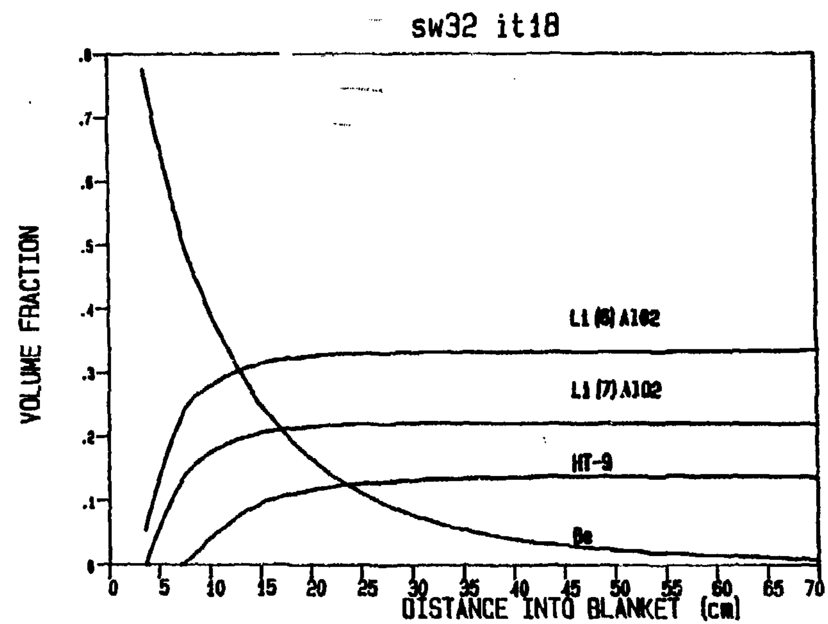

Fig. 5. Results of optimum material distributions for maximum TBR for He-cooled design

