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TRITIUM BREEDING IN A FUSION-FISSION
HYBRID BREEDER REACTOR

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ABSTRACT

In order to construct a D-T fusion-fission hybrid breeder reactor there must be some guarantee that tritium will be available to fuel the fusion reaction. This can be achieved by breeding tritium in the blanket of the reactor. A variety of blanket configurations have been studied with the intent of arriving at a blanket design capable of attaining sizeable values of power and fissile fuel production, and simultaneously assure self-sufficiency in tritium production for the reactor, i.e., a T-breeding ratio greater than 1.0.

Using a four group diffusion theory code, a number of possible blanket configurations were studied during the parametric phase of hybrid breeder blanket design. They included tritium breeding zones of natural or enriched lithium metal and natural lithium in dilithium oxide, neutron multiplier zones of uranium and thorium with varying amounts of plutonium, thorium fissile breeding zones (sometimes containing plutonium), and graphite reflector zones. The choice of clad used in the fuel rods has also been found to influence tritium breeding.

One of the more promising designs studied combines a tritium breeding ratio of 1.2 with a blanket fissile power of 16 GW(th), and a U-233 production rate of 2.7 tonnes per year.

TABLE OF CONTENTS

<u>Section No.</u>		<u>Page No.</u>
I	INTRODUCTION	I-1
II	COMPUTATIONAL TECHNIQUES AND PRELIMINARY RESULTS	II-1
III	REFERENCE CASE CRITERIA AND RESULTS	III-1
IV	DISCUSSION OF RESULTS	IV-1
V	SUMMARY	V-1
	ACKNOWLEDGEMENT	V-3
VI	REFERENCES	VI-1

LIST OF FIGURES

<u>Figure No.</u>		<u>Page No.</u>
I-1	General Hybrid Breeder Reactor Geometry (Top View)	I-10
I-2	Fuel Lattice Description (View Along Minor Plasma Radius)	I-12
II-1	Tritium Breeding Ratio Vs. a/o Pu and Pu Zone Width for Design #1	II-8
III-1	TBR in a Li-(Th,Pu) C-ThC-C Blanket Vs. a/o Pu and (Th,Pu)C Zone Width, at 23.86 a/o Li-6 in Li.	III-5
III-2	TBR in a Li-(Th,Pu)C-ThC-C Blanket Vs. a/o Pu and a/o Li-6 in Li. (Th,Pu)C Zone Width is 6 cm.	III-6
III-3	Fissile Power in a Li-(Th,Pu)C-ThC-C Blanket Vs. a/o Pu and (Th,Pu)C Zone Width, at 23.86 a/o Li-6 in Li.	III-7
III-4	U-233 Production in a Li-(Th,Pu)C-ThC-C Blanket Vs. a/o and (Th,Pu)C Zone Width, at 23.86 a/o Li-6 in Li.	III-8
III-5	TBR in a Li-(U,Pu)C-ThC Blanket Vs. a/o Pu and (U,Pu)C Zone Width, at 0.5 a/o U-235 in U.	III-10
III-6	Pu Consumption in a Li-(U,Pu)C-ThC Blanket Vs. a/o Pu and (U,Pu)C Zone Width, at 0.5 a/o U-235 in U.	III-11
III-7	Fissile Power in a Li-(U,Pu)C-ThC Blanket Vs. a/o Pu and (U,Pu)C Zone Width, at 0.5 a/o U-235 in U.	III-12
III-8	U-233 Production in a Li-(U,Pu)C-ThC Blanket Vs. a/o Pu and (U,Pu)C Zone Width, at 0.5 a/o U-235 in U.	III-13
III-9	TBR in a (U,Pu)C-(Th,Pu)C-Li ₂ O-C Blanket Vs. a/o Pu in (U,Pu)C and (U,Pu)C Zone Width.	III-15
III-10	Pu Consumption in a (U,Pu)C-(Th,Pu)C-Li ₂ O-C Blanket Vs. a/o Pu in (U,Pu)C and (U,Pu)C Zone Width.	III-16
III-11	Fissile Power in a (U,Pu)C-Th,Pu)C-Li ₂ O-C Blanket Vs. a/o Pu in (U,Pu)C and (U,Pu)C Zone Width.	III-17
III-12	U-233 Production in a (U,Pu)C-(Th,Pu)C-Li ₂ O-C Blanket Vs. a/o Pu in (U,Pu)C and (U,Pu)C Zone Width.	III-18
IV-1	Tritium Breeding Cross Sections and Blanket Neutron Flux Spectrum Vs. Energy	IV-3

LIST OF TABLES

<u>Table No.</u>		<u>Page No.</u>
I-1	FUEL LATTICE SPECIFICATIONS	I-11
II-1	EXAMPLE OF THE THREE VARIABLE CENTRAL COMPOSITE DESIGN VALUES	II-3
II-2	EXAMPLE OF THE COMBINATIONS FOR A THREE VARIABLE CENTRAL COMPOSITE DESIGN	II-4
II-3	DESIGN #1 EXAMPLE OF PERTINENT DATA	II-7
II-4a	LISTING OF BLANKET DESIGNS CONSIDERED	II-10
II-4b	LISTING OF BLANKET DESIGNS CONSIDERED	II-11
II-4c	LISTING OF BLANKET DESIGNS CONSIDERED	II-12
III-1	COMPILATION OF DATA LEADING TO REFERENCE CASE #1	III-4
III-2	COMPILATION OF DATA LEADING TO REFERENCE CASE #2	III-9
III-3	COMPILATION OF DATA LEADING TO REFERENCE CASE #3	III-14
III-4	SUMMARY OF REFERENCE CASE DATA	III-19
IV-1	TRITIUM BREEDING CROSS SECTIONS OF LITHIUM BLANKET SPECTRUM AVERAGED BY ANISN	IV-4

I INTRODUCTION

At present the most promising fusion reaction is that of deuterium and tritium. The occurrence of a 14 Mev neutron from the reaction is fortunate since tritium does not exist naturally, but can be created by neutron induced reactions with Li-6 and Li-7. The first commercial fusion reactors may, therefore, have a plasma zone surrounded or partially surrounded by a lithium containing blanket. Problems with control and containment of the plasma in magnetically confined reactors impose design requirements on the power production by the fusion reactor in addition to that of tritium production. In an effort to lessen the constraints imposed on the fusion reactor and enhance power production, the use of a fissile breeding blanket was suggested by the U.S. Atomic Energy Commission's program in controlled nuclear research in 1951 (Ref. #1). The results presented in this paper pertain to the blanket of a fusion-fission hybrid breeder reactor, which was under study at Westinghouse's Fusion Power Systems Department prior to October of 1977.

The performance requirements imposed on the blanket will ultimately reflect the way the hybrid breeder will fit into the total energy production system of the area where it is to be built. Since the exact conditions under which the hybrid breeder will enter the energy picture are unknown, the requirements chosen for the blankets in this study were imposed as a basis on which calculations and preliminary design work could proceed. The four requirements arbitrarily established for the hybrid breeder are presented below.

