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BLANKET DESIGN STUDY FOR A COMMERCIAL TOKAMAK HYBRID REACTOR (CTHR)

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BLANKET DESIGN STUDY FOR A
COMMERCIAL TOKAMAK HYBRID REACTOR (CTHR)

by

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ABSTRACT

In this report the results are presented of a study on two blanket design concepts for application in a Commercial Tokamak Hybrid Reactor (CTHR). Both blankets operate on the U-Pu cycle and are designed to achieve tritium self-sufficiency while maximizing the fissile fuel production within thermal and mechanical design constraints. The two blanket concepts that were evaluated were: 1) a UC fueled, stainless steel clad and structure, helium cooled blanket; and 2) a UO_2 fueled, zircaloy clad, stainless steel structure, boiling water cooled blanket. Two different tritium breeding media, Li_2O and LiH, were evaluated for use in both blanket concepts. The use of lead as a neutron multiplier or reflector and graphite as a reflector was also considered for both blankets.

The neutronic performance was calculated for a sufficient irradiation exposure time to achieve about 3% fissile fuel enrichment in the blankets. It was found that the UC, helium cooled blanket can produce up to 75% more fissile fuel than the UO_2 , boiling water cooled blanket, due to both the increased uranium density with the UC compared to the UO_2 and also because of the harder neutron spectrum with the helium compared to the boiling water. An important aspect of the analysis was to investigate the neutronic benefits of full fertile blanket wall coverage in the CTHR, as opposed to restricting the blanket on the inward portion of the torus to lithium bearing regions only. The results of the comparison of full (inner and outer) fertile blanket coverage versus outer fertile blanket coverage only indicated that increases in fissile fuel production of up to about 20% would be possible with the full blanket coverage. However, if the reactor plant availability decreases due to increased reactor downtime to service and refuel the inner blanket, the neutronic benefit of using full blanket coverage could be negated.

Other conclusions from the neutronic analyses indicated certain advantages in the use of LiH as compared to Li_2O for the tritium breeding material. A blanket using LiH can be thinner than one using Li_2O due to the excellent moderating properties of the hydrogen. Also, the same moderating effect with the LiH

decreases parasitic absorption in the structure in the lithium material zone compared to the use of Li_2O . An investigation of the use of lead in the blanket indicated that significant benefits could result by using it both as a reflector behind the fertile zone and as a neutron multiplier in the inner region for the case of outer fertile blanket coverage only.

For each of the blankets a conceptual mechanical design was developed and the thermal-hydraulic performance was evaluated. High pressure coolants were used in the blankets to obtain high power conversion efficiency and low pumping power requirement. The steam pressure at the turbine throttle valve for the UO_2 blanket is about 6.20 MPa (900 psia) and that for the UC blanket is about 8.62 MPa (1250 psia). With these pressures the present day nuclear turbine technology for economical commercial applications can be utilized.

These results represent an initial assessment, and further iterations between neutronic, thermal-hydraulic, structural, and mechanical design are required to arrive at an optimum design configuration. However, overall both of these blanket concepts appear feasible and attractive for use in the CTHR, with the UC fueled blanket giving improved performance relative to the UO_2 fueled concept but at the expense of employing more advanced materials and technology with a less well developed data base.

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1.0 INTRODUCTION

The fusion-fission hybrid reactor concept represents a potentially attractive method for producing fissile fuel from the fertile nuclides ^{238}U or ^{232}Th . The principal advantage of this system is based on the combination of the 14.1 MeV D-T fusion neutron energy and the neutron multiplying properties of the fertile materials. A number of preliminary conceptual designs of hybrid reactors have been developed to assess the performance and economics of both demonstration type and commercial type reactors. An example is a study which has recently been completed^(1,2) of a demonstration hybrid device based on a tokamak fusion driver, the Demonstration Tokamak Hybrid Reactor (DTHR). The DTHR was designed to operate under relaxed conditions with the main objective being to produce a significant amount of fissile fuel while demonstrating the feasibility of the tokamak hybrid reactor concept. The limited operating goals of the DTHR, such as not requiring tritium self-sufficiency and optimum blanket wall coverage, allowed considerable relaxation of the blanket design. Additional considerations such as low cost and a well developed technology base plus the possibility of operation on a proliferation resistant fuel cycle led⁽³⁾ to the choice of a water cooled, zircaloy clad, ThO_2 fueled blanket concept for the DTHR. Both liquid water cooled^(1,3) as well as two phase (boiling) water cooled^(2,4) concepts were studied for the DTHR.

In this report an evaluation has been made of the performance of two blanket concepts for use in a commercial device, the Commercial Tokamak Hybrid Reactor (CTHR). The CTHR differs from the DTHR in several features, such as increased plasma duty cycle, plant availability, and blanket wall coverage. In addition, tritium self-sufficiency was set as a requirement for the CTHR. In order to increase the power and fissile fuel production, operation on the $^{238}\text{U}/^{239}\text{Pu}$ fuel cycle was selected for the CTHR.

The two blanket concepts that were evaluated were: 1) a UC fueled, stainless steel clad and structure, helium cooled blanket (UC-SS-He); and 2) a UO_2 fueled, zircaloy clad, stainless steel structure, boiling water cooled blanket (UO_2 -Zr-BW). The neutronic performance of each of these concepts was

calculated for a sufficient irradiation exposure time to achieve 3 percent fissile fuel enrichment in the blanket. A representative mechanical design along with the thermal-hydraulic performance was also evaluated for both blankets. The motivation for investigating the UO_2 -Zr-BW blanket is that it is similar to the concept developed for the DTHR (with a change of fuel from ThO_2 to UO_2) and it utilizes many features of proven light water fission reactor technology. The UC-SS-He blanket concept offers the potential for improved neutronic performance relative to the UO_2 concept, but at the expense of employing more advanced, less well-developed technology and materials.

For the tritium breeding medium, solid lithium bearing compounds clad in stainless steel and helium cooled were selected. Due to their high lithium atom densities, the compounds used for the analysis were Li_2O and LiH . The different neutronic behavior of these two materials was investigated and compared for both the UO_2 and UC fertile fueled blanket. In addition, the use of a lead (Pb) zone as a neutron multiplier and/or reflector or a graphite zone as a reflector was studied for both of the blanket concepts.

A very important part of the analysis reported herein was to investigate the neutronic benefits of full blanket wall coverage in a tokamak fusion driver hybrid device. Due to accessibility and maintenance considerations in a tokamak, it may be desirable to restrict the blanket on the inward portion of the torus to tritium breeding regions only and use fertile breeding regions only on the outer portions. The neutronic advantage of full (inner and outer) fertile blanket coverage versus the outer fertile blanket coverage only has been investigated and compared for both the UO_2 -Zr-BW and UC-SS-He blanket concepts with both the Li_2O and LiH tritium breeding media options. During this analysis the various zone thicknesses in both the inner and outer blankets were determined such that each configuration was self-sufficient in breeding tritium.

In section 2.0 of this report the calculational methods and assumptions used in the neutronic analyses are outlined. Considerations of the poloidal distribution of the neutron first wall loading and the inner and outer blanket coverage fractions are presented in section 3.0. The results of the neutronic

performance of the UO_2 blanket and the UC blanket are presented in sections 4.0 and 5.0, respectively. A conceptual mechanical design configuration for each of the blankets is discussed in section 6.0, and the thermal-hydraulic design and analysis results are given in section 7.0. A summary and conclusions from the work are presented in section 8.0.

2.0 NEUTRONIC CALCULATIONAL PROCEDURE AND ASSUMPTIONS

The one-dimensional cylindrical geometry model used to determine the nuclear performance of the various blankets is shown schematically in Figure 2-1. This type of representation, with the cylinder axis along the centerline of the torus, was selected as the most appropriate for the assumed tokamak hybrid design in which the compositions of the inner and outer blanket regions may be different. For example, due to access and maintenance limitations, the inner blanket may be used for tritium breeding with the use of fertile materials limited to the outer blanket only. An important aspect of this study was to assess the neutronic advantage of using either fertile material or a lead neutron multiplier in the inner blanket relative to using only lithium bearing compounds in that region.

As illustrated in Figure 2-1, the 14 MeV D-T fusion neutron source is assumed to be uniformly distributed over the plasma region. The plasma is surrounded by a void (scrape-off) layer and then by the vacuum vessel/first wall. For these calculations the tokamak major radius R_0 was 6.0 m, the plasma minor radius was 1.4 m, the scrape-off layer thickness was 0.2 m, and the first wall was an equivalent 0.5 cm thick layer of stainless steel (SS-316). Various assembly gaps were included in the analysis, and a shield of SS-316 and borated water was used behind the inner and outer blanket zones.

Two blanket concepts were investigated for this study: a UO_2 fueled, zircaloy clad, SS-316 structure, boiling water (BW) cooled concept (UO_2 -Zr-BW) and a UC fueled, SS-316 clad and structure, helium cooled concept (UC-SS-He). Two different tritium breeding media were also considered, Li_2O and LiH. The volume fractions and material compositions of the fertile and lithium zones assumed in the neutronics calculations are shown in Table 2-1. In performing the nuclear analyses, both the clad and structure in the UO_2 fueled concept were treated as being composed of zircaloy since the division between these two materials was not established at the time of the nuclear calculations. The ratio of fertile fuel volume to clad plus structure volume in the blankets was varied about the reference value of $0.5/0.2 = 2.5$ shown in the table to

NOTE: All Dimensions in cm

* BORATED WATER (35%) AND STAINLESS STEEL

** VACUUM VESSEL WALL, DENSITY FACTOR 0.5

Figure 2-1. Schematic Diagram of the One-Dimensional Cylindrical Geometry Used in the Neutronics Analysis.

TABLE 2-1

COMPOSITION AND VOLUME FRACTIONS USED
FOR THE NEUTRONICS ANALYSIS OF THE TWO BLANKET CONCEPTS

		<u>VOLUME FRACTION</u>	<u>UO₂ BLANKET</u>	<u>UC BLANKET</u>
FERTILE ZONE	FERTILE FUEL	0.50	UO ₂	UC
	CLAD, STRUCTURE	0.20	Zirc-4	SS-316
	COOLANT	0.30	Boiling Water	Helium
LITHIUM ZONE	FUSILE FUEL	0.70	Li ₂ O or LiH	
	CLAD, STRUCTURE	0.10	SS-316	
	COOLANT	0.20	He	

study the advantages of increasing this ratio. The mass and atomic densities of the various fertile and fusile fuels used in the analysis are shown in Table 2-2. The UC fuel has about a 30% larger U atom density relative to the UO_2 fuel. For all cases the uranium was assumed to be depleted (tails) with 0.25% ^{235}U and the lithium was natural isotopic composition.

The zone thicknesses in the blanket were varied for all cases to achieve a tritium breeding ratio of $\sim 1.05 - 1.10$ at the beginning of life (BOL). The general configuration used for most of the analyses consisted of a fertile material zone followed by a lead zone followed by the lithium zone. Variations of the configuration for use in the inner blanket region consisted of a lead layer followed by the lithium zone, and a lithium zone only. The volume fractions assumed for the lead zone were the same as those for the lithium zone, i.e., 70% Pb, 10% SS-316, and 20% He. When the lead zone was directly facing the plasma, it served as both a neutron multiplier and reflector. However, when the lead was behind the fertile zone its main function was as a neutron reflector.

The neutron transport calculations were performed using the one-dimensional discrete-ordinate code ANISN⁽⁵⁾ with an S_8 treatment of the angular quadrature and a P_3 approximation to the anisotropic scattering. The burn-up and depletion were analyzed using a combination of the CINDER and HIC codes which were appended⁽⁶⁾ to the ANISN code. For each time step in the calculation, ANISN supplies collapsed five group fluxes and cross sections to the CINDER-HIC module. The burn-up and isotope depletion are then calculated using a library of five group cross sections and decay constants, and new isotope number densities are returned to ANISN for the next time step. In this way, the program cycles through the various time steps requested, computing the isotope number densities at each step. Using a library containing the fission product yield and cross section data, the code also calculates a lumped fission product concentration and cross section which is then used in the ANISN calculation. A number of isotopes and transmutation and decay chains included in the computation which are of interest in the U-Pu cycle, are shown in Figure 2-2.

TABLE 2-2

DENSITIES USED FOR VARIOUS MATERIALS IN THE
NEUTRONICS CALCULATIONS (AT A VOLUME FRACTION OF 1.0)

	<u>UO₂</u>	<u>UC</u>	<u>Li₂O</u>	<u>LiH</u>
MASS DENSITY, gms/cm ³	10.4	12.6	2.0	0.8
U DENSITY ¹ , 10 ⁻²² atoms/cm ³	2.32	3.02	--	--
C DENSITY, 10 ⁻²² atoms/cm ³	--	3.02	--	--
O DENSITY, 10 ⁻²² atoms/cm ³	4.64	--	4.05	--
Li DENSITY ² , 10 ⁻²² atoms/cm ³	--	--	8.10	6.21
H DENSITY, 10 ⁻²² atoms/cm ³	--	--	--	6.21

¹Depleted U, 0.25% ²³⁵U

²Natural Li, 7.5% ⁶Li

Figure 2-2. Isotopes, Half-Lives and Decay Chains of Interest in U-Pu Fuel Cycle.

A 27 neutron group cross section set was used for the ANISN transport calculations. It was prepared by collapsing the 171 group CTR processed multi-group cross section library⁽⁷⁾ over an appropriate flux spectrum. Resonance effects in the U and Pu isotopes were accounted for by using the self-shielding capabilities of the library. Kerma factors for the nuclear heating calculations and displacement and gas production cross sections for the radiation damage analysis were taken from the MACKLIB-IV data library⁽⁸⁾.

The most important tokamak device parameters with regard to the nuclear analysis of the blanket performance were assumed to have values as shown in Table 2-3. Although the neutron wall load during the pulse was 2.0 MW m^{-2} , the plant factor averaged wall load was 1.2 MW m^{-2} due to scaling by the product of the plasma duty cycle and plant availability (0.6). This value of 1.2 MW m^{-2} was then used to calculate the neutron source strength for input to the ANISN-CINDER-HIC calculations.