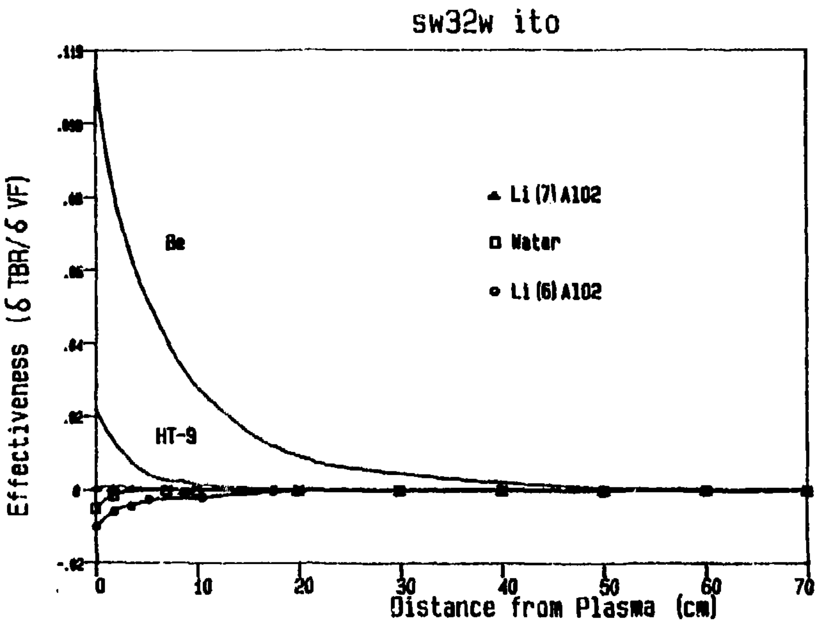

Fig. 6. Dependence of effectiveness on distance for water-cooled design

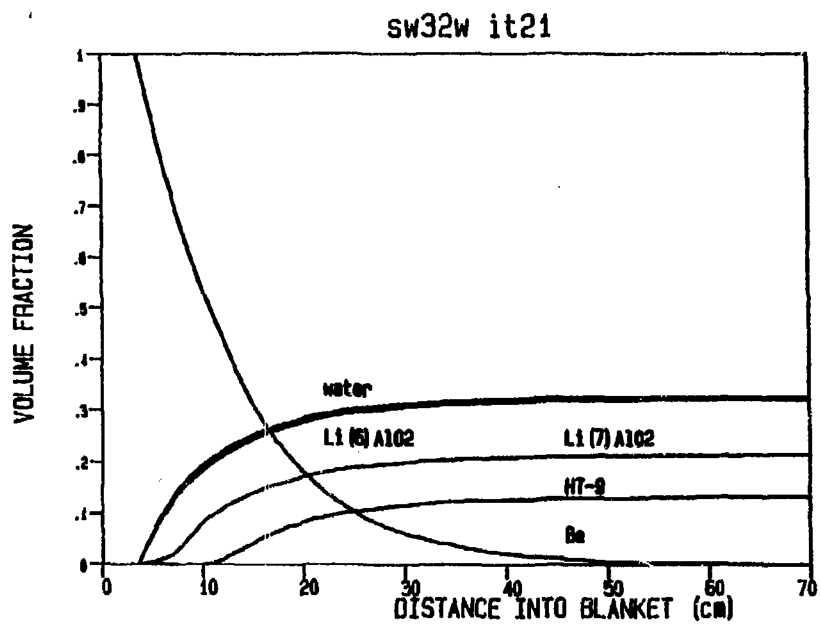

Fig. 7. Optimum material distributions for water-cooled design

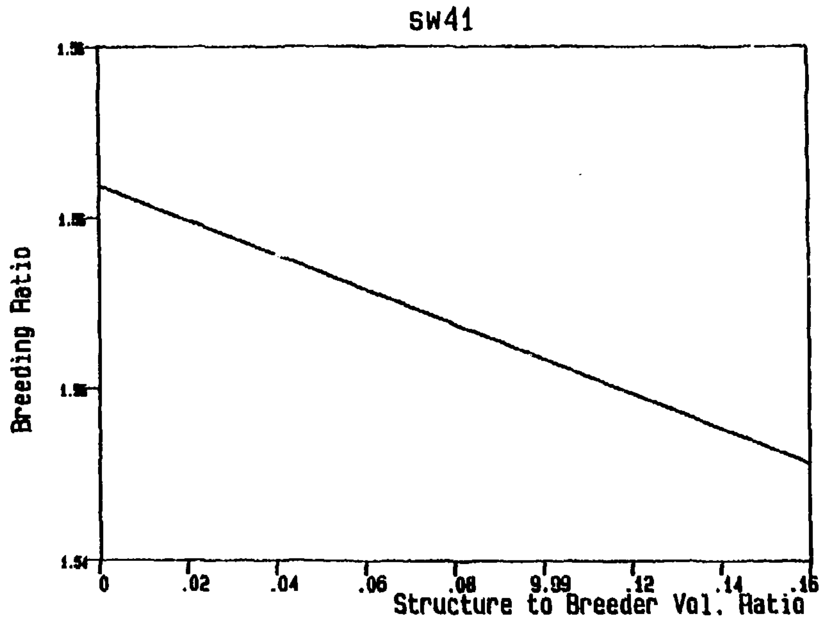

Fig. 8. Dependence of breeding ratio on the structure to breeder volume fraction.

4.4 Stage-III Optimization: The Reference Design

First, nuclear heating and rod temperature limits were considered as drivers for the blanket arrangements. Based upon the results of previous stages, a three zone arrangement was set

Zone 1: 6 rows of bare Be rods (1.7 cm od)
Zone 2: 5 rows of $LiAlO_2$/9-C rods (1.22 cm od)
Zone 3: 5 rows of LiAlO2/9-C rods (3.22 cm od)

The cladding thickness was chosen as 0.1 cm, with a nominal gap thickness of 0.01 cm. The distance between adjacent pins was taken as 0.3 cm. Thermal and mechanical considerations for these choices can be reviewed in references [6] and [7]. The final blanket configuration is shown in Figure (9).

Throughout all previous optimization studies, the breeder density was taken as 95%. However, modeling of tritium transport indicated that such a high density may result in unacceptable tritium inventory [13]. Consequently, the density of $LiAlO_2$ was reduced to 85% of the theoretical density. Furthermore, the total blanket depth was reduced to about 0.5 m, including helium gas manifolds, so that modules can fit either in the out-board or in in-board sections of the Tokamak.

It was found that with all of the previous constraints, and the reasonably short blanket modules, the tritium breeding ration is 1.175.