The first of these is that the reactor be self-sufficient in tritium production. Since there are no major civilian facilities in existence for the production of tritium, consideration of self-sufficiency in tritium production seems a logical requirement for the first hybrid breeder. Fortesque (Ref. #2) argues that optimization of the performance of fission and fusion reactors should occur together. He concludes that tritium breeding may be easier in the fission reactors than in the hybrid reactor. The observation that this mutual arrangement could be more efficient is recognized; however, logistic and safety considerations make it desirable to design for self-sufficient in-plant tritium production. The calculations of the tritium breeding required for a reactor one-fifth the size of the hybrid breeder appear in reference #3, and specify a tritium breeding ratio (TBR) of 1.16, corresponding to approximately 100 kg of tritium produced per year, for the reactor under consideration. The value of 1.16 rather than 1.0, for the TBR, takes into account loss of tritium in handling and to radioactive decay.

The second requirement, set on the hybrid breeder blanket considered here, is to produce enough fissile fuel to supply five fission reactors of high conversion ratio, each using 180 kg makeup fissile material per year. The reactions available for fissile production are the conversion of U-238 to Pu-239 and the conversion of Th-232 to U-233 by capture reactions. Designs considered in this paper employ the thorium reaction to avoid proliferation of Pu-239. In one scenario the U-233 is mixed with depleted uranium, thus forming

a reactor grade (3% fissile) fuel that would be difficult to transform to weapons grade material without the use of gaseous diffusion equipment to separate and concentrate the fissile isotopes. The arbitrary choice of five fission reactors, corresponding to 0.9 tonnes per year fissile production, may be subject to large changes as future fuel requirements become more specific. To reflect this uncertainty the value of 1.5 tonnes per year will be used as a goal for fissile production.

The next requirement of the hybrid blanket is that it produce enough energy to power the hybrid breeder reactor, itself, and have power for sale. This implies that through the fast fission neutron multiplication within the blanket, enough thermal power be generated to supply, after conversion to electrical power, the neutral beams, magnetic field coils, auxiliary systems, etc. This has been assumed to require 3 GW(th). The blanket power is supplemented by approximately 250 MW(th) alpha particle power generated by the plasma. An additional 3 GW(th) is also assumed to be generated for sale to consumers. Thus, with a thermal to electrical conversion efficiency of 33%, a fission power of 7 to 8 GW(th) is set as a goal for the blanket. This power can be generated by blankets containing natural uranium or blankets enriched in plutonium obtained from the discharge of fission reactors. The composition of the discharge was assumed to be 69.7 a/o Pu-239 and 30.3 a/o Pu-240. Pu-241 was lumped with Pu-239 and Pu-242 with Pu-240. Pu-239 also plays the role of a fast fission neutron multiplier, enhancing the neutron production in a machine that must produce one triton, some fissile fuel, and fissile power from one fusion neutron.

The final requirement imposed on the hybrid breeder in this study is that of zero net plutonium production or consumption at the end of one year's irradiation . The fear of proliferation of weapons grade material, or a substance easily transformed to weapons grade material, is the reason for the requirement of zero net Pu-239 production, imposed on the hybrid as part of this study. With zero net consumption, the need for transportation of Pu to the reactor site, after initial loading, is avoided. Using the "same" plutonium from one loading to the next requires some sort of a reprocessing plant within the reactor facility. The reprocessing plant would be an added expense to the hybrid breeder project, however, stipulating zero plutonium production avoids the cost of purchasing Pu to replenish the blanket. Achievement of zero net Pu production (consumption) is accomplished by adding U-238 to the blanket in an amount that creates an equilibrium of Pu consumption and production. A more detailed discussion of the goals and the potential of a fusion-fission breeder is presented by Draper and Gage (Ref.4).

Early blanket designers recognized similar goals for their blankets, with the exception of zero plutonium production. Leonard (Ref.#5) summarizes seven early (prior to 1965) design studies. None of these studies were very specific, however, they all require some form of uranium and recognize the fact that tritium should be bred using a compound of lithium. One interesting concept involves the use of enriched uranium in a UO_2SO_2 coolant. This design, proposed by Barrett (Ref.#6), approached criticality

and bred tritium in regions outside the blanket by making use of the delayed fission neutrons.

The first of what Leonard terms the "moderd" fusion-fission hybrid concepts was developed by Lontai in a 1965 MIT thesis (Ref.#7). Lontai's study considered two zone blankets, a coolant zone and an attenuator zone. The coolant zone consisted of fused salts of lithium and beryllium (taking advantage of an $(n,2n)$ reaction in Be). The outer attenuator zone contained the same fused salts plus some natural uranium in the form of UF_4 . The total width of Lontai's blanket was approximately 50 cm, and was able to produce adequate amounts of tritium and fissile fuel (in the form of Pu-239). However, power production was four times less than the requirement set for the hybrid breeder studied here. Lontai's calculation also assumed an infinite slab wall loading of 5 MW/m^2 , of fusion neutrons, whereas the calculations presented in this paper use a 4 MW/m^2 wall loading in the cylindrical geometry of Figure I-1. Lontai's conclusion was that his calculations showed his concept of the blanket would be both feasible and practical, when applied to the real geometry.

In two articles (Ref. #8 and 9), J.D. Lee discusses sub-critical fast fissioning blankets. His designs are spherical blankets, consisting of two zones, with an inner radius of 200 cm and total widths of 100 cm. The first 30 cm zone is Li depleted in Li-6, and the second zone is depleted Li and depleted U in the ratio of 1/2. Lee's results are within the constraints placed on the hybrid breeder studied in this report, except for zero Pu produc-

tion; however, he has made no provision for the cooling of his blankets. Lee's work, which is complimented by some more recent designs developed by Cook and Maniscalco (Ref. #10), pertains to mirror and laser fusion machines and does not correctly model the toroidal geometry of the hybrid breeder studied here. Lee, Cook, and Maniscalco's work does specify blankets similar in structure and materials to the hybrid of Figure I-1, and their results meet the requirements set for the hybrid breeder studied here as far as fissile production (Pu production allowed), TBR, and blanket power multiplication are concerned.

In another recent blanket design, Su, Woodruff, and McCormick (Ref. #11) discuss designs similar to those studied here for the hybrid breeder. Their machine also has the goal of U-233 production, an equilibrium concentration of U-238 and Pu-239, self-sufficiency in tritium production, and an initial Pu loading of 30 tonnes. Su et al, report that their blanket is capable of producing 2.4 tonne U-233/yr with a blanket power of 10 GW(th). The major differences between their designs and the designs presented in this paper are the specifications of a 1.5 times smaller (volume) blanket, the use of Be as a neutron multiplier, the use of liquid lithium as a coolant, and no consideration of neutron leakage.

The survey of available literature on fusion blankets, given above, covers the designs for fast neutron blankets. These studies were mentioned, rather than articles treating thermalized spectra, because the hybrid breeder calculations described in this paper

used blankets with fast neutron spectra. The decision to use a fast spectrum in the blanket was made because fast spectra tend to give the higher fissile breeding rates. Furthermore, space considerations in the blanket limit expansion spaces for the fission gases, helium, and the tritium produced. Therefore, a decision to use vented fuel pins was made to keep fuel rods short. The choice of vented fuel pins, in turn, made helium the logical choice as coolant to prevent chemical reactions between coolant and fuel from occurring. Any attempt to add substantial amounts of moderator to the lattice would subtract from the space necessary for plutonium, thorium, lithium, etc. One final reason for the fast spectrum is that a higher neutron yield is obtained from fast fissions; this makes better use of each atom fissioned in terms of neutron multiplication. Leonard and Wolkenhauer (Ref.#12) and Jenquin, Leonard, Thomsen, and Wolkenhauer (Ref.#13) present discussions of thermal lattices for a fusion reactor blanket.