Several key parameters will be used to describe the blanket performance, as defined below:

$$F = \text{net no. of } {}^{239}\text{Pu} \text{ atoms produced per fusion neutron}$$

$$= \frac{\text{no. } {}^{238}\text{U} (n, \gamma) \text{ reactions} - \text{no. } {}^{239}\text{Pu} \text{ absorption reactions}}{\text{fusion neutron}}$$

$$T = \text{tritium breeding ratio}$$

$$= \frac{\text{no. } {}^6\text{Li} (n, \alpha) \text{T} + {}^7\text{Li} (n, n'\alpha) \text{T} \text{ reactions}}{\text{fusion neutron}}$$

$$M = \text{blanket energy multiplication factor}$$

$$= \frac{14.1 + 200 \times (\text{no. fission reactions/fusion neutron})}{14.1}$$

These blanket "per fusion neutron" parameters are then used to calculate macroscopic quantities such as fissile fuel production per year and thermal power produced for a tokamak with performance as shown in Table 2-3. One other important device parameter which scales the per fusion neutron values is the blanket wall coverage factor, which is discussed in the following section.

TABLE 2-3

IMPORTANT TOKAMAK DEVICE PARAMETERS
ASSUMED FOR THE NEUTRONICS ANALYSIS

MAJOR RADIUS	6.0 m
MINOR RADIUS	1.4 m
PLASMA ELONGATION	1.6
SCRAPE-OFF THICKNESS	0.2 m
NEUTRON WALL LOADING (AT PULSE)	2.0 MW m ⁻²
PLASMA D-T POWER	1200 MW
PLASMA DUTY CYCLE (DC)	90%
PLANT AVAILABILITY (PA)	70%
PLANT FACTOR (DC x PA) AVERAGED NEUTRON WALL LOADING	1.2 MW m ⁻²

3.0 WALL LOAD DISTRIBUTION AND BLANKET COVERAGE

An important factor with regard to tokamak hybrid reactor design is the blanket wall coverage fraction and the percentage of the D-T fusion neutrons intercepted by the fertile blanket. In this section the poloidal variation of the neutron wall load about the first wall and the fraction of neutrons intercepted by the outer and inner blanket regions is investigated for a Commercial Tokamak Hybrid Reactor (CTHR) design. These results are then combined with estimated losses due to penetrations, neutral beam injectors, and a bundle divertor to estimate potential blanket coverage factors for the CTHR. The per fusion neutron values determined from the neutronics calculations are then scaled by these coverage factors to estimate total fissile and power production.

The poloidal variation of the neutron wall load about the first wall of the CTHR is shown in Figure 3-1. Here the "nominal" wall load is the total D-T neutron power divided by the first wall area, and the peaking factor is the ratio of the wall load at a particular point of the wall to the nominal value. For the CTHR the nominal value during the pulse is 2.0 MW m^{-2} .

The wall load distribution has been determined from a detailed, three-dimensional calculation⁽⁹⁾ of the distribution of the uncollided 14 MeV neutron angular and scalar flux and current around the wall of a tokamak fusion reactor. The effect of the toroidal geometry is implicitly included in the analysis by numerically solving the integral form of the neutron transport equation for streaming of neutrons from the plasma to the wall. For the analysis for the CTHR, the major device parameters assumed were: major radius of 6.0 m, minor radius of 1.4 m, scrape-off layer thickness of 0.2 m, elongation factor of 1.6, and elliptical plasma and first wall shape. A realistic, nonuniform distribution of the fusion neutron source⁽⁹⁾ was modeled for the calculation by a set of nested ellipses shifted outward from the nominal plasma center. This is a reasonably good model of the flux surface contours of the plasma along which the neutron source is a constant. The location of the maximum source strength was determined⁽⁹⁾ by the torus aspect ratio and the poloidal plasma beta, and was estimated to be about half-way out along the plasma minor radius, i.e., at the point 0.7 m from the major radius R_0 .

Figure 3-1. Distribution of Fusion Neutron Wall Loading About the First Wall of the CTHR.

As shown in Figure 3-1, the wall load is $\sim 52\%$ above nominal at the outermost wall point and is most heavily distributed along the upper approximately two thirds of the outer wall. At the top point on the wall it is $\sim 19\%$ below nominal and on the innermost point it is $\sim 14\%$ below nominal. From this analysis it becomes clear that there is a strong incentive to utilize as fully as possible the areas behind the outer first wall for fertile blanket modules. By integrating the wall load distribution in Figure 3-1 over the wall area, the fraction of the fusion neutrons that pass through (intercept) various segments on the first wall can be calculated.

This was done to produce the values shown in Figure 3-2, where the fraction of the source neutrons intercepted by various segments of the blanket is illustrated. It can be seen that $\sim 70\%$ of the fusion neutrons strike the outer blanket regions, while $\sim 28\%$ penetrate toward the inner blanket and $\sim 2\%$ are lost through gaps in the blanket placement above and below the vacuum vessel. For the one-dimensional ANISN calculation, these outer and inner blankets are approximated by vertical segments as shown in Figure 2-1. Hence, to correctly normalize the ANISN results, the poloidal coverage factors shown in Figure 3-2 must be consistent with the source fractions directed towards the inner and outer blankets for the ANISN geometry of Figure 2-1. It turned out that small adjustments in the ANISN values were necessary to calculate macroscopic fuel and power production quantities consistent with the coverage factors in Figure 3-2.

Besides these considerations of the poloidal wall coverage factors, there are blanket coverage losses in the toroidal direction due to penetrations for neutral beam injectors, vacuum pumps, and a bundle divertor. For the CTHR it has been conservatively estimated that $\sim 15\%$ of the area on the outer periphery is lost due to these penetrations, so that the effective toroidal wall coverage factor is 0.85.

Thus, for the full blanket coverage with fertile fueled zones on both the inner and outer regions, the maximum wall coverage or fraction of neutrons intercepted by the fertile blanket is 0.28 (inner blanket) plus 0.60 (outer

Figure 3-2. Schematic Diagram Showing the Fraction of Fusion Neutrons Intercepted by Various Wall Segments

blanket, 0.70×0.85), which equals 0.88. Hence, 12% of the fusion neutrons are "non-productive". For fertile fuel in the outer blanket only, the fraction of fusion neutrons intercepting the fertile zones is $0.70 \times 0.85 = 0.60$. In this case, 28% of the fusion neutrons are still "useful" in the sense that they intercept the inner blanket for tritium breeding. These values are then used to appropriately scale the ANISN results from the one-dimensional geometry approximation.

4.0 UO_2 FUELED BLANKET NUCLEAR ANALYSIS

4.1 INTRODUCTION

UO_2 was selected as a candidate fertile fuel for the CTHR blanket because of the extensive fabrication and irradiation experience with this fuel form from light water reactor programs. It was also expected that, except for the cyclic nature of tokamak operation, irradiation conditions (power density and burnup) would not differ greatly from those in a pressurized water reactor (300 w/cm^3 average in the fuel and less than 30,000 MWD/T burnup). Zirconium was selected as the accompanying clad material, with additional structure to be stainless steel. Boiling water was established as the fertile fuel coolant, with first iteration average steam qualities of 80% and 30% to be considered. This coolant selection and its properties were based upon earlier DTHR results.^(2,4) It was expected that some second iteration studies on the coolant would possibly have to be performed based upon the results of the nuclear heating and thermal-hydraulic calculations from the first iteration work.

Tritium breeding material selection was restricted to solid lithium compounds. These compounds have the advantage of well known chemical-physical properties, and impose no MHD effects as compared to flowing liquid lithium. Solid lithium compounds also present the possibility of in-situ, on line separation of the tritium produced.

From an earlier study of candidate solid lithium compounds, lithium hydride (LiH) and lithium oxide (Li_2O) were found to be the best performers from the point of view of total product (tritium plus fissile fuel) production. The other two compounds considered were lithium aluminate (LiAlO_2) and lithium lead (Li_7Pb_2). While the latter has been shown to have some advantages in pure fusion systems due to the multiplying effects of the lead, this advantage is lost in hybrid systems because of the moderation of the 14 MeV source neutrons by the fertile fuel. LiAlO_2 suffers from a low lithium density (a factor of nearly four compared to Li_2O). Nonetheless, its comparative chemical

inertness could later prove to be an advantage and LiAlO_2 may have to be reconsidered. For the studies here, only LiH and Li_2O were examined.

All structure in the lithium zone was assumed to be stainless steel, with helium as the primary coolant candidate. This choice was made to avoid possible lithium-water reactions in the event of clad failure, and because it offers the possibility of on-line tritium recovery.

All structure in the graphite reflector zones was also assumed to consist of stainless steel with helium coolant.

In establishing material volume fractions for these first iteration studies, values were selected to be reasonably conservative with regard to probable thermal-hydraulic, structural, and mechanical requirements. The volume fractions thus established are shown in Table 4-1.

TABLE 4-1

INITIAL CTHR BLANKET VOLUME FRACTIONS

ZONE	MATERIAL	VOLUME FRACTION (PER CENT)	
		COOLANT	CLAD (INCLUDING STRUCTURE)
Fuel	50	30	20
Lithium	70	20	10
Graphite	70	20	10
Lead	70	20	10

Since the division between clad (zirconium) and structure (stainless steel) in the fuel zone was not initially known, it was assumed in the neutronics calculations for simplicity to be entirely zirconium. The mechanical design analysis in section 6.1 indicates what the division between stainless steel and zirconium would be in such a blanket, and neutronics calculations using this composition will be performed in a second iteration of the blanket design

study. It is to be expected that incorporation of stainless steel structure would exact some small penalty in product production rate because of the higher parasitic capture of stainless steel compared to zirconium.

One important purpose of the present study was to evaluate the economic consequences of including or not including fertile material in the inner blanket region. Therefore, calculations with the two candidate lithium compounds were included for both these cases. In order to provide a valid comparison of results, and to insure tritium self sufficiency, a tritium breeding ratio between 1.05 and 1.1 at beginning of life was required. Arrangements of fertile/fusile/reflector zones were varied for each case until this condition was satisfied. No further attempts were made to optimize the system (e.g., with respect to total fuel production or burnup characteristics) beyond this constraint. Such optimization studies may be fruitful once a reference system is established.

4.2 Li_2O TRITIUM BREEDING MATERIAL

4.2.1 FERTILE MATERIAL IN BOTH INNER AND OUTER BLANKETS

The first studies were conducted with a constant 80% quality steam (water density 0.032 grams/cm^3) in the fertile zones. The arrangement that produced a tritium breeding ratio near 1.1 at beginning-of-life (BOL) is shown in Figure 4-1. Table 4-2 lists important parameters characterizing the blanket neutronics behavior for a 2 year burnup calculation. The "per fusion neutron" values F, T, and M are defined in section 2.0. Other important parameters shown in Table 4-2 are: G--the net instantaneous ^{239}Pu production rate, kg/yr; C--the cumulative ^{239}Pu in the blanket, kg; E_n --the average total fissile enrichment, percent; H--the blanket heavy metal (HM) inventory, tonnes; and P_{Th} --the duty cycle averaged blanket thermal power, MWt. Also given in Table 4-2 is a sometimes used figure-of-merit (FM) for the blanket performance, defined as the ratio of the fissile production rate to the plant thermal power in units of kg fissile/MWt-yr. To calculate FM for these blankets, the thermal power value in Table 4-2, which is pulse duty cycle averaged, has been scaled by the plant availability factor, which is 0.7 for the CTHR as shown in Table 2-3. This is consistent with the fissile

NOTE: All dimensions in cm

Figure 4-1. Neutron Transport Calculation Configuration For The Li_2O , UO_2 Full Fertile Blanket

TABLE 4-2

NEUTRONIC PERFORMANCE FOR THE $\text{UO}_2\text{-Li}_2\text{O}$, FULL BLANKET,
80% CONSTANT STEAM QUALITY IN FERTILE ZONE

		<u>BOL</u>	<u>1 YEAR</u>	<u>2 YEARS</u>
Integrated Wall Load,	$\frac{\text{MW-Yrs}}{\text{m}^2}$	0	1.2	2.4
T,	$\frac{\text{Tritium Atoms}}{\text{Fusion Neutron}}$	1.06 (1.09)	1.10 (1.13)	1.13 (1.17)
F,	$\frac{\text{Net } ^{239}\text{Pu atoms}}{\text{Fusion Neutron}}$	0.81 (0.79)	0.73 (0.72)	0.66 (0.65)
T + F		1.87 (1.88)	1.83 (1.85)	1.80 (1.82)
G,	$\frac{\text{kg } ^{239}\text{Pu}}{\text{yr}}$	2280	2070	1870
C,	Cumulative ^{239}Pu (kg)	0	2170	4140
E_n	enrichment, pct.	0.25	1.02	1.73
H,	heavy metal inventory (THM)	--	274	273
M,	energy multiplication	5.1	5.8	6.5
P_{Th}	thermal power, MWt	3960	4490	5020
FM,	$\frac{\text{kg}}{\text{MWt-yr}}$	0.82	0.66	0.53

*Parentheses are for 80% quality steam cooling in the Li_2O zone as well as the fertile zone

production rate (G) definition, which has already been scaled by this value. The time dependence of the parameters is typical of all of the blankets studied. That is, the tritium breeding ratio increases due to the increase in the fission source, while the net ^{239}Pu production rate decreases as a result of the burnup of the ^{239}Pu produced. The sum of the two (T&F) slowly decreases with the time (burnup) primarily as a result of the increase in parasitic absorptions.

The importance of a graphite reflector zone for tritium production in Li_2O systems is illustrated in Figure 4-2. It can be seen that the tritium production rate peaks near the graphite, as a result of thermalization and reflection of high energy neutrons leaking from the lithium zone. An examination of the neutron flux leaking into the graphite zones indicates a significant increase (up to 8%) in tritium production may be obtained by increasing the lithium zone thicknesses. This would, of course, increase the total product rate.