For a neutron wall loading of 5 MW/m^2, the integrated values of nuclear heating were as follows:

First wall	0.87 MW/m^2
Beryllium	1.86 MW/m^2
$LiAlO_2$	3.14 MW/m^2
9-C cladding	0.54 MW/m^2
Manifold section	0.19 MW/m^2
Shield-1	0.78 MW/m^2
Shield-2	0.12 MW/m^2
Total	7.49 MW/m^2

As can be seen, the heat generated in the first wall, Be, $LiAlO_2$/9-C and manifold section is 6.59 MW/m^2, yielding an energy multiplication factor of 1.32. The actual volumetric heat distribution in the blanket is shown in Figure (10). It is to be noted that the maximum values for $LiAlO_2$ heat generation rate is 179 w/cm^3, while that for Be is 35 w/cm^3. However, when averaging is performed over rod dimensions, the following maximum values are obtained:

Beryllium	35 w/cm^3
$LiAlO_2$ (1.22 cm od)	127 w/cm^3
$LiAlO_2$ (3.22 cm od)	21 w/cm^3

These values were found to result in acceptable temperature distributions within the breeder and Be rods [6].

The distribution of tritium production in the $LiAlO_2$ throughout the

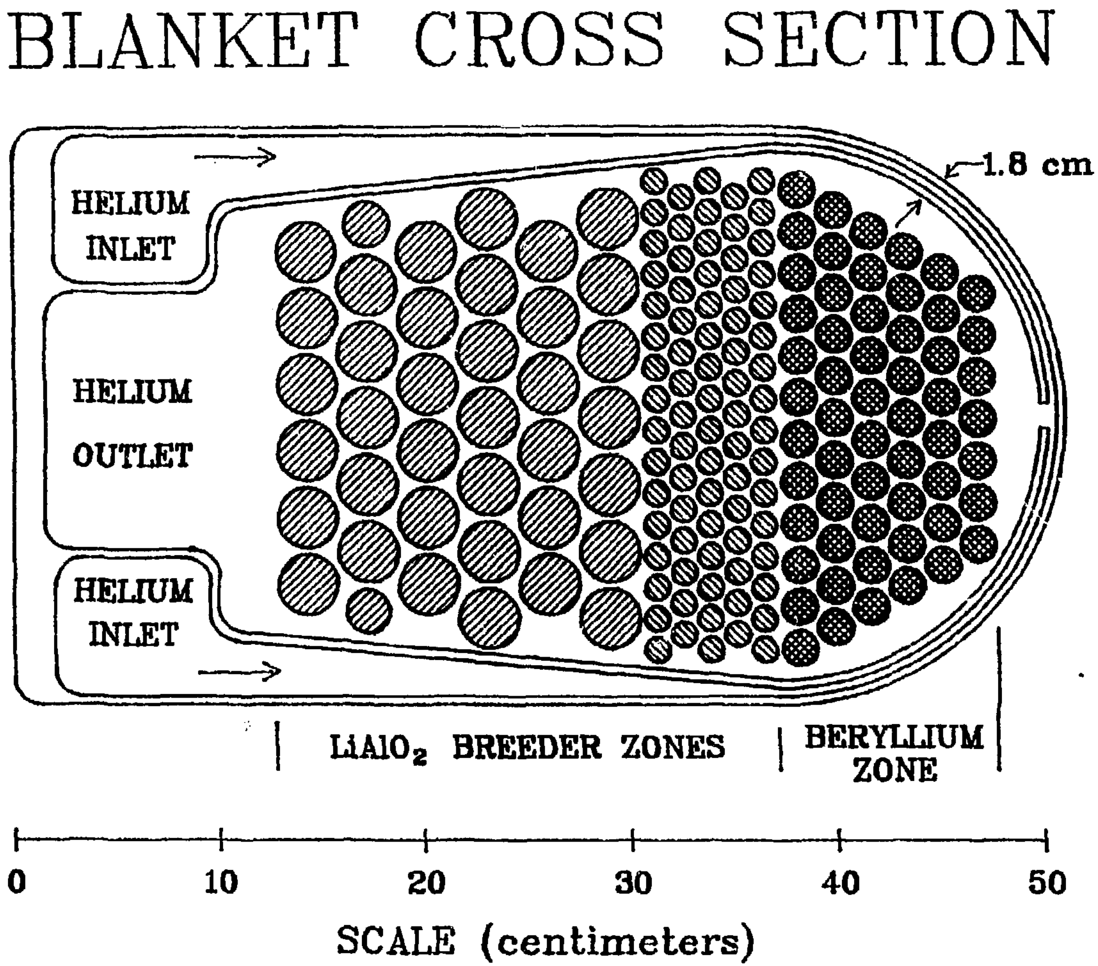

Fig. 9. Final configuration of optimized reference blanket

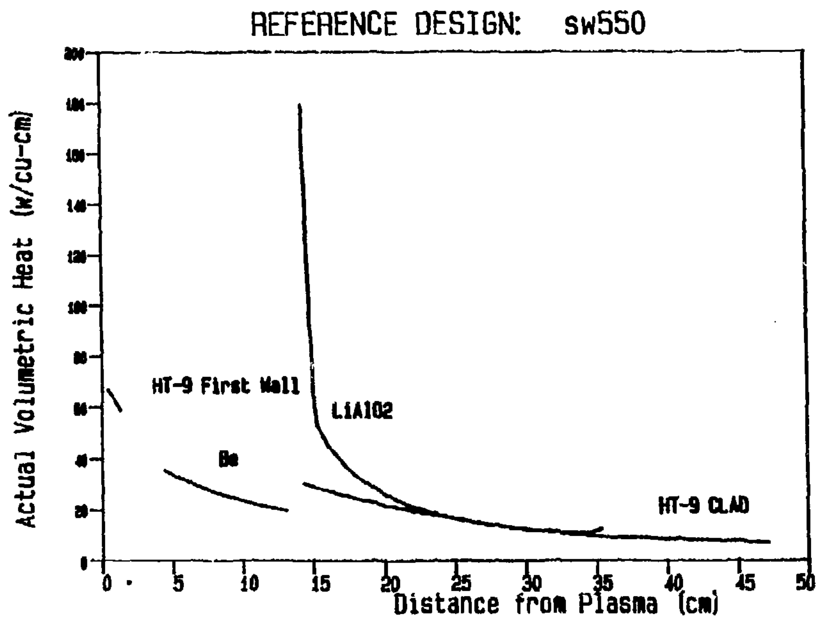

Fig. 10. Distribution of volumetric heat generation rate within the reference blanket

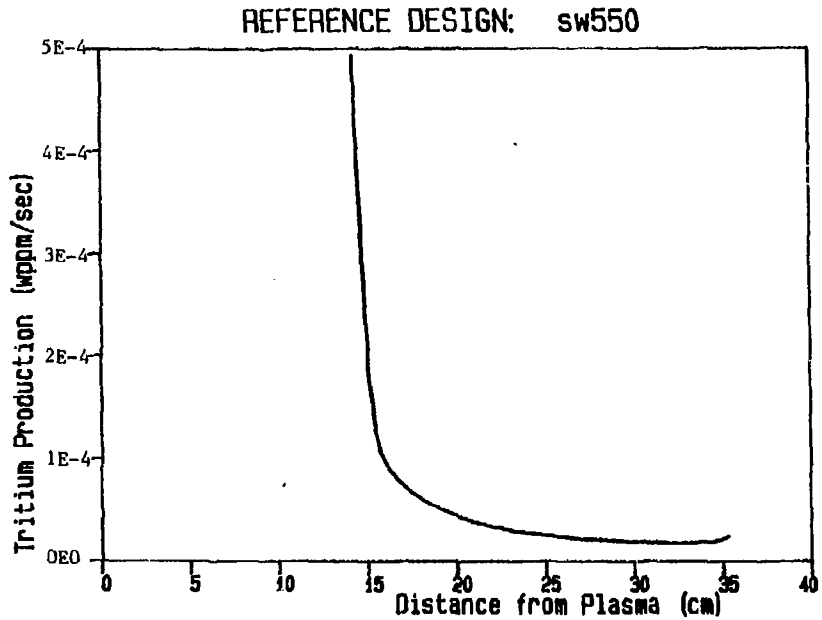

Fig. 11. Distribution of tritium production within the reference blanket

blanket is shown in Figure (11). It was found that the average tritium production rate is 4.31×10^{-5} wppm/s (1359 wppm/yr). However, 76% is produced in the first breeder zone and only 24% is produced in the back breeder zone. Nearly 99% of produced tritium was in Li^6 reactions. Tritium was also found to be produced in Beryllium, with an average rate of 122.8 appm/y, and a maximum value of 204.5 appm/y.