Before proceeding to the computational techniques and data, an explanation should be given concerning the nature and intent of the parametric calculations used as data in this paper. The parametric phase of calculations is intended to take in a wide range of possible designs. The designer is then able to infer the functional relationships of the dependent variables to the design variables. These relationships can be used to graphically map a design space. Design constraints can then be projected on the response surface, and the region acceptable for the required design can be located.

The method of calculation consisted of four-group diffusion theory calculations on a digital computer, with the hybrid reactor represented by a cylindrical geometry of Figure I-1. This figure represented a section of a top view of the tokamak reactor after a horizontal cut midway up the reactor. Progressing outward from the center of the reactor, the zones of interest are; the inner shield and reflector protecting the inner legs of the magnetic field coils, the inner tritium breeding zone providing approximately 35% of the tritium bred, the two liner zones around the plasma, the banded region containing a neutron multiplier zone, fissile breeding zone, T-breeding zone, and sometimes a neutron reflecting zone, and an outer reflector and shield protecting outer coils and the reactor exterior. For calculational purposes, each of these zones was considered homogenous. Note that the four blanket requirements, mentioned earlier for the hybrid, were to be satisfied by the banded region of Fig. I-1, with the exception of some tritium breeding which takes place in the inner zones of the blanket. A description of the fuel rod lattice represented by the banded region is summarized in Table I-1 and pictured in Figure I-2. Since transport theory was not being used in the calculations, except in cross section collapsing, high energy phenomena such as anisotropic scattering, resonance self-shielding, etc., are only approximately treated. The diffusion theory was used to perform a wide range of scoping calculations, as a preliminary step for more detailed calculations. The designs derived during the parametric calculations are termed reference cases, and are intended to serve as the basis for more comprehensive calculations with transport theory.

The remainder of this paper is divided into four sections. Section two presents the computational methods employed in the parametric phase of hybrid breeder blanket design. This section also contains a list of the various blanket configurations that led to the reference designs. Section three contains a detailed compilation of the data directly leading to the three reference cases. Section four contains a discussion of the results appearing in section three. The fifth section summarizes the results of section four, and presents a short discussion of the relationships between these results and further hybrid breeder blanket design.

The blanket designs presented here are for a stand-alone, tritium self-sufficient hybrid reactor, which would be a prolific producer of power and fissile material for sale. The designs represent the most demanding performance requirements, and the upper limit of the hybrid capability. In actual application, the hybrid may be integrated into a power park containing a number of light-water reactors, which could supply the hybrid breeder with part of its tritium and power requirement in exchange for fissile fuel. In this case less demanding hybrid design requirements would be applicable. Tritium breeding ratios less than 1.0 would be acceptable and the neutron economy in the hybrid would change substantially.

FIGURE I-1
GENERAL HYBRID BREEDER REACTOR GEOMETRY
(TOP VIEW)

I-10

TABLE I-1
FUEL LATTICE SPECIFICATIONS

Fuel pellet diameter, d	= 0.75 cm
Triangular rod pitch, P	= 0.95 cm
$\frac{P}{d_{\text{pellet}}} = 1.267$	$\frac{P}{d_{\text{clad}}} = 1.12$
Volume fraction of fuel	= 0.5652
clad	= 0.1507
He	= $\frac{0.2841}{1.0000}$
Wire wrap diameter	= 0.10 cm
Clad thickness	= 0.05 cm

FIGURE I-2
FUEL LATTICE DESCRIPTION
(VIEW ALONG MINOR PLASMA RADIUS)

II COMPUTATIONAL TECHNIQUES AND PRELIMINARY RESULTS

The parametric study described here was carried out with the use of a method adopted from the methodology of the design of experiments (Ref.#14). To examine the relationship of the breeding and energy generating characteristics of the hybrid breeder to the design variables, a number of central orthogonal designs were set up. The central orthogonal design method allows the designer to systematically examine the effects of changes in the design variables on dependent variables. The choice of the central orthogonal design method is made in order to reduce the number of combinations of design variables necessary to define a quadratic surface representing the dependent variable as a function of the design variables, and consequently reduce the number of calculations required.

As an example of the reduction in calculations obtained by using the central orthogonal design method, consider the relationship of the tritium breeding ratio to the width of the neutron multiplier zone, the percentage of fissile material in this zone, and the width of the lithium containing tritium breeding zone. To define a quadratic response of a dependent variable to one design variable requires three values of the design variable, and to define a quadratic response surface of a dependent variable to k design variables requires 3^k combinations of the design variables. This means that for the example mentioned above, the TBR would have to be determined for the twenty-seven possible

combinations of the three design variables in order to define the TBR as a quadratic function of them. Application of the central orthogonal design method to a design with k design variables, defines $2^k + 2k + 1$ combinations of the design variables, however, their values are not chosen arbitrarily. The designer decides upon a range of interest for each design variable. The center of this range is termed the "zero value" of the particular design variable. The half width of the range of interest is called the "scale factor," and two points are defined as the zero value plus and minus the scale factor. The central orthogonal design method defines two more points for each design variable as the zero value plus and minus a number " δ " times the scale factor. For a three variable design δ is 1.2154, and for a two variable design δ is 1.0 so that no new values of the design variable are defined. Possible values for the three design variables of the example are tabulated in Table II-1. The combinations of these variables, as designated by the central composite design method, appear in Table II-2.

After the design variable values and their combinations are defined, the designer calculates the value of the dependent variable for each combination. For the three design variable case, the computed results are then fit to the equation;

$$y = a_{11}x_1^2 + a_{22}x_2^2 + a_{33}x_3^2 + a_{12}x_1x_2 + a_{13}x_1x_3 + a_{23}x_2x_3 + a_1x_1 + a_2x_2 + a_3x_3 + a_0$$

TABLE II-1
EXAMPLE OF THE THREE VARIABLE
CENTRAL COMPOSITE DESIGN VALUES

1) NEUTRON MULTIPLIER ZONE WIDTH

Coded Variable X_1	Real Variable X_1	
-1.2154	8.923	
-1.0	10.0	
0.0	15.0 = $\frac{\text{Zero}}{\text{Value}}$	SCALE FACTOR = $\frac{5.0 \text{ cm}}{\text{unit } X_1}$
1.0	20.0	
1.2154	21.077	

2) a/o FISSILE MATERIAL IN NEUTRON MULTIPLIER ZONE

Coded Variable X_2	Real Variable X_2	
-1.2154	7.5692	
-1.0	8.0	
0.0	10.0 = $\frac{\text{Zero}}{\text{Value}}$	SCALE FACTOR = $\frac{2.0 \text{ a/o}}{\text{unit } X_2}$
1.0	12.0	
1.2154	12.4308	

3) TRITIUM BREEDING ZONE WIDTH (cm)

Coded Variable X_3	Real Variable X_3	
-1.2154	17.846	
-1.0	20.0	
0.0	30.0 = $\frac{\text{Zero}}{\text{Value}}$	SCALE FACTOR = $\frac{10.0 \text{ cm}}{\text{unit } X_3}$
1.0	40.0	
1.2154	42.154	