The effect of cooling the lithium zone with the 80% quality water mixture was also examined for this system, and the results are shown in parentheses in Table 4-2. The total product rate is seen to increase slightly ($\sqrt{1\%}$), as a result of an increase of 3% in the tritium production rate and a 2% decrease in fissile production. These are due primarily to a decrease in leakage from the lithium zones. The effect stays nearly constant with burnup. Therefore, it appears there is no deterioration in system neutronic performance in steam cooling the lithium zone. This conclusion is also valid for LiH systems because of the high number density of hydrogen already present.

Li_2O full fertile blanket performance with a constant 30% (0.055 g/cm^3) quality steam in fertile zone was next examined, and the results are listed in Table 4-3 for a four (4) year burnup calculation. The zone configuration was the same as that above and shown in Figure 4-1. It can be seen that the change in system performance is very small over the two years comparison period, especially if the results are normalized to the same tritium production rate (preserving the total product rate). We conclude that neutronic performance is not significantly degraded for steam quality changes from 80% to 30%.

Figure 4-2. Spatial Distribution of the Tritium Production Rate in the UO_2 - Li_2O Blanket With Constant 80% Quality Steam Coolant.

TABLE 4-3

NEUTRONIC PERFORMANCE FOR THE $\text{UO}_2\text{-Li}_2\text{O}$, FULL FERTILE BLANKET
30% CONSTANT QUALITY STEAM IN FERTILE ZONE

	<u>BOL</u>	<u>1 YEAR</u>	<u>2 YEARS</u>	<u>3 YEARS</u>	<u>4 YEARS</u>
T	1.03	1.08	1.12	1.16	1.18
F	0.84	0.75	0.67	0.61	0.55
T + F	1.87	1.83	1.79	1.77	1.73
G	2350	2100	1900	1720	1550
C	0	2230	4230	6040	7680
E_n	0.25	1.03	1.75	2.41	3.02
H	--	276	274	272	271
M	5.1	6.0	6.7	7.4	8.0
P_{Th}	3940	4610	5190	5700	6150
FM	0.85	0.65	0.52	0.43	0.36

The time variation of tritium and fissile (^{239}Pu) production rates is shown in Figure 4-3. Also shown is the buildup in the average fissile enrichment, which is seen to reach just over 3% in four (4) years. The maximum enrichment at this time is about 3.4% in the regions adjacent to the plasma.

Coupled neutron-gamma transport calculations were run at beginning-of-life (BOL) to obtain the gamma-ray and neutron flux distributions. The corresponding end-of-life (EOL) fluxes were obtained in an approximate way by using heavy isotope number densities taken from the four (4) year burnup for the neutron only transport problem. Fission products were not included, so that the calculated fluxes are slightly high. However, the resultant error in the heating rates is expected to be very small. The fluxes were thus used, along with KERMA factors obtained from the MACKLIB-IV library⁽⁸⁾, to obtain heating rate profiles through the entire blanket. These are shown in Figure 4-4.

It can be seen that, as expected, the largest heating rates occur in the fertile zones with average values of 51 and 69 w/cm^3 at BOL in the inner and outer blankets, respectively. The corresponding peak to average values are 2.2 and 1.9. The heating rate profile at EOL (four years) is also shown in Figure 4-4. The fertile zones average heating rates have been increased to 98 and 120 watt/cm^3 for the inner and outer blankets. However, peak to average power has decreased to 1.8 and 1.5 respectively, in the two zones.

It is also of interest to note from Figure 4-4 that significant power peaking occurs in the Li_2O zones adjacent to the graphite reflectors. This is a consequence of thermalization in the graphite followed by capture in ^6Li . It was shown in Ref. 10 that the calculational model employed underpredicts thermal flux. Therefore, the power peaking is likely to be larger than that observed, although still below the average value (about 5 W/cm^3 at BOL) in these zones.

An important consideration in the design of the blanket/shield is the neutron fluence outside the shield. Because of the insulation in the TF coils, the shield must limit this fluence to less than $10^{18} \text{ n}/\text{cm}^2$ over the lifetime of the TF coils.

Figure 4-3. Time Dependent Variation of the Tritium and Fissile Production Rates and the Fissile Enrichment for the $\text{UO}_2\text{-Li}_2\text{O}$ Full Fertile Blanket With Constant 30% Quality Steam Coolant.

Figure 4-4. Heating Rate Spatial Profiles in the $\text{UO}_2\text{-Li}_2\text{O}$ Full Fertile Blanket With Constant 30% Quality Steam Coolant.

The first line in Table 4-4 shows the results calculated at BOL for the configuration being considered, as well as for those studied below. The change with lifetime is negligible. It can be seen that the thinner inner blanket/shield is most limiting but still well within the maximum allowable. This margin is necessary, since one-dimensional analyses generally underpredict fluence. Regions near shield penetrations would also require more detailed analyses.

TABLE 4-4
FLUENCES AT BLANKET/SHIELD BOUNDARIES*

SYSTEM	LEFT BOUNDARY FLUENCE (n/cm ²)	RIGHT BOUNDARY FLUENCE (n/cm ²)
Li ₂ O - Full Blanket	2.39 x 10 ¹⁷	1.72 x 10 ¹⁵
Li ₂ O - Outer Only	3.23 x 10 ¹⁷	1.69 x 10 ¹⁵
LiH - Full Blanket	2.06 x 10 ¹⁷	1.07 x 10 ¹⁵
LiH - Outer Only	2.61 x 10 ¹⁷	2.17 x 10 ¹⁵

*Over all neutron energies, 30 year lifetime

4.2.2 FERTILE MATERIAL IN THE OUTER BLANKET ONLY

The configuration for Li₂O fusile material with fertile material in the outer blanket only is shown in Figure 4-5, and the results for a two-year burnup are listed in Table 4-5. This table may be compared with the full fertile blanket case of Table 4-2. However, no attempt was made to optimize the various zone dimensions with respect to total product production, and some caution must be exercised in making this comparison. In particular, an increase in tritium production up to 12% may be realized by increasing the thicknesses of the Li₂O zones. In order to decrease the excess tritium then being produced, the fertile zones would be enlarged, increasing fissile production. The values in parentheses were obtained in an approximate way by assuming a realistic 6% increase in tritium production.

Figure 4-5. Neutron Transport Calculation Configuration for the Li_2O , UO_2 Outer Fertile Blanket Only

TABLE 4-5

NEUTRONIC PERFORMANCE FOR THE $\text{UO}_2\text{-Li}_2\text{O}$, OUTER FERTILE BLANKET ONLY,
80% STEAM QUALITY IN FERTILE ZONE

	<u>BOL</u>	<u>1. YEAR</u>	<u>2. YEARS</u>
T	1.13	1.16	1.18
F	0.528 (0.596)	0.480 (0.550)	0.439 (0.510)
T + F	1.66 (1.73)	1.64 (1.71)	1.62 (1.69)
G	1405 (1585)	1264 (1450)	1159 (1346)
C	0	1334 (1495)	2546 (2890)
E_n	0.25	0.922	1.543
H	--	193	191
P_{Th}	3072	3357	3642
FM	0.65	0.54	0.45

*Values in parentheses are estimates for enlarged Li_2O zones.

4.3 LiH TRITIUM BREEDING MATERIAL

4.3.1 FERTILE MATERIAL IN BOTH INNER AND OUTER BLANKETS

The configuration for this case is shown in Figure 4-6, and the results for a two-year burnup at 80% steam quality are listed in Table 4-6. The total product rate of 2.17 at BOL is the highest for any of the cases studied, and is a consequence of both low leakage into the shield and lower parasitic absorptions in the stainless steel of the lithium containing zones. The latter is a result of the ^6Li being better able to compete for neutron absorptions with the stainless steel in the softer spectrum provided by the slowing down in the hydrogen of the LiH. This is shown in Table 4-7, which lists the neutron balance for all of the configurations studied. It was also unnecessary for this reason to provide a graphite reflector. Because of the effectiveness of LiH as a shield material, large zones were retained, as shown in Figure 4-6 with the resultant boundary fluences listed in Table 4-4. However, as displayed in Figure 4-7, only about the first 15 cm of the LiH zones are effective for tritium production. This can be compared to Figure 4-2 for the Li_2O blanket with the graphite reflector.

It can be seen from a comparison of Figures 4-1 and 4-6 that the fertile zones are much larger in the LiH system, resulting in a 60% increase in the UO_2 inventory (from 274 to 445 tonnes). As a result the enrichment is lower, 1.47% at the end of two years compared to 1.73%, despite the higher ^{239}Pu production rate. In order to reduce this inventory, lead reflector zones were added behind the fertile zones, as shown in Figure 4-8. Again, no attempt was made to optimize the arrangement, requiring only that the tritium production at BOL be between 1.05 and 1.10. The results for a two-year burnup study are listed in Table 4-8.

It can be seen that the 40% reduction in fertile inventory has resulted in only a 4% decrease in product production. There is a 10% decrease in fissile production at BOL, but this is partially a result of increased tritium production, so that some optimization is possible. The enrichment at the end of two years has increased 21% to 1.81. The use of lead to decrease UO_2 inventory and increase enrichment rate therefore appears to be a very promising option.

Figure 4-6. Neutron Transport Calculation Configuration for the LiH, UO₂ Full Fertile Blanket

TABLE 4-6

NEUTRONIC PERFORMANCE FOR THE $\text{UO}_2\text{-LiH}$ FULL FERTILE BLANKET,
80% QUALITY STEAM IN FERTILE ZONES

	<u>BOL</u>	<u>1 YEAR</u>	<u>2 YEARS</u>
T	1.06	1.13	1.19
F	1.11	1.01	0.93
T + F	2.17	2.14	2.12
G	3098	2826	2600
C	0	2962	5675
E_n	0.25	0.89	1.49
H	--	445	442
M	6.0	7.0	7.9
P_{Th}	4630	5380	6140
FM	0.96	0.75	0.61

TABLE 4-7 NEUTRON SOURCES AND SINKS FOR UO₂ FUELED BLANKETS AT BOL

All Values Are Per Fusion Neutron Born In The Plasma

<u>Neutron Sources</u>	<u>Li₂O Blankets</u>		<u>LiH Blankets</u>	
	<u>Full Fertile</u>	<u>Outer Only</u>	<u>Full Fertile</u>	<u>Outer Only</u>
²³⁵ U Fission	0.055	0.031	0.079	0.054
²³⁸ U Fission	0.990	0.767	1.054	0.841
²³⁸ U (n,2n) + (n,3n)	0.248	0.194	0.257	0.203
SS (n,2n)	0.048	0.056	0.047	0.056
Zr (n,2n)	0.111	0.087	0.116	0.092
Pb (n,2n)	--	--	0.068	0.186
TOTAL + 1	2.452	2.135	2.621	2.432
<u>Neutron Sinks</u>				
²³⁸ U (n,γ) = F	0.836	0.528	1.038	0.827
²³⁸ U (n,f)	0.266	0.205	0.286	0.228
²³⁵ U Absorption	0.030	0.016	0.043	0.029
SS Absorption	0.215	0.251	0.104	0.132
Zr Absorption	0.052	0.034	0.061	0.049
⁶ Li (n,α) T=T ₆	0.962	1.006	1.017	1.096
Other	0.091	0.095	0.072	0.071
TOTAL	2.452	2.135	2.621	2.432
T ₆ + F	73%	72%	78%	79%
<u>TOTAL</u>

Figure 4-7. Spatial Distribution of the Tritium Production Rate in the UO₂-LiH Blanket With Constant 80% Quality Steam Coolant.

Figure 4-8. Neutron Transport Calculation Configuration for the LiH, UO₂ Full Fertile Blanket, Lead Reflectors

TABLE 4-8

NEUTRONIC PERFORMANCE FOR THE $\text{UO}_2\text{-LiH}$, FULL FERTILE BLANKET,
80% QUALITY STEAM, LEAD REFLECTORS

	<u>BOL</u>	<u>1 YEAR</u>	<u>2 YEARS</u>
T	1.09	1.16	1.23
F	0.99	0.89	0.80
T + F	2.08	2.05	2.03
G	2802	2505	2263
C	0	2653	5037
E_n	0.25	1.06	1.81
H	--	316	314
M	5.4	6.5	7.5
P_{Th}	4185	5022	5752
FM	0.96	0.71	0.56

A four (4) year burnup calculation was next performed for the above configuration with 30% steam quality. The results are listed in Table 4-9. This table may be compared with that for the Li_2O case, Table 4-3. However, it should be kept in mind, as pointed out earlier, that the performance of the latter may be improved by increasing the lithium zone thicknesses. In particular, the difference in accumulative ^{239}Pu inventory can be decreased from 20% to about 13%. However, since this will involve an increase in the amount of UO_2 , enrichment will be essentially unchanged.

The time variation in fuel production rate and enrichment for the LiH full blanket case is shown in Figure 4-9. It can be seen that the net ^{239}Pu production rate drops more rapidly with time than with the Li_2O blanket (Figure 4-3) as a consequence of the softer neutron spectrum. However, tritium production increases more rapidly, so that the variation in total product rate is nearly the same for the two cases. Another consequence of the softer neutron spectrum for the LiH containing blanket is illustrated in Figure 4-10, where we have plotted the increase in thermal fission power with time for the Li_2O and LiH full fertile blankets. The results have been normalized to the LiH blanket value at BOL for convenient comparison. It can be seen that the rate of change of power is greater for the LiH blanket system.

Heating rate calculations were also performed for this system, and the results are shown in Figure 4-11.