Helium was found to be produced in all blanket materials, as shown in Figure (12). However, the production rate of helium in $LiAlO_2$ was found to be exceptionally high, reaching a maximum of 85,000 appm/y. Displacement damage rates in the first wall, $LiAlO_2$ breeder and the 9-C cladding, are shown in Figure (13). The maximum first wall value is 64 dpa/y.

5. CONCLUSIONS AND RECOMMENDATIONS

The present calculations have shown the viability of a pin-type solid breeder blanket for tritium breeding. Engineering materials and geometrical constraints were all considered self-consistently, without a sacrifice of tritium self-sufficiency. Moreover, the final reference blanket is quite thin, with a total depth of 46.3 cm including the manifolds and first wall zones. The amount of breeder material is small, since the solid breeder zone is only 21.8 cm. The reference blanket can therefore be used in in-board or out-board configurations. It was also found that all reaction rates display a steep dependence on distance into the blanket. Thus, across the breeder zone only, the tritium production rate decreases by a factor of 25, the heating rate by a factor of 16 and helium production by a factor of 26. As in many other blanket designs, this behavior is a result of the nature of the fusion reaction, producing neutrons that impinge on one side of the blanket. The volume utilization of the blanket is less than optimal, and consequently, the design can be further improved.

ACKNOWLEDGMENT

This work was supported by the U.S. Department of Energy, Office of Fusion Energy, Grant #DE-FG03-80ER52061 with UCLA.