TABLE II-2

EXAMPLE OF THE COMBINATIONS FOR
A THREE VARIABLE CENTRAL COMPOSITE DESIGN

CASE	CODED VARIABLES			REAL VARIABLES		
	X_1	X_2	X_3	X_1 (cm)	X_2 (a/o)	X_3 (cm)
1	-1.2154	0.0	0.0	8.923	10.0	30.0
2	-1.0	-1.0	-1.0	10.0	8.0	20.0
3	-1.0	-1.0	1.0	10.0	8.0	40.0
4	-1.0	1.0	-1.0	10.0	12.0	20.0
5	-1.0	1.0	1.0	10.0	12.0	40.0
6	0.0	-1.2154	0.0	15.0	7.569	30.0
7	0.0	0.0	-1.2154	15.0	10.0	17.846
8	0.0	0.0	0.0	15.0	10.0	30.0
9	0.0	0.0	1.2154	15.0	10.0	42.154
10	0.0	1.2154	0.0	15.0	12.431	30.0
11	1.0	-1.0	-1.0	20.0	8.0	20.0
12	1.0	-1.0	1.0	20.0	8.0	40.0
13	1.0	1.0	-1.0	20.0	12.0	20.0
14	1.0	1.0	1.0	20.0	12.0	40.0
15	1.2154	0.0	0.0	21.077	10.0	30.0

where y is the dependent variable, and x_1, x_2 , and x_3 are the independent variables. The computer code PARDES II (Ref. #15) calculates the coefficients of the equation and five numbers related to the goodness of fit of the assumed equation form to the actual data. PARDES also extrapolates the data by allowing one to input values of the independent variables outside the range of interest. It must be remembered that these extrapolations fit a quadratic curve containing calculated data points, however a calculation at the extrapolation point may yield a different value than PARDES predicts. A detailed explanation of the central orthogonal design method appears in Anderson and McLean (Ref. #14).

The bulk of the neutronic calculations was carried out by the four group diffusion theory code FORGOD (Ref. #16). Cross sections for FORGOD were collapsed and mixed by the codes ANISN (Ref. #19) and PAN (Ref. #17), from the DLC-2F library (Ref. #18). Vacuum spaces and helium coolant passages were given zero absorption, and a diffusion coefficient of 1000 cm. The code operated in the cylindrical geometry of Figure I-1. The radii and compositions of the various zones, a 14 Mev neutron source (calculated to model a 4 MW/m^2 wall loading), the energy limits of the four groups, and a transverse dimension were the other major input parameters required for the calculations. The main output consisted of the four group fluxes at 250 radial mesh points and the absorption rates at these points. Fission power densities, total power, and the eigenvalue for the reactor, with the source turned off, are also printed. Fluxes from FORGOD are

indirectly input to the code RBURN (Ref. #17), with the appropriate material data to obtain the burn-up characteristics of the blanket. RBURN output parameters include; U-233 production, Pu-239 production or consumption, thorium consumption, and time dependent energy production rates. The tritium breeding rates are calculated from the fluxes in the lithium regions. All the principal performance parameters obtained by these neutronics codes were then correlated to the primary design variables through the use of the PARDES code, and response surfaces were plotted.

The first configuration considered consisted of a (U,Pu)C neutron multiplying and energy generating zone, a ThC fissile breeding zone, and an Li_2O tritium breeding zone. All three zones have TZM clad in the lattice described in section I. A two-variable central composite design, with the atom percentage of Pu in the (U,Pu)C zone and the zone's thickness being the design variables, yielded the results listed in Table II-3. Since a two-variable design was chosen, there are enough data points to plot response surfaces without a PARDES run. Figure II-1 shows the response of the TBR as a function of the two design variables. Similar plots could have been made for the other variables of interest. Ranges of interest can then be superimposed (for example, the TBR of 0.5 in Figure II-1), and intersections of these regions from various plots become design points, or starting points for further computation. A PARDES run on the data used for Figure II-1 would have allowed specification of larger values of a/o Pu and (U,Pu)C zone width, and could point the way to calculations that would bring the TBR up to unity.

TABLE II-3

DESIGN #1 EXAMPLE OF
PERTINENT DATA

Case	Design Variables a/o Pu in (U,Pu)C	(U,Pu)C Zone Width	TBR	Blanket Power	Pu-239 Prod.	U-233 Prod.
#	a/o	cm	-	GW(th)	tonne/yr	tonne/yr
1	0.0	2.0	0.44	1.2	0.48	3.71
2	0.0	6.0	0.46	2.0	1.47	3.18
3	0.0	10.0	0.47	2.5	2.39	2.60
4	5.0	2.0	0.45	1.5	0.34	3.83
5	5.0	6.0	0.48	2.9	1.09	3.51
6	5.0	10.0	0.51	4.2	1.88	3.08
7	10.0	2.0	0.46	1.8	0.18	3.96
8	10.0	6.0	0.51	4.1	0.63	3.91
9	10.0	10.0	0.56	6.6	1.19	3.75

FIGURE II-1 Tritium Breeding Ratio Vs. a/o Pu and Pu Zone Width for Design #1

A detailed listing of all the data for each complete central composite design studied during the preliminary hybrid breeder parametric calculations would not be enlightening. The series of changes between designs (designs are listed in Table II-4) two, three, and four reflect the need to obtain TBR's greater than unity. Other changes result from a restatement of the ground rules. Uranium was removed between designs one and two in order to avoid Pu-239 production. Cases two, three, four, and five form a sequence which leads to reference case one. Another change in ground rules leads to the sequence six and seven. Uranium was allowed in the designs in an effort to create a zero plutonium producer. The production of Pu-239 in U-238 is to directly balance the consumption of Pu-239 in fission. Design case seven leads to the second reference case. Cases eight and nine are an effort to back track, and try designs without a clearcut multiplier region. Case nine has the substitution of stainless steel clad for TZM and leads to the third reference case. In general, it can be stated that on the basis of these preliminary runs, the design requirement most difficult to meet was the self-sufficiency in tritium breeding, without sacrificing adequate values of fissile fuel and power production.

TABLE II-4a
LISTING OF BLANKET DESIGNS CONSIDERED

1)	(Pu,U)C - ThC - Li ₂ O	TZM Clad		
	<u>Design Variables</u>	<u>Zero Value</u>	<u>Scale Factor</u>	
	1) a/o Pu in (Pu,U) zone	5a/o	5a/o per unit X ₁	
	2) Pu zone width	6 cm	4 cm per unit X ₂	
	Comments;			
	((Pu,U)C & ThC) width is 50 cm			
	Li ₂ O width is 20 cm			TBR _{max} = 0.56
	U in (Pu,U)C and Li in Li ₂ O are natural			
2)	(Th,Pu)C - Li ₂ O - ThC - C	TZM Clad		
	<u>Design Variables</u>	<u>Zero Value</u>	<u>Scale Factor</u>	
	1) a/o Pu in (Th,Pu)C zone	5a/o	4.114a/o per unit X ₁	
	2) (Th,Pu)C zone width	6 cm	4 cm per unit X ₂	
	3) ThC zone width	35 cm	5 cm per unit X ₃	
	Comments;			
	((Th,Pu)C & Li ₂ O) width is 20 cm			
	(ThC & C) width is 50 cm			TBR _{max} = 0.63
	Li in Li ₂ O is natural			
3)	Li ₂ O - (Th,Pu)C - ThC - C	TZM Clad		
	<u>Design Variables</u>	<u>Zero Value</u>	<u>Scale Factor</u>	
	1) a/o Pu in (Th,Pu)C zone	5a/o	4.114a/o per unit X ₁	
	2) (Th,Pu)C zone width	6 cm	4 cm per unit X ₂	
	3) ThC zone width	35 cm	5 cm per unit X ₃	
	Comments;			
	((Th,Pu)C & Li ₂ O) width is 20 cm			
	(ThC & C) width is 50 cm			TBR _{max} = 0.56
	Li in Li ₂ O is natural			

TABLE II-4b
LISTING OF BLANKET DESIGNS CONSIDERED (cont.)