4.3.2 FERTILE MATERIAL IN OUTER BLANKET ONLY

While the use of lead in the preceding case was optional, with the goal of reducing the fertile material inventory, lead was found to be necessary for the LiH outer fertile blanket only configuration. A problem run without lead, and a thick (24 cm) UO_2 outer blanket thickness, yielded a tritium breeding ratio of 1.41 and a total product rate of 1.94. This high tritium breeding could be reduced by utilizing some reflector material to redirect neutrons into the fertile zone and away from the LiH zones. A thick lead zone was first added to the inner blanket, adjacent to the plasma. This succeeded in

TABLE 4-9

NEUTRONIC PERFORMANCE FOR THE $\text{UO}_2\text{-LiH}$, FULL FERTILE BLANKET
30% CONSTANT QUALITY STEAM, LEAD REFLECTOR

	<u>BOL</u>	<u>1 YEAR</u>	<u>2 YEARS</u>	<u>3 YEARS</u>	<u>4 YEARS</u>
T	1.04	1.12	1.19	1.25	1.30
F	1.04	0.91	0.81	0.73	0.65
T + F	2.08	2.03	2.00	1.98	1.95
G	2925	2560	2280	2050	1830
C	--	2740	5160	7325	9265
E_n	0.25	1.08	1.83	2.53	3.17
H	--	320	316	314	311
M	5.5	6.7	7.7	8.6	9.4
P_{Th}	4260	5200	5950	6630	7225
FM	0.98	0.70	0.55	0.44	0.36

Figure 4-9. Time Dependent Variation of the Tritium and Fissile Production Rates and the Fissile Enrichment for the UO₂-LiH Full Fertile Blanket With Constant 30% Quality Steam Coolant.

Figure 4-10. Time Dependent Variation of the Thermal Power for the UO_2 -LiH Full Fertile Blankets With Constant 30% Quality Steam Coolant.

Figure 4-11. Heating Rate Spatial Profiles in the UO₂-LiH Full Fertile Blanket With Constant 30% Quality Steam Coolant.

reducing the tritium breeding ratio to 1.19, with a slight increase in total product rate to 2.00 due to $(n, 2n)$ multiplication in the lead. To reduce the tritium production rate still further, it was necessary to add a lead zone behind the UO_2 zone in the outer blanket. This configuration is shown in Figure 4-12, and the calculated results for a two-year burnup are listed in Table 4-10. For comparison with the full blanket case, this may be compared with Table 4-8. To facilitate comparison, the fissile production rate normalized to the same tritium production has been recalculated. These values are shown in parentheses. It can be seen that there is a 25% reduction in fissile production rate for the outer blanket only case. This is somewhat smaller than the 30% reduction found for the Li_2O .

The effect of a reduction in fertile zone structure was examined for this case. Since the structure consists only of zirconium, with a low parasitic absorption rate, the sensitivity would be expected to be less than for systems containing stainless steel structural material. The results are shown in Figure 4-13, where it can be seen that a 20% increase in fuel volume fraction results in about a 12% increase in both fissile production and power.

4.4 SECOND ITERATION COOLANT RESULTS

It should be noted that the previous analyses assumed a constant coolant quality of 30% or 80% throughout the blanket. The blanket design for the CTHR (see Figures 6-7 or 6-9), however, consists of three straight sections around the plasma in the poloidal direction with the coolant flowing along the fuel rods. Using a heating rate of 200 W/cm^3 , the thermal-hydraulic analysis determined that the coolant inlet and outlet qualities were approximately 3.5% and 30%, respectively. The mass weighted average water density in the blanket is then about 0.224 g/cm^3 . The effect of this higher water density on the blanket neutronics performance was estimated for both the LiH and Li_2O full fertile blanket cases by a simple increase in the fertile zone coolant density to this value. The geometrical configurations remained the same. The results are listed on Tables 4-11 and 4-12 and the results compared in Figures 4-14 and 4-15. In order to facilitate comparison, the results were normalized to the same tritium production at BOL, while preserving the total product production rate. These estimated normalized values are indicated in parentheses in the tables.

*Density Factor 0.5

Figure 4-12. Neutron Transport Calculation Configuration for the LiH, UO₂ Outer Fertile Blanket Only

TABLE 4-10

NEUTRONIC PERFORMANCE FOR THE $\text{UO}_2\text{-LiH}$, OUTER FERTILE BLANKET ONLY,
80% QUALITY STEAM

	<u>BOL</u>	<u>1 YEAR</u>	<u>2 YEARS</u>
T	1.11 (1.09)	1.18	1.24
F	0.83 (0.85)	0.73	0.66
T + F	1.94	1.91	1.90
G	2177 (2240)	1934	1741
C	0	2055	3892
E_n	0.25	1.07	1.82
H	--	243	241
M	4.6	5.5	7.2
P_{Th}	3428	4070	4568
FM	0.91	0.68	0.54

*Values in parentheses normalized to same tritium production as Table 4-8,
LiH full fertile blanket.

Figure 4-13. Effect of Variations in the Fuel-to-Structure Volume Ratio on the Power and Fissile Fuel Production in the UO₂-LiH Outer Fertile Blanket With Constant 80% Quality Steam Coolant.

TABLE 4-11

NEUTRONIC PERFORMANCE FOR THE $\text{UO}_2\text{-Li}_2\text{O}$, FULL FERTILE BLANKET, 30-40%
STEAM EXIT QUALITY (0.224 g/cm^3) WATER COOLANT IN FERTILE ZONES

	<u>BOL</u>	<u>1 YEAR</u>	<u>2 YEARS</u>	<u>3 YEARS</u>	<u>4 YEARS</u>
T	0.91 (1.03)*	0.98 (1.09)	1.02 (1.13)	1.07 (1.18)	1.10 (1.21)
F	0.97 (0.85)	0.77 (0.66)	0.66 (0.54)	0.58 (0.46)	0.50 (0.38)
T + F	1.88	1.75	1.68	1.65	1.60
G	2735 (2405)	2178 (1848)	1854 (1524)	1634 (1304)	1407 (1076)
C	--	2456 (2127)	4472 (3813)	6216 (5227)	7736 (6417)
E_n	0.25	0.99	1.71	2.36	2.96
H	--	308 (270)	297 (255)	291 (245)	285 (235)
M	5.5	7.0 (6.8)	7.9 (7.5)	8.8 (8.3)	9.4 (8.7)
P_{Th}	4249	5437 (5278)	6115 (5840)	6755 (6356)	7243 (6733)
FM	0.92 (0.81)	0.57 (0.50)	0.43 (0.37)	0.35 (0.29)	0.28 (0.23)

*Values in parentheses are normalized to the same tritium production as Table 4-3, 0.055 g/cm^3 coolant density.

TABLE 4-12

NEUTRONIC PERFORMANCE FOR THE $\text{UO}_2\text{-LiH}$, FULL FERTILE BLANKET, 30-40%
STEAM EXIT QUALITY (0.224 g/cm^3) WATER COOLANT IN FERTILE ZONES

	<u>BOL</u>	<u>1 YEAR</u>	<u>2 YEARS</u>	<u>3 YEARS</u>	<u>4 YEARS</u>
T	0.81 (1.04)*	0.93 (1.15)	1.00 (1.22)	1.09 (1.31)	1.15 (1.37)
F	1.24 (1.02)	0.94 (0.72)	0.79 (0.57)	0.69 (0.47)	0.59 (0.37)
T + F	2.05	1.87	1.79	1.78	1.74
G	3487 (2862)	2661 (2035)	2235 (1609)	1961 (1335)	1666 (1039)
C	--	3074 (2449)	5522 (4271)	7620 (5743)	9434 (6930)
E_n	0.25	1.00	1.76	2.45	3.08
H	--	378 (305)	355 (277)	344 (259)	336 (244)
M	6.2	8.2 (7.7)	9.4 (8.7)	10.6 (9.5)	11.4 (10.0)
P_{Th}	4834	6350 (6040)	7240 (6695)	8125 (7314)	8773 (7727)
FM	1.03 (0.84)	0.60 (0.48)	0.44 (0.34)	0.34 (0.26)	0.27 (0.19)

*Values in parentheses are normalized to the same tritium production as Table 4-9, 0.055 g/cm^3 coolant density.

Figure 4-14. Time Dependent Variation of the Fissile Fuel and Power Production for two Different Water Coolant Densities in the UO₂-Li₂O Full Blanket Concept.

Figure 4-15. Time Dependent Variation of the Fissile Fuel and Power Production for two Different Water Coolant Densities in the UO₂-LiH Full Blanket Concept.

As expected, the thermal power increases more rapidly with time while the ^{239}Pu production rate decreases in the softer neutron spectrum for the higher density coolant. Moreover, while the total product rates are nearly unchanged at BOL, they decrease about 10% at EOL (4 years) relative to the low density coolant cases. It is clear that the economics for the two coolant densities will be quite different.

5.0 UC FUELED BLANKET NUCLEAR ANALYSIS

While the preceding section discussed the neutronic performance of a UO_2 fueled, boiling water cooled blanket concept, in this section an investigation is made of a more advanced UC fueled, helium cooled concept. The structural and clad material was assumed to be stainless steel (SS), while in the UO_2 concept zircaloy was assumed. Because of the increased uranium density with the UC fuel and the harder neutron spectrum with the He, this concept was expected to result in improved neutronic performance relative to the UO_2 , boiling water concept.

The same two solid lithium compounds, Li_2O and LiH , were investigated for the UC fueled blanket. In addition, the use of a lead multiplier/reflector zone was evaluated. Both the lithium and lead zones were assumed to have SS as structure with He as the coolant. Volume fractions and densities for the various materials in the blanket were shown in Table 2-1.

The various blanket zone thicknesses were adjusted for the cases considered so that a tritium breeding ratio T of $\sim 1.05 - 1.10$ was achieved at beginning-of-life (BOL). The time dependent behavior of these blankets was then evaluated during irradiation of sufficient length to achieve a fissile enrichment of about three percent. The wall loading as well as other important tokamak parameters assumed for the exposure studies were listed in Table 2-3.

As mentioned previously, an important aspect of this study was to investigate the neutronic benefits in a tokamak of full blanket wall coverage vs. outer only fertile blanket coverage, with the inner regions reserved for tritium breeding only. Accordingly, the composition of the inner blanket region was varied to investigate the effect of using lithium only, lithium with a lead multiplier, and fertile material with lead and lithium in these zones.

5.1 UC - Li₂O OUTER BLANKET

The initial neutronics calculations were performed for a case with fertile blanket coverage on the outer region of the torus only, with Li₂O fueled tritium breeding zones on the inner region of the torus and in the back of the UC zone on the outer blanket. An illustration of the geometry and zone thicknesses used for the neutronics calculation is shown in Figure 5-1. The UC blanket zone thickness was adjusted to 24 cm in order to achieve the desired tritium breeding ratio, while the Li₂O zones were 60 cm thick in both the outer and inner regions.

The performance of this arrangement (denoted as blanket I) during a two year exposure period is shown in Table 5-1. As the table illustrates, the blanket tritium breeding ratio increases from 1.08 at BOL to 1.17 after two years, while the net fissile production F decreases and the sum of T and F is approximately constant. (See section 2.0 for definitions of T , F , and M). Scaling by the wall coverage and tokamak device parameters as discussed in sections 2.0 and 3.0 results in a net fissile production rate G of ~ 2200 kg ²³⁹Pu/yr, while the average fissile enrichment of ²³⁹Pu plus ²³⁵U in the blanket is calculated to be about 1.4%. There is about a 24% increase in the plant power production during the two year irradiation period, due to increased fissioning of the ²³⁹Pu being bred in the blanket. While these fissile and power production rates appear to be attractive for a tokamak hybrid, it should be stressed that this particular blanket concept is only an initial starting point. Various ways to increase the production rates are investigated below and in subsequent sections.

One obvious way to improve the performance is to increase the fuel to structure (UC/SS) ratio in the blanket. The results of varying this ratio for the He cooled outer UC blanket case with Li₂O are shown in Figure 5-2. Values of the fissile and power production relative to the case of Table 5-1 where the ratio was UC/SS = 0.5/0.2 = 2.5 are plotted vs. the fuel/structure ratio. For this comparison the F value was adjusted so that $T = 1.10$ at BOL. As Figure 5-2 illustrates, there is a strong incentive to maximize this ratio; e.g., going from the UC/SS = 0.5/0.2 case to the 0.6/0.1 case increases the fissile production by about 29% and the power production by about 18% at BOL.

Figure 5-1. Schematic Diagram of the Zone Thicknesses and Material Compositions Used in the Neutronics Analysis of the UC-Li₂O Outer Blanket Coverage Concept

TABLE 5-1

PERFORMANCE OF THE UC-Li₂O, OUTER BLANKET CONCEPT
DURING TWO YEARS OF IRRADIATION

		<u>BOL</u>	<u>1 YEAR</u>	<u>2 YEARS</u>
Integrated Wall Load,	$\frac{\text{MW-Yrs}}{\text{m}^2}$	0	1.2	2.4
T,	$\frac{\text{Tritium Atoms}}{\text{Fusion Neutron}}$	1.08	1.12	1.17
F,	$\frac{\text{Net } ^{239}\text{Pu atoms}}{\text{Fusion Neutron}}$	0.88	0.84	0.80
T + F		1.96	1.96	1.97
G,	$\frac{\text{kg } ^{239}\text{Pu}}{\text{yr}}$	2320	2210	2100
C,	Cumulative ^{239}Pu (kg)	--	2265	4420
M,	energy multiplication	4.98	5.61	6.24
P _{Th} ,	thermal power, MWt	3710	4160	4610
Enrichment, w/o fissile				
	In 1st 10 cm	0.25	0.95	1.61
	In 2nd 14 cm	0.25	0.75	1.23
	Blanket Average	0.25	0.83	1.39

Figure 5-2. Effect of Variations in the Fuel-to-Structure Volume Ratio on the UC-Li₂O Outer Blanket Fissile Atom Production and Energy Multiplication.

This could be significant in terms of economic impact, since it translates into the refueling of an additional one or two fission reactors per year. Conversely, decreasing the fuel-to-structure ratio can exact a potentially large penalty on the fuel production, especially if tritium self-sufficiency is required.

5.2 COMPARISON OF Li_2O VS. LiH

In this section a comparison is made of the performance of two possible tritium breeding media, Li_2O and LiH , in conjunction with the UC outer blanket. The volume fractions and densities of the materials are shown in Tables 2-1 and 2-2. The lithium in all cases was assumed to be of natural isotopic composition, i.e., 7.5% ^6Li .