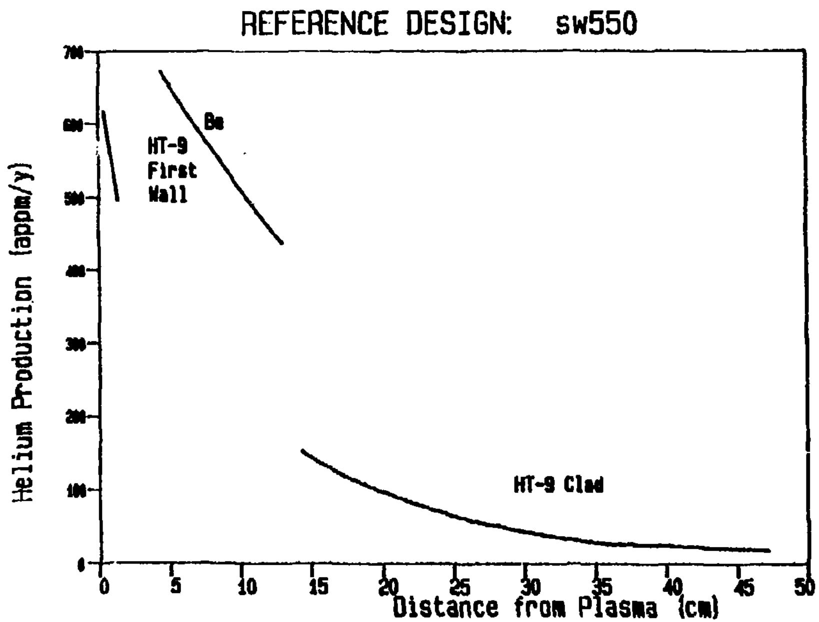

Fig. 12. Distribution of helium production within the reference blanket

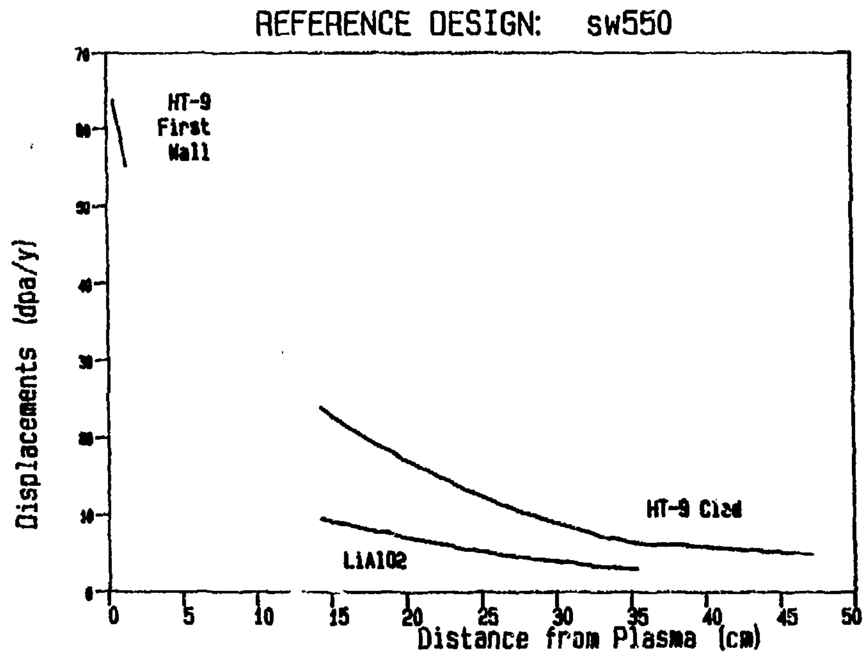

Fig. 13. Distribution of displacement damage rate within the reference blanket

References

(1) B. Badger et al., "UMWAK-II, A Conceptual Tokamak Reactor Design," University of Wisconsin Report, UWFDM-112, 1975.

(2) J.T.D. Mitchell, Culham Laboratory, Report CLM R126, 1974.

(3) D.L. Smith, G.D. Morgan et al., "Blanket Comparison and Selection Study, Final Report," Argonne National Laboratory Report ANL/FPP-84-1, Sept. 1984.

(4) C. Wong et al., "Helium-Cooled Blanket Design," Fusion Technology, 8, No. 1, p. 114-132, 1985.

(5) F. Carre, G. Chevereau, F. Gervaise and E. Proust, "Status of Fusion Blanket Evaluation Studies in France", Fusion Technology, Vol. 8, July 1985, p. 491.

(6) S. Grotz and N. Ghoniem, "Thermal Analysis of a Pin-Type Solid Breeder Blanket for Tokamak Reactors," UCLA Report UCLA-ENG-8612, 1986.

(7) J. Blanchard and N. Ghoniem, "Bowing of Breeder Pins in a Solid Breeder Fusion Blanket," UCLA Report UCLA-ENG-8605, 1986.

(8) G. Orient and N. Ghoniem, "Elastic Structural Analysis of the Pin-Type Solid Breeder Blanket First Wall," UCLA Report UCLA-ENG-8609, 1986.

(9) D.S. Gelles and M.L. Hamilton, "Effects of Irradiation on Low Activation Ferritic Alloys," Department of Energy Report, DOE/ER-0045/13, Sept. 1984.

(10) W.W. Engle, Jr., "A User's Manual for ANISN, a One-Dimensional Discrete Ordinates Transport Code with Anisotropic Scattering,: ORNL Report No. K-1693 (1967). Available also as Radiation Shielding Information Center Computer Code Collection CCC-248.

(11) Y. Gohar and M. Abdou, "MACK-LIB-IV: A Library of Nuclear Response Functions Generated with the MACK-IV Computer Program from ENDF/B-IV", Argonne National Laboratory, ANL/FPP/TM-106 (1978).

(12) E. Greenspan, W.G. Price, Jr. and H. Fishman, "SWAN: A Code for the Analysis and Optimization of Fusion Reactor Nucleonic Characteristics", Plasma Physics Laboratory, Princeton University, MATT-1008 (1973). Available also as Radiation Shielding Information Center Computer Code Collection CCC-248.

(13) R. Martin and N. Ghoniem, "Modeling of Tritium Transport in a Fusion Reactor Pin-Type Solid Breeder Blanket Using the DIFFUSE Code," Submitted to Intl. Conf. on Fusion Reactor Materials, Chicage, Il, April 1986.

ABSTRACT

In this report, a pressurized lobular blanket configuration is neutronically optimized. Among the features of this blanket configuration are the use of Beryllium and $LiAlO_2$ solid breeder pins in a cross-flow configuration in a Helium coolant. One-dimensional neutronic optimization calculations are performed to maximize the tritium breeding ratio (TBR). The procedure involves spatial allocations of Be, $LiAlO_2$, 9-C (Ferritic Steel), and He; in such a way as to maximize the TBR subject to several material, engineering and geometrical constraints. A TBR of 1.17 is achieved for a relatively thin blanket (= 43 cm depth), and consistency with all imposed constraints.