4)	Li - (Th,Pu)C - ThC - C	TZM Clad	
	<u>Design Variables</u>	<u>Zero Value</u>	<u>Scale Factor</u>
	1) a/o Pu in (Th,Pu)C zone	5a/o	4,114a/o per unit X_1
	2) (Th,Pu)C zone width	6 cm	4 cm per unit X_2
	3) ThC zone width	35 cm	5 cm per unit X_3
	Comments:		
	(Li & (Th,Pu)C) width is 20 cm		
	(ThC & C) width is 50 cm		$TBR_{max} = 0.76$
	Li is 40.2a/o Li-6 and is a solid zone (no clad,no void)		
5)	Li - (Th,Pu)C - ThC - C	TZM Clad	
	<u>Design Variables</u>	<u>Zero Value</u>	<u>Scale Factor</u>
	1) a/o Pu in (Th,Pu)C zone	35a/o	5a/o per unit X_1
	2) (Th,Pu)C zone width	6 cm	4 cm per unit X_2
	3) a/o Li-6 in Li zone	23.86a/o	13.44a/o per unit X_3
	Comments:		
	(Li & (Th,Pu)C) width is 20 cm		
	ThC width is 30 cm, C width is 20 cm		$TBR_{max} = 2.13$
6)	Li ₂ O - (U,Pu)C - ThC	TZM Clad	
	<u>Design Variables</u>	<u>Zero Value</u>	<u>Scale Factor</u>
	1) a/o Pu in (U,Pu)C zone	10a/o	5a/o per unit X_1
	2) a/o U-235 in U	0.46a/o	0.21a/o per unit X_2
	3) (U,Pu)C zone width	6 cm	4 cm per unit X_3
	Comments:		
	(Li ₂ O & (U,Pu)C) width is 30 cm		
	Li is natural		$TBR_{max} = 0.87$
	ThC width is 40 cm		

TABLE II-4c
LISTING OF BLANKET DESIGNS CONSIDERED (cont.)

- 7) Li - (U,Pu)C - ThC - C TZM Clad

<u>Design Variables</u>	<u>Zero Value</u>	<u>Scale Factor</u>
1) a/o Pu in (U,Pu)C zone	10a/o	5a/o per unit X_1
2) a/o U-235 in U	0.46a/o	0.21a/o per unit X_2
3) (U,Pu)C zone width	6 cm	4 cm per unit X_3

Comments;

(Li & (U,Pu)C) width is 30 cm

$$TBR_{\max} = 0.98$$

Li is 40.2a/o Li-6 and is a solid zone

ThC width is 40 cm

- 8) (U,Pu)C - (Th,Pu)C - Li₂O - C TZM Clad

<u>Design Variables</u>	<u>Zero Value</u>	<u>Scale Factor</u>
1) a/o Pu in (U,Pu)C zone	10a/o	6a/o per unit X_1
2) a/o Pu in (Th,Pu)C zone	5a/o	5a/o per unit X_2

Comments;

(U,Pu)C width is 10 cm

(Th,Pu)C width is 10 cm

Li₂O width is 25 cm and Li is natural

C width is 25 cm

- 9) (U,Pu)C - (Th,Pu)C - Li₂O - C SS-316 Clad

<u>Design Variables</u>	<u>Zero Value</u>	<u>Scale Factor</u>
1) a/o Pu in (U,Pu)C zone	10a/o	5a/o per unit X_1
2) (U,Pu)C zone width	10 cm	5 cm per unit X_2

Comments;

((U,Pu)C & (Th,Pu)C) width is 20 cm

(Th,Pu)C zone contains 5a/o Pu

$$TBR_{\max} = 1.50$$

Li₂O zone width is 35 cm and Li is natural

C zone width is 15 cm

III REFERENCE CASE CRITERIA AND RESULTS

As stated earlier the reference cases are not to be considered finished designs. They are meant to combine the good features of the calculations that lead to them. The reference cases and the central composite orthogonal design sets, immediately preceeding them, are presented in this section. The data of the three central composites are included in order to observe the interdependence of a number of the dependent variables on the design variables and each other. The reference case parameters themselves, although not refined, are a basis for calculations that more accurately describe the blanket's behavior.

The first central composite orthogonal design set to yield a reference case is number five from the previous section. Table III-1 compiles the data of the fifteen FORGOD and RBURN runs. This design has a number of cases with TBR's greater than 1.0, and also is the only one that shows the variation in TBR with variations in the isotopic composition of lithium. Figures III-1 and 2 are plots of the TBR versus a/o Pu and (Th,Pu)C zone width, and a/o Pu and a/o Li-6 in Li respectively. Also plotted versus a/o Pu and (Th,Pu)C zone width are fissile power and U-233 production in Figures III-3 and 4. Pu production (consumption) is not plotted for this central composite because, at the time of the run, there was no restriction on Pu consumption. This hybrid breeder was being considered as a dump for the Pu-239 created in the uranium

powered fission reactors. The choice of a reference case from this central composite takes a number of paths based on TBR, U-233, and fissile powers of Table III-1, however case 15 is the only one with an initial Pu inventory below 40 tonnes. This case is chosen for reference case #1 and the results are tabulated beside the others, in Table III-4 (following Figure III-12).

The central composite leading to reference case #2 is summarized in Table III-2. This central composite does not have any cases that meet the requirements of $TBR > 1.0$, $U-233 > 1.5$ tonne/yr., or fissile power > 4 GW-th. It does approach zero Pu production and does not require a large initial Pu inventory in order to achieve a $TBR \approx 1.0$. The PARDES runs of this central composite were extrapolated from 16a/o Pu, in the (U,Pu)C zone, to 50a/o. The width of the zone is slightly extrapolated also. After noting that the dependence on a/o U-235 is small (for the small values used), the TBR, Pu consumption, fissile power, and U-233 production are plotted as functions of a/o Pu in (U,Pu)C and (U,Pu)C zone width in Figures III-5 through 8. The reference case parameters are chosen in the first two figures. Drawing the $TBR=1$ line taken from Figure III-5 in Figure III-6 allows one to find the point of zero Pu production with a TBR of unity. The result is fortunate in that fissile power and U-233 production are about right, and, since the reference point does not extend much beyond the region of calculation, one can expect a fairly accurate prediction. The values of the design variables selected for the reference case are 12.5 a/o Pu in (U,Pu)C, (U,Pu)C zone width of 12 cm, and 0.5 a/o U-235 in U. The results appear in Table III-4.