Simple schematic diagrams of four blanket configurations resulting from this analysis, along with their performance at BOL, are shown in Table 5-2. Note that for blanket II, the only change from blanket I is the direct substitution of LiH for Li_2O . Otherwise the same configuration as shown in Figure 5-1 was used. As the table shows, there is a dramatic difference in the blanket performance between cases I and II.

The use of the LiH shifts the neutron absorption away from the fertile zone and into the lithium zone, although the total product rate T and F is about the same. The reason for this difference is the presence of hydrogen in the LiH compound, and its very efficient moderating ability relative to the oxygen in the Li_2O . What occurs is that there is a significant leakage of fission spectrum type (~ 1 MeV) neutrons, produced from neutron multiplying reactions in the UC zone, from both the front and rear of the UC zone. The neutrons leaking out the front of this zone stream toward the inner lithium bearing zone. With the Li_2O , a significant fraction of these neutrons are reflected back into the fertile blanket. However, with the LiH more of these neutrons are moderated and then absorbed in the ^6Li , hence the higher tritium production and lower fissile production. The same effect occurs for neutrons leaking out the rear of the fertile zone--with the Li_2O there is more

TABLE 5-2

COMPARISON OF Li_2O VS. LiH IN CONJUNCTION WITH THE
UC OUTER BLANKET COVERAGE CONCEPT

PERFORMANCE AT BOL

	I	II	III	IV
T , total	1.08	1.40	1.29	1.08
T , inner blanket	0.63	0.88	0.90	0.91
T , outer blanket	0.45	0.52	0.39	0.17
F	0.88	0.58	0.68	0.90
$T + F$	1.96	1.98	1.97	1.98
M	4.98	4.81	4.86	5.10
Leakage, $\frac{\text{neutrons}}{\text{fusion neutron}}$	0.01	0.001	0.009	0.003

reflection back into the UC zone while with the LiH there is more moderation and downscattering. There is also a slight decrease in M with the LiH because of this effect.

In an attempt to reduce the tritium production with the LiH in blanket II, a lead reflector zone was inserted in back of the UC zone. The composition by volume of this zone was assumed to be 70% Pb, 10% SS, and 20% He. The result for the case of a 10 cm thick Pb zone is shown as blanket III in Table 5-2. As the table shows, the LiH zone thicknesses were decreased in blanket III relative to blanket II because it was found that a 30-40 cm thickness of LiH was more than adequate to absorb most all of the neutrons. This change has only a very minor effect on the blanket performance, as shown by the neutron leakage values in Table 5-2 which indicate that the leakage is less than 0.5% of the T and F value for all the blankets that were analyzed.

The placement of the Pb reflector achieves the desired result of reducing T and increasing F, viz. blankets II and III in Table 5-2. Note that most of the decrease in T comes from a reduction in the outer LiH blanket production, and in fact the inner zone tritium production actually increases slightly. Based on this analysis, a series of runs were made in which the UC and Pb zone thicknesses were varied until the arrangement shown as blanket IV in Table 5-2 was chosen, with a UC zone 32 cm thick and a Pb zone 24 cm thick. Keeping the UC zone thickness a constant and simply increasing the Pb reflector zone thickness becomes counter-productive at a certain Pb thickness, because parasitic absorptions in the SS in the Pb zone off-set the zone reflection properties and the blanket production rate begins to decrease. Hence, to further reduce the tritium breeding it is necessary to increase the UC zone thickness as well as the Pb zone, as was done for blanket IV. While the zone thicknesses in IV have not been optimized, they are representative values of the configuration necessary for the UC-LiH outer blanket concept.

Comparing blankets I and IV in Table 5-2 indicates that the fissile production F is about 2% less for the Li_2O case relative to the LiH-Pb case. However, the UC fuel inventory is larger with the LiH blanket even with the Pb reflector, which would adversely affect the enrichment rate and the inventory charge.

The LiH does appear to allow lower lithium compound inventories due to the superior moderating properties of the hydrogen. Hence there clearly are trade-offs involved in the choice of Li_2O or LiH as the tritium breeding medium. Because of the thinner blankets needed for the LiH, and because of its slightly larger fissile production rate, the LiH case has been chosen for further investigation in the CTHR UC fueled blanket. Methods to reduce the required UC inventory (zone thickness) in the blanket by the use of multipliers (Pb or UC) on the inner blanket are investigated in the following section.

5.3 INNER BLANKET COMPOSITION VARIATIONS

While the preceding section considered outer fertile blanket coverage only, in this section variations on the composition of the inner blanket region to include neutron multipliers as well as just lithium have been evaluated. Simple schematic diagrams of the three inner blanket variations considered are shown in Figure 5-3. Two types of multipliers were analyzed, either Pb or UC. The latter corresponds to the case of full blanket coverage. Hence this analysis will provide important information on the neutronic advantages of full fertile blanket coverage, with its attendant complications, vs. outer only fertile coverage. In calculating the various zone thicknesses for the arrangements in Figure 5-3, the constraint of achieving a tritium breeding ratio T of ~ 1.05 - 1.10 at BOL was used.

The results of the blanket performance for the various inner blanket zone compositions are given in Figure 5-4, where F, T, and $F + T$ are plotted vs. irradiation time. As expected, the UC inner blanket has the largest production, followed by the Pb inner zone case and then the LiH only case. The UC inner blanket F value at BOL is $\sim 13\%$ and $\sim 22\%$ larger than the Pb inner and LiH inner cases, respectively. The difference in F decreases with exposure because T is increasing, as shown in Figure 5-4. Note that the sum of $T + F$ stays fairly constant with irradiation for all of these cases. While Figure 5-4 shows the F and T values for the tritium self-sufficient cases with different inner blanket compositions, it does not include other considerations such as zone thicknesses, material inventories, blanket coverage factors, enrichment rates, and macroscopic fissile and power production rates.

- He coolant, SS structure in all zones

Figure 5-3. Simple Schematic Illustration of Three Inner Blanket Arrangement Variations Considered for the CTHR

Figure 5-4. Time Dependent Variation of Fissile and Tritium Atom Production for Three Different Inner Blanket Zone Arrangements With a UC Outer Blanket.

These parameters at BOL and after four years (4.8 MW-yr-m^2) of exposure are given in Table 5-3 for the different cases. The thicknesses of the various zones as shown in Figure 5-3 were chosen through a series of parametric runs for each configuration so that T at BOL was ~ 1.05 - 1.10 . They do not, however, represent fully optimized results. It can be seen that the UC zone thickness decreases with the Pb and UC inner cases to 24 cm and 16 cm. This is an advantage with regard to the enrichment, which as shown in the table increases to $\sim 2.9\%$ fissile after four years in the full blanket case. The fissile production also is larger for the full blanket case, as is the relative power increase from BOL to four years. For all of the cases there is a slight decrease in the blanket heavy metal (U+Pu) inventory with exposure due to fuel burnup. Also given in Table 5-3 is the figure-of-merit (FM) for the blanket performance, defined in section 4.2.1 as the ratio of the fissile production rate to the plant thermal power in units of kg fissile/MWt-yr. The values for the FM decrease from ~ 0.9 - 1.0 at BOL to ~ 0.5 - 0.6 at four years of exposure, which is consistent with values calculated by other researchers⁽¹¹⁾.

The major reasons for the differences in performance among these three blankets can be seen in Table 5-4, where a detailed neutron balance of the various neutron sources and sinks at BOL in the blankets is given. Note that there are various trade-offs involved in the total production rate $T + F$ among the competing high neutron energy reactions. For example, the gain from Pb ($n, 2n$) reactions with the Pb inner zone case is partially off-set by the higher tritium production from the ${}^7\text{Li} (n, n'\alpha)T$ reactions in the LiH inner zone case. For each of the blankets, approximately 79% of the available neutron population results in fissile or tritium breeding through ${}^{238}\text{U}(n, \gamma)$ or ${}^6\text{Li}(n, \alpha)$ reactions.

The UC inner blanket zone case does result in ~ 13 - 21% higher F values due to increased ${}^{238}\text{U}$ neutron multiplying reactions. This increase is somewhat smaller than might be expected from simple inner-outer wall coverage factors, based on $\sim 72\%$ of the fusion neutrons intercepting the outer blanket in the ANISN one-dimensional approximation and $\sim 28\%$ intercepting the inner blanket.

TABLE 5-3

PERFORMANCE PARAMETERS OF BLANKETS WITH DIFFERENT INNER BLANKET ZONE COMPOSITIONS

All Are UC, LiH Fueled, With SS Structure And He Coolant

	<u>LiH Only</u> <u>Inner Blanket</u>		<u>Pb-LiH</u> <u>Inner Blanket</u>		<u>UC-Pb-LiH</u> <u>Inner Blanket</u>	
UC Zone Thickness	32 cm		24 cm		16 cm	
	<u>AT</u> <u>BOL</u>	<u>AT</u> <u>4 YEARS</u>	<u>AT</u> <u>BOL</u>	<u>AT</u> <u>4 YEARS</u>	<u>AT</u> <u>BOL</u>	<u>AT</u> <u>4 YEARS</u>
Integrated $\frac{\text{MW-Yrs}}{\text{m}^2}$ Wall Load,	--	4.8	--	4.8	--	4.8
F, $\frac{\text{Net } ^{239}\text{Pu atoms}}{\text{Fusion Neutron}}$	0.90	0.78	0.97	0.78	1.10	0.85
T	1.08	1.21	1.06	1.29	1.05	1.34
T + F,	1.98	1.99	2.03	2.07	2.15	2.19
G, $\frac{\text{kg } ^{239}\text{Pu}}{\text{yr}}$	2380	2050	2570	2050	3091	2406
Cumulative ^{239}Pu (kg)	--	8840	--	9210	--	10955
Relative Fissile Prod.	--	1.00	--	1.04	--	1.24
Avg. Fissile Enrichment	--	1.96%	--	2.64%	--	2.89%
Heavy Metal Inventory, THM	511	506	382	376	412	406
M	5.10	7.26	5.06	8.30	5.92	9.92
P_{Th} , thermal power, MWt.	3800	5340	3760	6080	4580	7620
FM, $\frac{\text{kg}}{\text{MWt-yr}}$	0.94	0.58	1.02	0.51	1.01	0.47

TABLE 5-4

DETAILED NEUTRON BALANCE ON THE NEUTRON SOURCES AND SINKS AT BOL
IN THE DIFFERENT INNER BLANKET ZONE COMPOSITION CASES

All Values Are Per Fusion Neutron Born In The Plasma

	<u>LiH</u> <u>Inner Zone</u>	<u>Pb</u> <u>Inner Zone</u>	<u>UC</u> <u>Inner Zone</u>
UC Zone Thickness	32 cm	24 cm	16 cm
Neutron Sources			
- Plasma	1.000	1.000	1.000
- U Fission	1.055	1.037	1.265
- U (n,2n) & (n,3n)	0.256	0.248	0.308
- Pb (n,2n)	0.010	0.154	0.054
- Other (n,2n)	<u>0.104</u>	<u>0.105</u>	<u>0.107</u>
- TOTAL	2.425	2.544	2.734
Neutron Sinks			
- ^{238}U (n, γ) = F	0.903	0.972	1.096
- ^{238}U fission	0.270	0.266	0.324
- ^{235}U absorption	0.025	0.026	0.029
- SS absorption	0.195	0.207	0.215
- ^6Li (n, α) T = T ₆	1.004	1.045	1.040
- Other	<u>0.028</u>	<u>0.028</u>	<u>0.028</u>
- TOTAL	2.425	2.544	2.734
^7Li (n,n' α) T = T ₇	0.073	0.017	0.013
T = T ₆ + T ₇	1.077	1.062	1.053
T + F	1.980	2.034	2.149
(T ₆ + F) ÷ Total Sources	79%	79%	78%

Based on this, an increase of roughly 0.28/0.72 or 39% should occur for the full fertile blanket coverage. That the actual difference is found to be somewhat smaller is due mainly to two reasons. First, some of the difference is off-set by higher tritium production from ^7Li or the $\text{Pb}(n, 2n)$ multiplication in the LiH and Pb inner zone cases. Secondly, the UC zone is thicker in the LiH and Pb cases, which results in proportionately higher per fusion neutron multiplication from the UC zone. Hence, while these results indicate that there is the potential for improved neutronic performance with full fertile blanket coverage, other factors including economic considerations must be included to determine the ultimate benefit.

One such factor that must be considered is a possible change in the tokamak plant availability factor PA due to the use of fertile material in the inner blanket zones. The results presented previously in Table 5-3 for the fissile production G (kg/yr) were based on a $\text{PA} = 0.70$, which was considered to be the maximum potential value for a CTHR device with outer fertile blanket coverage. Preliminary estimates⁽¹²⁾ made for the PA factor with full fertile blanket coverage range from 0.55 to 0.61, depending on whether the tokamak is assumed to have a separate vacuum vessel which must be removed to refuel the inner blanket or whether there is no vacuum vessel and the blanket modules directly face the plasma. Since the PA factor is a direct multiplier on the fissile production rate, this will be important in assessing the usefulness of full blanket coverage.

A plot of the fissile production vs. exposure for different values of PA for both the full blanket and outer blanket only with a Pb-Li inner blanket zone is shown in Figure 5-5. As the figure shows, the assumption of lower PA (0.61 or 0.55) for the full blanket tends to negate its performance advantage relative to the outer coverage case. Hence, if these lower plant availability factors cannot be improved, the potential benefit of using fertile material in the inner regions of the tokamak may not be realized.

Figure 5-5. Time Dependent Variation of the Fissile Fuel Production Rate for the Full Fertile Blanket and Outer Fertile Blanket Concepts With Different Values of the Reactor Plant Availability (PA).

5.4 UC-LiH FULL BLANKET CONCEPT PERFORMANCE

In this section a more detailed description of the performance of the UC-LiH blanket concept with full blanket coverage (UC in both the inner and outer tokamak regions) is presented and discussed. A schematic diagram of the geometry used for the neutronics analysis is given in Figure 5-6. The thickness of both UC zones was 16 cm with a 10 cm thick Pb reflector zone located behind the UC zone. The thickness of the LiH zones were set at 30 cm for the outer blanket and 34 cm for the inner blanket in order to keep the neutron leakage at a sufficiently small value.

Important blanket neutronic and performance parameters as a function of exposure in the CTHR are given in Tables 5-5 and 5-6. From the neutronic parameters in Table 5-5, it can be seen that while the total (T + F) particle production rate increases only slightly with exposure, the individual components T and F behave quite differently. That is, the tritium production T increases while the net fissile production F decreases during exposure of the blanket. This is due to the increased fissioning of ^{239}Pu , as shown in Table 5-5, which causes F to decrease and the leakage into the lithium zone to increase. Note the steadily increasing ^{239}Pu fission rate, which after four years contributes ~40% to the total blanket fission rate. The fractions of total tritium, fissile fuel, and power produced in the outer blanket regions range from ~66-72% as shown in Table 5-5.

By using the assumed CTHR blanket coverage factors discussed in section 3.0 and the fusion power and neutron wall load shown in Table 2-3, the blanket neutronic parameters can be converted to the macroscopic fuel and power production quantities which are shown in Table 5-6. The instantaneous ^{239}Pu production rate G decreases by ~22% over the four year exposure period, but the average production rate is still fairly large at ~2730 kg/yr. The fuel experiences a significant burn-up of >12,000 MWD/THM during the four years. After this time the fissile enrichment is ~2.9%, with composition as shown in Table 5-6. The blanket heavy metal (U+Pu) inventory decreases slightly during irradiation due to fuel burn-up. Note that the figure-of-merit (FM) is about 1.0 at BOL but decreases to ~0.5 after four years. The power production rises by ~66% to 7600 MWt, which may be unacceptable from a utility standpoint.

Figure 5-6. Schematic Diagram of the Geometry Used in the Neutronics Analysis of the UC-LiH Full Blanket Coverage Concept With Pb Reflectors

TABLE 5-5

NEUTRONIC PARAMETER VALUES VS. EXPOSURE FOR THE UC-LiH
FULL BLANKET COVERAGE CONCEPT

All Reaction Values Are Per Fusion Neutron

	<u>AT BOL</u>	<u>AT 1 YEAR</u>	<u>AT 2 YEARS</u>	<u>AT 3 YEARS</u>	<u>AT 4 YEARS</u>
Integrated $\frac{\text{MW-Yrs}}{\text{m}^2}$ Wall Load,	--	1.2	2.4	3.6	4.8
I , Tritium Breeding Ratio	1.053	1.126	1.197	1.269	1.342
- T_6 , ${}^6\text{Li}$ (n, α) T	1.040	1.113	1.184	1.256	1.329
- T_7 , ${}^7\text{Li}$ (n,n' α) T	0.013	0.013	0.013	0.013	0.013
No. ${}^{238}\text{U}$ (n, γ) rxns.	1.096	1.111	1.129	1.148	1.167
No. ${}^{239}\text{Pu}$ abs. rxns.	--	0.082	0.161	0.239	0.316
$F_{\text{net}} = {}^{238}\text{U}$ (n, γ) - ${}^{239}\text{Pu}$ abs.	1.096	1.029	0.968	0.909	0.851
$T + F_{\text{net}}$	2.149	2.155	2.165	2.178	2.193
No. ${}^{235}\text{U}$ fission rxns.	0.022	0.021	0.019	0.018	0.017
No. ${}^{238}\text{U}$ fission rxns.	0.325	0.334	0.345	0.354	0.364
No. ${}^{239}\text{Pu}$ fission rxns.	--	0.062	0.124	0.186	0.248
Total fissions rxns.	0.347	0.417	0.488	0.558	0.629
Multiplication M	5.918	6.921	7.918	8.912	9.916
Fractions produced in outer blanket:					
- Tritium	0.704	0.703	0.701	0.700	0.699
- Fissile	0.672	0.669	0.666	0.663	0.661
- Power	0.724	0.717	0.713	0.708	0.705

TABLE 5-6

IMPORTANT BLANKET PERFORMANCE PARAMETERS VS. EXPOSURE TIME
FOR IRRADIATION OF THE UC-LiH FULL BLANKET CONCEPT IN THE CTHR

	<u>BOL</u>	<u>AT 1 YR</u>	<u>AT 2 YRS</u>	<u>AT 3 YRS</u>	<u>AT 4 YRS</u>
Integrated Wall Load $\frac{\text{MW-yrs}}{\text{m}^2}$	--	1.2	2.4	3.6	4.8
G, $\frac{\text{kg } ^{239}\text{Pu}}{\text{yr}}$ (Instantaneous)	3090	2910	2730	2570	2410
Cumulative ^{239}Pu , Tonnes	--	3.00	5.82	8.47	10.96
Heavy Metal Inventory, THM	412	411	410	408	406
P_{Th} , Thermal Power, MWt	4580	5350	6100	6860	7620
Cumulative Burnup $\frac{\text{MWD}}{\text{THM}}$	--	2480	5420	8820	12670
Fissile Enrichment, Pct.	0.25	0.96	1.64	2.28	2.89
- ^{235}U	0.25	0.23	0.22	0.20	0.19
- ^{239}Pu	--	0.73	1.42	2.08	2.70
- ^{241}Pu	--	0.0002	0.0009	0.0023	0.0042
FM, $\frac{\text{kg } ^{239}\text{Pu}}{\text{MWt-yr}}$	1.01	0.82	0.67	0.56	0.47

Hence, most likely some sort of fuel management scheme involving blanket zone rotation or shuffling would be implemented for an optimized design.

The power densities due to nuclear heating (both neutron and gamma radiation) are shown as a function of position in the blanket at BOL and after four years in Figure 5-7. The values shown are those during the plasma burn with a wall loading of 2 MW/m^2 . The power densities are $>100 \text{ watts/cm}^3$ at the front of the UC zones, but fall off exponentially with depth into the blanket. The heat generation in the Pb zone is low, due mostly to gamma ray absorption; and then is larger in the LiH zone due to the ${}^6\text{Li}(n, \alpha)\text{T}$ reaction.

An expanded version of the nuclear heating rates in the fertile fueled blanket regions is shown in Figure 5-8. Here the heating has been broken up into that deposited in the UC fuel, which is 50% of the blanket volume, and in the stainless steel (SS) clad and structure, which is 20% of the blanket volume. The remaining 30% is the He coolant. Hence, to obtain the actual heat deposition in the UC or the SS, the values in Figure 5-8 should be divided by the corresponding volume fractions. This results in a maximum heating rate in the UC of $> 400 \text{ watts/cm}^3$ in the first few cm of the outer blanket after four years of operation. Corresponding peak heating rates in the SS are $\sim 25 \text{ watts/cm}^3$. The ratio of the peak-to-average heating in the blanket zones is ~ 2.2 at BOL but decreases to ~ 1.8 after four years, which is an important aspect with regard to efficient cooling design of the blanket.

To obtain an estimate of the decay heat behavior in this blanket during the pulsing of the plasma, a calculation was made using a simple analytical formula⁽¹³⁾ slightly modified to account for many repetitive cycles of operation. For an assumed plasma pulse minimum on-time of 135 sec followed by an off-time of 15 sec, the approximate afterheat due to fission product decay during the off-time is as shown in Figure 5-9. Here the afterheat is defined to be the percent of the operating power during the pulse. As shown in Figure 5-9, the afterheat at 1 sec after shut-off is $\sim 4.5\%$, and decays to $\sim 2\%$ after 15 sec. While these are fairly low values, they still illustrate the need both for some type of thermal energy storage system to smooth out the heat load to the turbines and also for some type of emergency cooling system in the event of a loss-of-coolant condition.

Figure 5-7. Heating Rate Spatial Profiles in the UC-LiH Full Fertile Blanket Concept.

Figure 5-8. Blanket Power Density Profiles in the UC and Stainless Steel (SS) in the Fertile Fueled Module Regions.

Figure 5-9. Approximate Behavior of the Decay Heat in the UC Blanket Concept.

While the spatial variation of the power densities in the blanket is important, another interesting property of the blanket design is the spatial distribution of the fissile enrichment. A profile of the fissile enrichment in the blanket after four years of irradiation, without any fuel shuffling, is shown in Figure 5-10. At BOL the blanket consists of depleted UC fuel with a ^{235}U enrichment of 0.25%. After four years of operation, the ^{235}U concentration has decreased to an average value of $\sim 0.18\%$. The ^{239}Pu enrichment is $\sim 3.1\%$ at the front of the blanket regions, as shown in Figure 5-10, and decreases with depth into the blanket to a value of $\sim 2.1\text{--}2.2\%$. This results in a spatially averaged ^{239}Pu enrichment of $\sim 2.6\text{--}2.7\%$. The peak to average enrichment ratio is $\sim 1.15\text{--}1.19$, and could be reduced further through the use of a fuel management scheme involving zone rotation or shuffling. Such a strategy was not considered during this analysis, but would be a part of future work to optimize the blanket design.

The variation of the isotopic composition of the bred Pu as a function of irradiation time of the blanket is shown in Figure 5-11. The Pu is fairly pure, being $>95\%$ ^{239}Pu even after four years of exposure. The other major isotopes present are ^{240}Pu and ^{238}Pu . The former is produced from ^{239}Pu capture reactions, while the latter results mainly from capture reactions in the ^{237}Np which is produced from $(n,2n)$ reactions on ^{238}U (see Figure 2-2 for the decay chains). The fissile isotope ^{241}Pu is present only in small amounts. Even less ^{242}Pu is produced, with a concentration smaller than 0.01%.

An important consideration in any fusion reactor design is radiation damage to the vacuum vessel and associated structural components of the first wall. The neutron fluence ($E > 0.1$ MeV) and neutron induced damage rates to the CTHR stainless steel 316 (SS-316) outer portion of the vacuum vessel with full blanket coverage for both the UC-LiH and $\text{UO}_2\text{-Li}_2\text{O}$ blankets are given in Table 5-7. Here the vessel was modeled as a 0.5 cm thick layer of SS-316 for the neutronics calculations. It can be seen that the atomic displacement rate increases by $\sim 12\%$ in both cases from BOL to four years exposure, due to increased fissioning in the blanket. This fissioning increases the neutron flux level in the 1-2 MeV energy range, which is a major contributor to the displacement cross section. On the other hand, the He production rate stays

Figure 5-10. Blanket Enrichment Spatial Profiles After Four Years of Irradiation in the UC Blanket Concept.

Figure 5-11. Time Dependent Variation of the Isotopic Composition of the Bred Plutonium in the UC Blanket Concept.

TABLE 5-7

RADIATION DAMAGE RATES IN THE STAINLESS STEEL VACUUM
VESSEL WALL OUTER SURFACE

	<u>UC-LiH</u>		<u>UO₂-Li₂O</u>	
	<u>AT BOL</u>	<u>AFTER 4 YEARS</u>	<u>AT BOL</u>	<u>AFTER 4 YEARS</u>
Integrated Wall Loading (MW-yrs/m ²)	--	4.8	--	4.8
Atomic Displacement Rate, 10 ⁻⁷ DPA ^(a) /s	6.90	7.76	6.58	7.32
Cumulative DPA	--	92	--	88
He Production Rate, 10 ⁻⁶ appm ^(b) He/s	7.39	7.40	7.39	7.41
Cumulative appm He	--	930	--	934
Fluence, E > 0.1 MeV (n/cm ²)	--	1.14 x 10 ²³	--	9.60 x 10 ²²

(a) DPA - displacements per atom

(b) appm - atomic parts per million

nearly constant with exposure because the He is produced mainly by neutrons with energies greater than ~ 8 MeV, whose flux level does not change appreciably with irradiation time. The cumulative damage values of about 90 DPA and 930 appm He after four years or 4.8 MW-yr-m^{-2} of exposure will result in some embrittlement and swelling of the SS-316, with the severity depending mainly on the operating temperature of the vessel. The values in Table 5-7 are for the outer portion of the first wall, where the wall loading is highest. The damage rates in the inner portion of the wall are calculated to be ~ 15 -20% lower, due to the lower wall loading in this region as discussed in Section 3.0 and illustrated in Figure 3-1. The radiation damage in the SS-316 clad and structure in the front of the fertile blanket zone is ~ 12 -21% lower than the first wall values shown in Table 5-7, and decreases further with depth into the blanket.

As a final step in the analysis of the UC-LiH full blanket coverage concept, a parametric study was performed on the ratio of fuel-to-structure in the fertile blanket regions. The thickness of the inner and outer blankets was kept at 16 cm, and the UC to SS volume ratio was varied from the reference value of $0.5/0.2 = 2.5$. The results of this analysis in terms of change in F, T, and M at BOL are shown in Figure 5-12. In going from the reference value of 2.5 to a UC-to-SS ratio of $0.6/0.1 = 6.0$, the fissile production increases by 24% while the tritium and power production also increase significantly. Conversely, decreasing the ratio to $0.45/0.25 = 1.8$ begins to severely degrade the blanket performance. Hence there is a strong motive for reducing the structural volume fraction in this blanket in order to maximize the fissile, power, and tritium production.

Figure 5-12. Effect of Variations in the Fuel-to-Structure Volume Ratio on the Tritium, Fissile, and Energy Production in the UC-LiH Full Blanket Concept.

6.0 BLANKET MECHANICAL DESIGN

The two generic blanket concepts were developed on the basis that a separate vacuum vessel-first wall was part of the overall reactor design. The vacuum vessel assumed in the reactor was similar to the vacuum vessel structure developed for DTFR⁽²⁾. The potential exists whereby the first structural wall of the blanket can be used as the first wall. However, additional analysis is required to substantiate this alternative.

Parametric neutronic studies discussed in sections 4.0 and 5.0 identified an acceptable range of volume fractions for the blanket materials which yield acceptable blanket performance with regard to fissile and fusile fuel breeding. These projected volume fractions were the basis for the initial mechanical design concepts. Final iteration with the actual blanket volume fractions remains to be done after optimization of fuel zones thicknesses is achieved.