The final central composite is summarized in Table III-3. The ranges covered by the dependant variables are much larger than either of the other central composites for similar changes in the design variables. With any comparison of this set of designs to another, the change in clad from TZM to stainless steel must be recalled. The reference case is again chosen by means of the combination of the $TBR \approx 1$ line and zero plutonium production line of Figures III-9 and 10. An acceptable TBR and Pu production does not require extrapolation beyond the range of experimental data. Figures III-11 and 12 show values of 11 GW-th fission power and 2.75 tonne/yr U-233 production. The value of the design variables used for reference case #3 are 14 a/o Pu in the (U,Pu)C zone and a zone width of 11.56 cm. These are not the variables marked on the plots; they have been shifted along the zero Pu production line to a point where $TBR \approx 1.2$, so the reference case will not duplicate the power and uranium production stated here. The actual results appear in Table III-4.

Figure III-3 points out the approximate nature of the calculations performed in the parametric phase of hybrid breeder design. In the plane of 25 a/o Pu in the (Th,Pu)C zone, fissile power is increasing as the zone width is decreased from 6 cm to 0 cm. Results of this kind are not uncommon when the graphed values of the design variables exceed the original ranges set on them.

TABLE III-1

COMPILATION OF DATA LEADING TO REF. CASE #1

Li-(Th,Pu)C-ThC-C

CASE	REAL VARIABLES			INITIAL INVENTORY		TBR	FISSILE POWER (BOL)	U-233 PROD.	PU CONS.	EIGEN- VALUE *
	a/o Pu	(Pu,Th)C WIDTH	a/o Li-6	Pu	Th					
#	a/o	cm	a/o	tonne	tonne	atm. T/fus.	GW-th	tonne/yr.	tonne/yr.	---
1	28.9	6.00	23.86	27.1	637	0.85	3.60	3.2	1.90	0.44
2	30.0	2.00	10.42	9.40	593	0.65	1.40	2.4	0.91	0.22
3	30.0	2.00	37.30	9.40	593	0.84	0.22	2.0	0.58	0.16
4	30.0	10.0	10.42	46.6	677	0.97	14.1	6.0	3.89	0.74
5	30.0	10.0	37.30	46.6	677	1.08	8.03	4.6	2.76	0.64
6	35.0	1.14	23.86	6.20	583	0.76	0.76	2.0	0.56	0.13
7	35.0	6.00	7.520	32.8	631	0.77	7.37	4.5	3.55	0.61
8	35.0	6.00	23.86	32.8	631	0.92	4.80	3.6	2.81	0.52
9	35.0	6.00	40.20	32.8	631	0.98	3.80	3.3	2.38	0.47
10	35.0	10.9	23.86	59.1	679	1.42	19.9	7.7	5.02	0.81
11	40.0	2.00	10.42	12.5	590	0.67	1.80	2.6	1.68	0.27
12	40.0	2.00	37.30	12.5	490	0.86	1.20	2.1	1.12	0.20
13	40.0	10.0	10.42	62.2	663	2.13	49.1	12.2	7.49	0.91
14	40.0	10.0	37.30	62.2	663	1.53	17.0	7.3	5.63	0.79
15	41.1	6.00	23.86	38.4	626	1.00	6.30	4.1	4.08	0.59

*With the fusion neutron source turned off.

FIGURE III-1. TBR in a Li-(Th,Pu) C-ThC-C Blanket Vs. a/o Pu and (Th,Pu)C Zone Width, at 23.86 a/o Li-6 in Li.

FIGURE III 2. TBR in a Li-(Th,Pu)C-ThC-C Blanket Vs. a/o Pu and a/o Li-6 in Li.
(Th,Pu)C Zone Width is 6 cm.

FIGURE III-3. Fissile Power in a Li-(Th,Pu)C-ThC-C Blanket Vs. a/o Pu and (Th,Pu)C zone Width, at 23.86 a/o Li-6 in Li.

FIGURE III-4. U-233 Production in a Li-(Th,Pu)C-ThC-C Blanket Vs. a/o and (Th,Pu)C Zone Width, at 23.86 a/o Li-6 in Li.

TABLE III-2

COMPILATION OF DATA LEADING TO REF. CASE #2

Li-(U,Pu)C-ThC

CASE	REAL VARIABLES			INITIAL INVENTORY		TBR	FISSILE POWER	U-233 PROD.	Pu CONS.	Eigen- VALUE **
	a/o Pu IN (U,Pu)C	a/o U-235 IN U	(U,Pu)C WIDTH	Pu	U*					
#	a/o	a/o	cm	tonne	tonne	atm. T/fus.	GW-th	tonne/yr.	tonne/yr.	----
1	3.92	.4555	6.00	3.73	82.7	0.91	1.05	0.94	-0.03	0.19
2	5.00	.2453	2.00	1.59	27.7	0.90	0.38	0.89	-0.08	0.07
3	5.00	.2453	10.0	7.90	137.	0.90	1.48	0.97	-0.05	0.22
4	5.00	.6657	2.00	1.59	27.7	0.90	0.39	0.89	-0.08	0.07
5	5.00	.6657	10.0	7.90	137.	0.92	1.53	0.95	-0.05	0.23
6	10.0	.2000	6.00	9.51	82.7	0.90	1.17	0.97	-0.16	0.21
7	10.0	.4555	1.14	1.81	15.8	0.92	0.32	0.89	-0.03	0.06
8	10.0	.4555	6.00	9.51	82.7	0.93	1.18	0.97	-0.16	0.21
9	10.0	.4555	10.9	17.1	149.	0.92	2.38	1.09	-0.31	0.35
10	10.0	.7110	6.00	9.51	82.7	0.91	1.19	0.97	-0.16	0.21
11	15.0	.2453	2.00	4.77	27.7	0.97	0.53	0.91	0.00	0.11
12	15.0	.2453	10.0	23.7	137.	0.91	2.87	1.20	-0.02	0.41
13	15.0	.6657	2.00	4.77	27.7	0.97	0.52	0.91	0.0	0.11
14	15.0	.6657	10.0	23.7	137.	0.98	2.93	1.21	-0.02	0.41
15	16.1	.4555	6.0	15.3	82.7	0.94	1.57	1.05	+0.02	0.28

*Initial Thorium Inventory = 637 tonne

** With the fusion neutron source turned off.

FIGURE III-5. TBR in a Li-(U,Pu)C-ThC Blanket Vs. a/o Pu and (U,Pu)C Zone Width, at 0.5 a/o U-235 in U.

FIGURE III-6. Pu Consumption in a Li-(U,Pu)C-ThC Blanket Vs. a/o Pu and (U,Pu)C Zone Width, at 0.5 a/o U-235 in U.

FIGURE III-7. Fissile Power in a Li-(U,Pu)C-ThC Blanket Vs. a/o Pu and (U,Pu)C Zone Width, at 0.5 a/o U-235 in U.

FIGURE III-8. U-233 Production in a Li-(U,Pu)C-ThC Blanket Vs. a/o Pu and (U,Pu)C Zone Width, at 0.5 a/o U-235 in U.