6.1 UC/LiH-HELIUM COOLED BLANKET MODULE DESIGN

The UC/LiH combination of blanket materials provides the necessary ingredients for an acceptable performing blanket with regard to fissile-fusile yield and power density. The He coolant provides a compatible medium with regard to corrosion and chemical reaction with the environment and has low neutron absorption characteristics not affecting the neutron population in the blanket. In order to have high thermal efficiency for the power conversion cycle the blanket coolant exit temperature should be as high as practical consistent with the structural temperature limit. The structural material selected for the blanket and associated components - fuel clad - coolant tubes - etc., is 316 stainless steel because of its high operating temperature limit of 500°C for pressure containment walls and 550°C for the clad. The coolant exit temperature can then be as high as 450°C. In order to have low pumping power requirements, it is necessary to minimize flow path length and maximize the flow cross-sectional area consistent with fuel production requirements.

The helium cooled UC/LiH blanket concept selected is shown in Figure 6-1. The basic characteristics of the blanket are listed in Table 6-1. The individual blanket module consists of a cylinder filled with microspheres (or alternately 2.5 cm thick disks) of UC fuel with coolant tubes passing through the fertile fuel. The fuel region located towards the plasma has a tight coolant tube lattice which is necessary to remove the high rate of heat generation associated with that zone. The tube pitch becomes relaxed radially deeper into the UC fuel region. This is due to the decreasing power density distribution through the fuel where the peak is located at the front wall and rapidly falls off radially from the front zone, as shown in Figures 5-7 and 5-8. The cylindrical UC fuel section is contained in a canister type module that has an individually cooled front wall--which is necessary to keep the structural temperature below the allowable limit of 500°C. The canisters are oriented in the toroidal direction around the plasma with the inlet and outlet manifolds running in the poloidal direction. The containing vessel has hemispherical end caps with inlet/outlet flow channels that are fed from and return to the radially outward coolant manifold as shown in Figure 6-1. The intent is to have the fertile fuel canister designed with mechanical seals on the hemispherical end caps. This will provide a non-destructive means of extracting the bred fissile fuel as an assembly. The fuel assemblies will have a different length for each specific location. This is due to the decreasing circumferential envelope associated with the torus geometry as the major torus radius is decreased. The fertile fuel canister is illustrated in Figures 6-1 and 6-2, in which a two tube pitch arrangement is shown.

Mechanically, the assembled fuel canister manifold section is attached to the Pb-LiH blanket. The Pb-LiH blanket is designed to support the fertile blanket. The support is mechanical and does not constrain the thermal joint growth due to the elevated temperature operation of the blanket zone. This attachment scheme is illustrated in Figure 6-3. This allows replacement of the more highly irradiated fertile fuel zone and reuse of the reflector/fusile fuel zone. To remove the fertile/fissile blanket, it is rotated off the Pb-LiH support structure.

TABLE 6-1

UC BLANKET CROSS SECTION MATERIAL VOLUME FRACTIONS

Zone 1. Fertile Fuel Zone: (Average percentages based on a radial canister cross section not including end caps and tube sheets*)

45% UC Fuel

33% He Coolant

22% 316 SS Structure

Zone 2. Coolant Manifold Zone: 6% Stainless Steel - 94% void He coolant
(The coolant manifold zone varied in percentage of steel in the poloidal direction due to the termination of manifolds poloidally)

Section A - All 6 Headers	- 9% 316 SS
B - 4 Headers	- 6% 316 SS
C - 2 Headers	- 3% 316 SS
mean	- 6% 316 SS

Zone 3. Reflector Zone

77% Pb

12% SS

11% He

Zone 4. Fusile Fuel Blanket

77% LiH

12% SS

11% He

*Lost fuel in rounded corners must be analyzed and blanket coverage reduced appropriately.

Figure 6-1. CTHR Helium Cooled Uranium Carbide Blanket Concept.

Figure 6-2. CTHR UC Fertile Fuel Canister.

Figure 6-3. CTHR Fertile Fuel Assembly Attachment Concept.

When the complete blanket is assembled the cross section at the upper and lower section of the blanket module is illustrated in Figure 6-4. Due to the low power densities of the Pb and LiH zones, the poloidal flow path can be the entire sector length. The maximum temperature for the Pb section is 300°C , which is below the melting temperature of the material. The LiH zone can be operated somewhat hotter, but is still cooled well below the structural material temperature limit of 500°C . The coolant pressure tubes passing through the Pb and LiH zones will be the only primary pressure boundary, since the outer containing boundary serves as a clad, rather than a pressure boundary.

The outside modules have a radius of curvature of 3.1 m and associated poloidal length of 8 m. Approximately 70% of the neutron population is intercepted by the outer blanket modules, as was illustrated in Figure 3-2. The inner blanket modules are similar in cross-section design. However, this section has less curvature and intercepts an additional 28% of the plasma neutrons. Thus, the total poloidal blanket wall coverage potential with the full blanket is 98%.

The blanket module is designed so that maintenance and replacement can be accommodated by remote operations. The sectional geometry allows individual sectors to be removed by simple linear motions to withdraw it from between two TF coils. There are three blanket sectors between each pair of TF coils. Thus, the center outside sector can be directly removed, while the two adjacent outer sectors have to be circumferentially rotated to the unobstructed center area between TF coils to be removed. Figure 6-1 shows the TF coil side of the blanket sector with the remote handling attachments on the outer blanket structure. These two attachments allow remote handling of the complete blanket sector.

The UC/LiH blanket has a number of advantages and attractive features as follows:

- 1) The fertile blanket zone has the versatility of being readily detached from the fusile blanket and support structure.

Figure 6-4. CTHR Helium Cooled Uranium Carbide Blanket Cross Section.

- 2) The internal fuel assemblies are fabricated as a single unit which slides into the containing canister. Thus, to refuel the assembly, the hemispherical end cap is removed, the irradiated fuel assembly removed, and a new assembly inserted.
- 3) The potential exists for integrating the blanket module and first wall into an integral structure.
- 4) The potential of designing the LiH fusile blanket zone for in-situ tritium recovery can be achieved via use of alternate structural materials than used in the fertile blanket.

A disadvantage of this design is its mechanical complexity, and, if disks are used, the UC is formed in a special way so that fuel rotation would be difficult. To fully evaluate the proposed blanket concept it is necessary to continue the design effort with in-depth structural and thermal-hydraulic analysis and serviceability to determine if this design is viable. Based on the present level of analysis, the concept appears to be attractive.

6.2 UO_2/LiH OR Li_2O BOILING WATER COOLED BLANKET MODULE DESIGN

The basic blanket design is similar in cross section to that of the DTHR blanket design⁽⁴⁾. The structural material is 316 Stainless Steel. UO_2 fuel clad with Zircaloy-4 is the fertile zone fuel. The reflector zone is helium cooled Pb using stainless tubes passing through the Pb medium. The fusile zone is composed of LiH or Li_2O as the tritium breeding fuel, helium cooled by stainless steel tubes passing through the medium. Clad rods could be used with a chambered coolant housing as an alternate design with some sacrifice in fuel volume fraction. Scoping neutronic calculations discussed in section 4.0 were performed to assess the blanket material combinations to determine the volume fractions necessary to achieve the projected blanket performance. These identified volume fractions were the basis for the reference blanket arrangement.

The blanket concept cross section is illustrated in Figure 6-5. The rear section is rectangular in cross section containing the reflector, Pb, and

Figure 6-5. CTHR Dual Control UO₂ Blanket Concept Cross Section.

fusile breeding material. The front section of the module is a split cylindrical pressure tube containing vertical fuel rods 2.5 m long, cooled by two-phase boiling water in an axial upward flow scheme. The reflector and fusile blanket zones are cooled axially with upward flowing helium. These individual zones consist of a block of Pb and LiH respectively with stainless steel coolant tubes passing through the materials. The Pb has a maximum temperature of 300°C, which is below the melting temperature of the material. The Li₂O concept could use clad rods, which would permit in-situ tritium removal via the He if vented rods are used.

Since the Pb zone operates at a lower temperature than the fertile blanket section, the two zones are separated by a double stainless steel wall. The two walls are rib stiffened and sized to withstand the coolant pressure acting across the surface of the wall. Insulating materials may be required to fill the space between the two walls. The Pb and fusile fuel zones are contained in a stainless steel housing. Since the coolant tubes were assumed to be the pressure boundary for the coolant, the fusile zone stainless steel container (clad) was not sized to be a primary pressure boundary.

The Li and Pb zones can be cooled by helium or water, subject to a safety analysis of its compatibility with lithium compounds. A schematic diagram for the design using helium as the coolant is shown in Figure 6-6. Preheated water (from the vacuum vessel, shield, or other sources) enters from the bottom plenum and passes through an orifice plate that lowers the coolant operating pressure so that the coolant flashes to a low quality mixture. The wet steam then passes axially through the fertile blanket zone and exits as a wet steam having a quality between 30 and 40%. Due to the efficient coolant properties of boiling water, the fuel rods can be effectively cooled in a relatively tight lattice. The proposed fuel rod lattice is a triangular array with P/D of 1.1.

Four individual modules are bundled together toroidally with three bundled vertical sets secured together by a strongback structural framework as shown in Figure 6-7. This outer blanket occupies the poloidal region where approximately 70% of the neutron population is intercepted. These integral

Figure 6-6. CTHR Dual Coolant Blanket Concept.

Figure 6-7. CTHR Dual Coolant Blanket Sector and Bundle Concept.

module sections are arranged such that they can be installed and removed from the reactor as a unit and by remote operation. The strongback holding the individual modules together will have a built-in lifting member for handling the complete module sector. The module length is slightly shorter for the three vertical sets of modules bundled together to occupy the inner blanket region. The bundled sector has less curvature than the outer blanket sector. Typical blanket material volume fractions are shown on Table 6-2.

An alternate concept using boiling water coolant for the entire blanket was considered. However, the compatibility of LiH or Li_2O with boiling water coolant must be studied further to assure the safety of these systems. The concept is shown in Figures 6-8 and 6-9. The water is preheated by the tritium breeding blanket instead of the vacuum vessel cooling system or the shielding cooling system to reach saturation temperature before it enters the fertile blanket, flashing to steam at the orifice plate in the reversing plenum. The fertile region requires conventional technology with regard to fuel and fuel form for both coolant concepts. The fusile and reflector zones are not state-of-the-art technology; however, they do not appear to involve unreasonable development schemes. The refueling requires individual sector and module replacement, which could become an economic factor affecting plant availability.

TABLE 6-2

UO₂ BLANKET CROSS SECTION MATERIAL VOLUME FRACTIONS

Zone 1: Fertile Fuel Zone

45% UO₂

8% Zircaloy Clad

11% 316 SS

1.6% Helium Gap

34.4% Boiling Water Coolant

Zone 2: Reflector Zone

77% Pb

12% SS

11% Helium Coolant*

Zone 3: Fusile Fuel Zone

77% LiH or Li₂O

12% SS

11% Helium Coolant*

*Boiling water alternate coolant pending safety studies

Figure 6-8. CTHR Wet Steam Cooled UO_2 Blanket Concept Coolant Flow Path.

Figure 6-9. CTHR Wet Steam Cooled UO_2 Blanket Sector Concept.

7.0 THERMAL-HYDRAULIC DESIGN AND ANALYSIS

As discussed in the preceding sections, two blanket design concepts have been studied for the CTHR. One consists of UC as the fertile fuel which is cooled by helium gas and the other consists of UO_2 fuel rods which are cooled by boiling water. In order to provide sufficient power conversion capability for commercial applications and still maintain a low pumping power requirement, high pressured coolant was utilized. In the helium-cooled concept the helium flows inside circular stainless steel pressure tubes with the carbide fuel filling the space between the tubes. Each module consists of 96 tubes which are contained inside a double-wall cylinder. The outer wall cylinders are cooled by helium through the annulus. The cylinder is oriented along the toroidal direction with the inlet and outlet headers running along the poloidal direction. For the boiling water-cooled concept the UO_2 pellets are clad with zircaloy tubes. The rod assembly is arranged in a tight triangular array with the water flowing along the length of the rods. The rods and water are contained inside a semi-circular cylindrical vessel. The lead neutron reflector/multiplier and the fusile fuel (tritium breeding) zones are placed behind the fertile fuel zone. The fusile zones are cooled by helium. The thermal-hydraulic design and analysis of these blanket concepts are presented in this section.

7.1 HELIUM-COOLED UC BLANKET

The advantage of having the coolant flowing inside the pressure tubes is that if the thermal conductivity of the fuel material is high, the tubes can be placed far apart without resulting in a high temperature difference between the tube wall and the peak fuel temperatures. This also results in a high fuel volume fraction if the local nuclear heating rate is not extremely high.

For hybrid blanket applications, the stainless steel tube wall (clad) temperature is limited to 550°C for stress corrosion and fatigue considerations. This temperature limit is much lower than the fuel material temperature limit

(approximately half of its melting temperature). The thermal design objective is therefore, to establish the tube size, length of channels and tubes pitch to tube outer diameter ratio (P/D), so that the tube wall temperature limit is not exceeded and the pumping power requirement is minimized.

A parametric study was performed using a steady state, coupled heat conduction and hydraulic calculation. The tube wall and peak fuel (in triangular array) temperatures and the coolant pressure and temperature distributions along a channel of a given length were computed at various heat generation rates and P/D ratios. Preliminary calculations indicated that a tube OD of 1.0 cm and a channel length of 1.5 m are about the optimum values. These values were used in the conceptual design.