TABLE III-3

COMPILATION OF DATA LEADING TO REF. CASE #3

(U,Pu)C-(Th,Pu)C-Li₂O-C

CASE	REAL VARIABLES		(Th,Pu)C WIDTH	INITIAL INVENTORIES			TBR	FISSILE POWER (BOL)	U-233 PROD.	Pu CONS.	EIGEN- VALUE *
	a/o Pu In (U,Pu)C	(U,Pu)C WIDTH		Pu	U	Th					
#	a/o	cm	cm	tonne	tonne	tonne	atm.T/fus	GW-th	tonne/yr.	tonne/yr.	----
1	5.0	5.0	15.	15.3	72.3	214.	0.68	4.90	2.9	-0.21	0.35
2	5.0	10.	10.	15.3	145.	143.	0.75	6.20	1.8	-1.60	0.40
3	5.0	15.	5.0	15.3	219.	71.9	0.80	7.10	0.8	-3.00	0.45
4	10.	5.0	15.	19.2	68.2	214.	0.74	6.30	3.2	+0.26	0.42
5	10.	10.	10.	23.0	137.	143.	0.88	9.60	2.2	-0.75	0.54
6	10.	15.	5.0	26.9	206.	71.9	1.03	13.0	1.1	-2.00	0.62
7	15.	5.0	15.	23.0	64.7	214.	0.81	8.10	3.5	+0.85	0.49
8	15.	10.	10.	30.7	130.	143.	1.08	14.0	2.8	+0.50	0.66
9	15.	15.	5.0	38.4	196.	71.9	1.50	25.0	1.7	-0.07	0.77

*With the fusion neutron source turned off.

FIGURE III-9. TBR in a (U,Pu)C-(Th,Pu)C-Li₂O-C Blanket Vs. a/o Pu in (U,Pu)C and (U,Pu)C Zone Width.

FIGURE III-10. Pu Consumption in a (U,Pu)C-(Th,Pu)C-Li₂O-C Blanket Vs. a/o Pu in (U,Pu)C and (U,Pu)C Zone Width.

FIGURE III-11. Fissile Power in a (U,Pu)C-(Th,Pu)C-Li₂O-C Blanket Vs. a/o Pu in (U,Pu)C and (U,Pu)C Zone Width.

FIGURE III-12. U-233 Production in a (U,Pu)C-(Th,Pu)C-Li₂O-C Blanket Vs. a/o Pu in (U,Pu)C and (U,Pu)C Zone Width.

TABLE III-4
SUMMARY OF REFERENCE CASE DATA

	REFERENCE CASE		
	#1	#2	#3
TBR	1.00	1.16	1.18
Pu PRODUCTION (tonne/yr)	-4.08*	0.23	-0.07
Pu INVENTORY (tonne)	38.4	23.6	33.2
EIGENVALUE	0.59	0.43	0.67
U-233 PRODUCTION (tonne/yr)	4.10	1.10	2.67
POWER (BOL) (GW-th)	6.3	3.26	15.9
POWER (1 yr) (GW-th)	5.9	3.31	18.0
BURN UP (1 yr) (GW-D/tonne)	36.8	5.5	27.8

* (-) MEANS CONSUMPTION

IV DISCUSSION OF RESULTS

In the course of the parametric calculations the tritium breeding requirement was the most difficult to satisfy and still keep a realistic design. The first two reference cases can not be called realistic since the lithium zones have the idealized characteristics of a solid's density, and no consideration of cooling or structure, although if density is adjusted properly these solid zones would correctly model liquid lithium flowing through the blanket. Because the hybrid breeders will probably be required to replenish their own tritium supply, the placement and composition of the tritium breeding zone becomes a matter of concern.

The tritium breeding zone can not be thought of as a pure absorbing medium. The high energy tritium breeding reaction of Li-7 conserves the neutron through inelastic scattering. The lower energy neutron is then useful for the tritium producing reaction of Li-6, but this reaction will not conserve a free neutron. Therefore, a pure zone of Li-7 wastes a low energy neutron (unless it can be easily transported to a zone that can make use of it), and a pure zone of Li-6 makes little use of the high energy neutrons that enter it. Some combination of the two isotopes will maximize tritium production, and the ratio of the two will depend on the flux spectrum existing in the region occupied by the lithium. Since reference case one uses Li-6 atom percentage as a design variable, it serves to illustrate some of the relationships between the TBR and the amount of Li-6 present in the lithium zone. Figure III-1

indicates that for large amounts of plutonium, and consequently high eigenvalues, the tritium breeding ratio becomes large. The increased neutron multiplication, taking place as the multiplier zone approaches criticality, explains the excess number of neutrons available to the lithium. The continued increase in initial plutonium loading of the hybrid blanket is limited by the finite amount of Pu-239 thought to exist (presently, approximately 30 tonnes), and the cost of obtaining and processing the plutonium. Figure III-2 suggests that for a given neutron multiplier zone thickness, there should be an inverse relationship between plutonium inventory and Li-6 enrichment to achieve the largest TBR. The lithium tritium breeding cross section curves appear in Figure IV-1, and in the lowest energy group the plutonium fission cross section is the same order of magnitude as the Li-6 (n,t) reaction. The inverse relationship is a result of the nearly equal competition between Li-6 and Pu-239 for the limited number of low energy neutrons. The fact that the lithium zone is closer to the plasma than the plutonium zone also contributes to higher TBR's at lower Li-6 enrichments, because Li-7 can make better use of the 14 Mev fusion neutrons. Table IV-1, summarizing the blanket spectrum averaged tritium breeding cross sections used in the calculations, shows that Li-7 can make ten times better use of these neutrons. The Li-7 reaction conserves a free neutron which can then enter the plutonium zone, with a larger probability of causing a fission than the 14 Mev neutron.

The effect of fission neutrons on the lithium zone is better when

FIGURE IV-1

TRITIUM BREEDING CROSS SECTIONS & BLANKET NEUTRON FLUX SPECTRUM vs. ENERGY

the Li-6 enrichment is high. The fission spectrum, used in the calculations, placed 35% of the neutrons in group two and 65% in group three. The lithium cross sections, and a short calculation, show that the tritium breeding contributed by Li-6, from fission neutrons, is greater than that from Li-7 when the Li-6 concentration is greater than 15 a/o.

TABLE IV-1
TRITIUM BREEDING CROSS SECTIONS OF LITHIUM
BLANKET SPECTRUM AVERAGED BY ANISN

ENERGY GROUP	CROSS SECTION (b)	
	Li-6	Li-7
I	0.033	0.33
II	0.104	0.244
III	0.859	0.0
IV	6.43	0.0

This reasoning leads one to suspect that the TBR graph at 45 a/o Pu, in Figure III-2, could curve back up if the Li-6 enrichment were allowed to increase past 40 a/o. Consideration of enrichment difficulties make large Li-6 enrichments undesirable due to cost increase, and calculations beyond 40 a/o were considered unnecessary. Applying these ideas to the other two reference cases might give one cause to change them, and it may be advantageous to lower the concentration of Li-6 in order to allow more of the neutron conserving reactions in Li-7. This would decrease the TBR somewhat, but more low energy neutrons would be available to fission

Pu-239 and breed U-233 in thorium. The third reference case uses Li_2O with natural lithium. This tritium breeding zone is separated from the plasma by the zones containing fissile material, hence it sees less of the highest energy neutrons and more of the fission neutrons. For this kind of a spectrum, Li-6 is a better tritium breeder. This means that a value of the TBR greater than that listed in Table III-4 could be obtained if the lithium were enriched.

The down scattering characteristics of the other elements in the lithium compound should also play some part in the compound's effectiveness as a tritium breeder. A lithium compound containing a very good moderator should contain a larger proportion of Li-6 to take advantage of the lower energy flux spectrum that can be expected. The choice of the compound in which the lithium appears depends on the parasitic absorption in the compound, and the atom densities the lithium occupies in the compound. The calculations performed as a basis for this paper did not contain any form of lithium other than metal or oxide, because data was not readily available on more exotic compounds such as plumbates or aluminates. The thermal absorption cross sections of lead (180 mb) and aluminum (243 mb), in comparison with that of oxygen (0.27 mb), show that the other compounds will probably have detrimental nuclear effects, however the question remains as to their mechanical performance. The oxygen in dilithium oxide will probably have little effect on the tritium breeding, since the lithium atom density remains nearly the same in a solid metal zone and an oxide zone with clad

and cooling void. Some of the sequences listed in section II progress from an Li_2O zone to solid metal under the assumption that the oxygen was absorbing too many of the neutrons. The cross sections show this assumption to be in error, and the designs leading to reference case three suggest that the TZM clad was parasitically absorbing too many neutrons. It was hoped that the TZM would take advantage of the high energy $(n,2n)$ reaction in molybdenum, however this reaction does not seem to occur enough to offset the lower energy absorption reaction.

A comparison of the tritium breeding and fissile power plots of section III, point out a direct correlation between the fissile power and TBR. This is not surprising since the power comes, for the most part, from the neutron multiplier zone, and as the power in this zone goes up, the number of neutrons available to leak into other zones also increases. It should also be noted that this direct correlation exists whether the lithium zone is in front of or behind the other zones. A comparison that may yield more information about tritium production, is that between TBR and the initial fissile inventory of the blanket. Referring to Table III-4, it is evident that more neutron multiplier (Pu inventory) does not always yield a higher TBR. The TBR will depend on how efficiently the fissile fuel is used and how well it is coupled to the lithium zone. The efficiency of the neutron multiplying zone is reflected directly as the eigenvalue (k -effective of the blanket), and inversely as the fissile inventory. An indication of how well the

neutron multiplier zone and lithium zone couple, is given by a comparison of the tritium breeding performances and the other performance variables; for example, in reference case two the TBR is satisfactory, however power and fissile production are low. This implies the neutron multiplying zone is coupled very well to the lithium zone and not too well to the other zones. A comparison of reference cases one and two show that if the lithium is to be in front of the Pu zone, better coupling is obtained if the Pu zone is wider with a lower concentration of plutonium. Comparison of these cases to reference case three indicates that good coupling can be achieved with the lithium zone behind the Pu zone without sacrificing good coupling to the other zones, therefore it is advantageous to put the lithium zone behind the Pu zone if a $TBR > 1.0$ can be achieved.

From the calculations presented, the choice of clad for the fuel rods seems obvious. The switch from TZM (used for most of the parametric calculations) to stainless steel made meeting the requirements set on the performance variables much easier. Consideration of the remarks made about the calculations in section I, suggests that before a final choice is made, a detailed calculation be performed to confirm the superiority of stainless steel.

One final point, concerning the calculational procedure and how it relates to the interpretation of the results, should be made. The four group cross sections used in the diffusion theory description of this hybrid breeder blanket are neutron flux averaged by ANISN from 100 group transport theory cross sections

using the scalar flux spectrum of Figure IV-1. This flux spectrum was generated by a cylindrical ANISN solution to one of the early blanket designs; it is taken from the center of the outer blanket. Since all of the four group cross sections are weighted using this spectrum, FORGOD solutions are best when the fluxes it generates have the approximate shape of the weighting flux. Fitting the four group fluxes from FORGOD to the spectrum in Figure IV-1 could not duplicate the curve very well. But, in the region of interest (the banded region of Figure I-1), FORGOD fluxes always peaked in the third energy group, were slightly lower but of the same order of magnitude in the first and second groups, and dropped one or two orders of magnitude between groups three and four.

V SUMMARY

If one wanted to build a fusion-fission reactor that would do the most (in terms of power and fissile fuel production) in the most efficient manner, the idea of breeding tritium in fission reactors, presented by Fortescue, is attractive. By removing the tritium breeding requirement from a hybrid breeder, the initial fissile loading of the blanket could be reduced, because the poisonous effects of Li-6 are not present. Hybrid breeders may ultimately operate in this manner, however for preliminary calculations, it was considered to be a self-sufficient reactor. With this self-sufficiency in mind, the parametric phase of the hybrid breeder calculations has pointed out a number of the relationships between tritium breeding and the design variables. No gain in lithium atom density is obtained by using solid lithium metal rather than Li_2O . The oxygen does not hinder blanket performance, but if lithium were present as a lead or aluminum compound, parasitic absorption would probably be noticeably detrimental. The best coupling of the tritium breeding zone to the neutron multiplying zone, without sacrificing good coupling to the other zones, seems to be attained when the lithium is behind the other zones. Coupling also seems better when the plutonium zone is wider and less concentrated. In regions of high fission neutron spectra, Li-6 makes better use of the neutrons, however the Li-7 tritium breeding reaction conserves the neutron (but not at a high enough energy to be useful to another Li-7 atom), and the proportion of Li-6 to Li-7 for a maximum TBR is not a clearcut choice. The optimum proportion of Li-6 in the lithium compound can not be determined for a general case; it will depend on all of the other characteristics of the particular blanket

under study. A maximization calculation should be done to determine this optimum value. Finally, the calculations show that for the fast neutron spectrum averaged cross sections employed, the SS-316 clad is superior to TZM.

The progression to the central composite of reference case three came about as more of these relationships became evident. Before proceeding to the more detailed calculations with the design from this reference case, one more central composite design set should be run. The power in reference case three may be too large to handle efficiently, and decreasing the amount of plutonium, while changing some zone widths and perhaps increasing the Li-6 atom density in Li_2O , could bring the power down without decreasing the TBR or fissile production.

Final comments concerning the hybrid breeder designs studied in this paper are again related to the question of realism. Section IV mentioned that the solid lithium zones used in the calculations could not exist as modeled. Some consideration must be made for cooling and structural support. The Li_2O design must also be modified to prevent chemical reactions from occurring between the oxide and the carbides used in the multiplier and fissile breeding regions. A plate welded between zones of different composition can separate them, but helium is produced in the tritium breeding reactions, and the lithium can not be completely isolated from the coolant flow unless unvented fuel pins are used. The method of cooling, used for the designs in this paper, may also have to be

modified. Most of the power generated in the blanket comes from the neutron multiplier region. Since this zone is only six to ten centimeters wide, the coolant flow, for the lattice of Figure I-2, may have to be very large to cool this zone, but not too large to cool the other zones. A mixture of different lattices may give fast enough coolant flow in high power regions and slower flow in low power regions. Some preliminary ideas for the blanket layout and coolant considerations appear in Ref. #3 and its preceeding quarterly reports. Lastly, this paper mentions very little about the economic aspects of the reactor. Any decision to build commercial hybrids will rest completely on a cost versus return analysis of a hybrid breeder reactor.

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