Two helium coolant inlet conditions were studied: (1) $P_{in} = 8.62$ MPa (1250 psia) and $T_{in} = 280^{\circ}\text{C}$ and (2) $P_{in} = 6.55$ MPa (950 psia) and $T_{in} = 260^{\circ}\text{C}$. Since the coolant outlet temperature is limited by the maximum allowable pressure containment tube wall temperature, the coolant inlet temperatures were selected to be as high as practical, so that an efficient power conversion system can be designed. The selection of two coolant system pressures was to investigate the effect of two coolant pressures on the pumping power requirement and the attainable steam pressure for the indirect helium-to-steam power conversion system. The calculated results for the two cases are shown in Figure 7-1 and 7-2. In the figures the pumping power requirement (assumed to be twice the value due to the frictional pressure loss in the tube to account for the entrance and exit losses and the other losses in the circuit) and the channel exit temperature are plotted versus the P/D ratios at various heat generation rates. The curves are at the steady state condition with the tube mid-wall temperature operating at the maximum allowable limit by proper inlet coolant flow orificing. For example, as shown in Figure 7-1, at $Q = 500 \text{ W/cm}^3$ and $P/D = 1.2$, the required pumping power is 3% and the channel exit temperature is about 475°C . Since the radial distribution of the nuclear heating rate decreases rapidly across the fertile blanket thickness, from about 400 W/cm^3 to 150 W/cm^3 over a 16 cm thickness with a 50% fuel volume fraction as discussed in Section 5.4 and Figure 5-8, the tube spacing can be varied radially with each tube being

COOLANT TUBE OD = 1.0 cm (0.04 cm WALL)
 CHANNEL LENGTH = 1.5m
 OPERATING AT MAX. TUBE WALL TEMP. OF 550°C

Figure 7-1. UC Blanket Module Thermal Performance as a Function of Coolant Tube Spacing and Nuclear Heating Rate. Helium P_{in} = 8.62 MPa (1250 psia), T_{in} = 280°C.

COOLANT TUBE OD = 1.0 cm (0.04 cm WALL)
 CHANNEL LENGTH = 1.5m
 OPERATING AT MAX. TUBE WALL TEMP. OF 550°C

Figure 7-2. UC Blanket Module Thermal Performance as a Function of Coolant Tube spacing and Nuclear Heating Rate. Helium $P_{in} = 6.55$ MPa (950 psia), $T_{in} = 260^\circ\text{C}$.

orificed at the inlet, so that the tubes' pressure drops in a cylindrical module are the same and the peak tube wall temperature in the module is below the allowable limit.

Preliminary calculations indicated that the helium mixed mean temperature during the plasma-on time is about 478°C for the high pressure case (Case 1 above) and 475°C for the low pressure case (Case 2). The pumping power requirements are about 2.0 and 2.5% of the blanket thermal power, respectively. The design tube spacing in a module should vary, therefore, from $P/D = 1.2$ to 1.8, depending on the local heat generation rate. In this preliminary design, two zones were considered: the plasma side zone at $P/D = 1.2$ and the outer zone at $P/D = 1.6$.

The solid fuel volume fraction (excluding the space occupied by the coolant tubes) in a unit triangular array at the various P/D ratios is shown in Figure 7-3. With a tube OD to the tube wall thickness ratio (α) equal to 25 for this particular canister design, the unit cell fuel volume fraction is 0.53 at $P/D = 1.2$ and 0.70 at $P/D = 1.6$. The average value would be much higher than 0.50. However, when the other structural material and cooling annulus of the canister are included, the overall fuel volume fraction is about 0.45.

The small difference in pumping power requirements between the two cases having different helium pressures is due to a higher coolant temperature rise selected for the lower pressure case. The net effect is a lower steam pressure that can be obtained from the helium/water steam generator for power conversion.

Since the plasma duty cycle for the CTHR is 90%, with a proper design of a thermal energy buffer-storage system, the blanket coolant exit temperature averaged over one completed burn cycle can be assumed to be 95% of the value during the plasma-on period. The time averaged blanket exit temperature is therefore about 450°C . This temperature was used as the design value for the power conversion system.

With the helium gas operating between 450°C and 280°C for the high pressure case and between 450°C and 260°C for the low pressure case, the steam generator

Figure 7-3. UC Fuel Volume Fraction as a Function of Tubes Pitch to Tube OD Ratio at a Triangular Tubes Array.

characteristics for the two cases are as shown in Figure 7-4. In the figure the water/steam temperature is plotted against water enthalpy showing the regions of the economizer, evaporator and superheater. The helium temperature line in the primary side of the steam generator is also shown in the figure. In this study the pinch point temperature difference was taken to be 26°C (50°F) and the feedwater inlet temperature to the steam generator was taken to be 150°C (300°F). For the high helium pressure case with a helium exit temperature of 280°C from the steam generator, a steam pressure of 8.62 MPa (1250 psia) can be achieved. On the other hand for the low pressure case with a helium exit temperature of 260°C , the steam pressure can only be about 5.86 MPa (850 psia). The thermal efficiency of the steam turbine-generator cycle depends on the number of feedwater heaters used and the stage efficiencies of the turbines. Assuming there are six feedwater heaters and an 80% stage efficiency, the gross thermal efficiency for the cycle with the steam pressure of 8.62 MPa (1250 psia) is 37% and that with the steam pressure of 5.86 MPa (850 psia) is 35%. The corresponding net cycle thermal efficiencies were estimated to be 35% and 32%, respectively. The blanket-turbine operating conditions are summarized in Table 7-1.

The peak fuel temperatures at the various heating rates and pitch-to-diameter ratios are shown in Figure 7-5. The effect of the helium operating conditions on the fuel temperature is small. The results for the case with the higher helium pressure are presented. The peak fuel temperature distribution in the blanket is also shown in the figure. The maximum fuel temperature is about 660°C .

7.2 BOILING WATER - COOLED UO_2 BLANKET CONCEPT

In a water cooled blanket the mass density of the water has a profound effect on the neutronic performance of the blanket. Dry steam cooling was considered, but had high pumping power (over 50% of the blanket thermal power due to low operating temperature limit for the Zr clad material) and there was difficulty in producing high pressure steam outside the blanket, in spite of the low density of the steam. High quality mixture is also not desirable because of the problems in flow stability and mixture dryout at high heat

Figure 7-4. Steam Generator Characteristics for the Helium/Steam Indirect Power Conversion System.

TABLE 7-1
BLANKET AND TURBINE OPERATING CONDITIONS

BLANKET PARAMETERS

Fertile Fuel	UO ₂	UC	
Max. Fuel Temp., °C	810	660	
Clad Material	Zr-4	316 SST	
Max. Clad Mat'l. Temp., °C	325	550	
Structural Material	316 SST	316 SST	
Coolant	2 Phase Water ^(A)	He	
Avg. Coolant Inlet Pressure psia	1000	950	1250
Avg. Coolant Outlet Temp., °C	280	450 ^(B)	450 ^(B)
ΔT Through Blanket, °C	---	215	198
Max. Coolant Channel Length, m	2.5	1.5	1.5
Pumping Power To Thermal Power Ratio, %	0.5	2.5	2.0
Average Thermal Power, MW _t	4000-6000	5000-7000	

TURBINE PARAMETERS

Steam Cycle	Direct	Indirect	
Pressure, psia	900 ^(C)	850 ^(C)	1250 ^(C)
Temperature, °C (Superheated °C)	278 (0)	395 (120)	420 (120)
Feed Water Temperature, °C	175	150	150
Size, MW _e	800	1000	1000
No. of Units in Parallel	2 ^(D)	2 ^(D)	2 ^(D)
Gross Thermal Efficiency, %	34	35	37
Estimated Net Cycle Efficiency, %	33	32	35

(A) 3.5% Inlet Quality, ~35% Exit Quality For Flow Stability.

(B) Limit Due to Low Film Coefficient of Coolant and Structural Material Limitations. Effect of Plasma-Off Included.

(C) Steam Pressure at Steam Separation Exit for Direct Cycle or at SG-Superheater Exit for Indirect Cycle.

(D) Tandem Compound Units With 6 Feedwater Heaters and 80% Stage Efficiency.

COOLANT TUBE OD = 1.0 cm (.04 cm WALL)
 CHANNEL LENGTH = 1.5 m
 OPERATING AT MAX. TUBE WALL TEMP. OF 550°C

Figure 7-5. UC Blanket Peak Fuel Temperature Distribution.

fluxes. The critical heat flux for two-phase water flow decreases rapidly for a channel exit quality over 40-50%. The coolant exit quality of a typical boiling water reactor fuel channel is about 40%. Test data^(14,15) showed that at a pressure of 6.89 MPa (1000 psia) and a channel length to diameter ratio (L/D) of 150, the critical heat flux is about 120-150 W/cm², when the channel exit quality is ~35%. The critical heat flux is slightly higher at the same exit quality, if the coolant is two-phase at the channel inlet other than subcooled. Therefore, the thermal design objective using boiling water cooling is to establish an operating condition so that (1) the mixture exit quality is below 40%; (2) the mass weighted coolant average density in the blanket is as low as practical; (3) the clad temperature is within the allowable limit of 320°C for the zircaloy and (4) the coolant pressure is as high as practical so that a useful power conversion system for commercial application can be utilized. Due to these conflicting requirements the neutronic performance of the blanket would have to be compromised due to the high volume fraction of structural material needed to support the high coolant pressure and the less than ideal density of a low quality water mixture.

Two coolant inlet conditions were investigated. For one, the water at the blanket inlet is subcooled. In the other, the water is two-phase with a quality about 3.5% at the blanket inlet. Because of the 320°C temperature limit for the zircaloy clad, the saturation temperature of the cooling water must be somewhat below this temperature to account for the clad-water film temperature drop. The highest water pressure that can be used would be about 6.89 MPa (1000 psia). This pressure was then selected as the reference system pressure at the blanket inlet. The thermal performance of the cooling system at the fertile blanket region for the two inlet conditions mentioned above are compared as shown in Table 7-2. In each inlet condition, two heat generation rates were analyzed: 300 W/cm³ representing a peak power rod in the blanket and 200 W/cm³ representing an average region of the blanket.

As seen from Table 7-2, the system pressure drop, fuel and clad temperatures and the pumping power requirement are comparable for the two different inlet conditions. The only significant different is the mass weighted coolant

TABLE 7-2

THERMAL PERFORMANCE OF THE UO_2
BLANKET COOLED BY BOILING WATER
(Rod OD = 1.45 cm, P/D = 1.10, Zircaloy Clad)

CASE	<u>1</u>	<u>2</u>	<u>3</u>	<u>4</u>
Water Condition @ Channel Inlet	<u>Subcooled</u>		<u>Two-Phase</u>	
Heat Generation Rate, Q , W/cm^3	300	200	300	200
Heated Length, m	2.5	2.5	2.5	2.5
Inlet Pressure, MPa (psia)	6.896 (1000)		6.896 (1000)	
Inlet Temperature, $^{\circ}\text{C}$	281	280	285	285
Exit Temperature, $^{\circ}\text{C}$	280	280	280	280
Pressure Drop, MPa (psia)	.04 (58.7)	.04 (58.7)	.042 (61.5)	.042 (61.5)
Peak Heat Flux, W/cm^2	87.6	58.7	87.6	58.7
Peak Clad Temperature, $^{\circ}\text{C}$	320	307	310	302
Peak Fuel Temperature, $^{\circ}\text{C}$	817	618	809	613
Inlet Density, kg/m^3	749.6	749.6	435.4	435.4
Exit Density, kg/m^3	99.8	168.5	83.5	126.7
Mass-Weighted Coolant Average Density, kg/m^3 (lb/ft ³)	372.7 (23.2)	441.4 (27.5)	216.6 (13.5)	252.9 (15.7)
Inlet Quality, Steam Weight %	0.0	0.0	3.5	3.5
Exit Quality, Steam Weight %	30.7	16.0	38.0	23.0
Pumping Power, % of Thermal Power*	0.23	0.44	0.40	0.70
Coolant Mass Flow Velocity, kg/sec-cm^2	0.38	0.50	0.36	0.43

*Twice the Value Through the Heated Length

density in the blanket. With subcooling at the inlet, the coolant density is about twice that with two-phase mixture at the inlet. Since neutronic performance requires low water density, the two-phase inlet condition was selected as the reference condition for this blanket design concept. The coolant quality, coolant, clad and fuel temperatures and density distributions at the reference operating conditions are shown in Figures 7-6 and 7-7, respectively.

The peak heat flux in this blanket concept is about 88 W/cm^2 as shown in Table 7-2. This is well below the critical heat flux of about 120 to 150 W/cm^2 mentioned previously. The operating conditions selected should therefore, satisfy the flow stability and dryout requirements.

In this blanket concept the lead reflector and the fusile breeding zones are behind the fertile blanket region and helium cooled. However, if a safety analysis can justify a water cooled option, the coolant could be arranged to cool the lead and the fusile zones in parallel first before entering the fertile blanket region. The water pressure at the inlet header would be just high enough to account for the pressure drop across the lead and the fusile zones, so that the water would just reach its saturation temperature at the bottom plenum. The water would then flash to a quality of about 3.5% across an orifice plate before entering the fertile blanket at 6.89 MPa (1000 psia). The pressure drop across the orifice plate was estimated to be about 1.0 MPa (150 psia). The coolant pressure at the inlet header would then be on the order of 8 MPa (1160 psia).

For power conversion a direct steam turbine cycle, similar to existing boiling water reactors, could be utilized. A steam separator is required to separate the moisture and to supply steam for the turbine from the blanket. With a blanket inlet pressure of 6.89 MPa (1000 psia) and a blanket pressure drop of 0.42 MPa (61.5 psia), a steam pressure of 6.2 MPa (900 psia) at the steam separator exit could be achieved. This steam supply pressure is within the current nuclear steam turbine technology. With a reheat cycle and sufficiently high feedwater temperature, a net cycle efficiency of 34% could be obtained. The blanket and turbine operating conditions are summarized in Table 7-1.

Figure 7-6. UO₂ Blanket Fuel, Clad, and Coolant Temperatures and Steam Quality Distributions.

Figure 7-7. Coolant Density Distributions in the UO_2 Blanket.

8.0 SUMMARY AND CONCLUSIONS

In this report the results of a study on two generic blanket design concepts for application in a Commercial Tokamak Hybrid Reactor (CTHR) have been presented. Both blankets operate on the U-Pu cycle and were designed to achieve tritium self-sufficiency while maximizing the fissile fuel production within thermal and mechanical design constraints. The two blanket concepts that were evaluated were: 1) a UC fueled, stainless steel clad and structure, helium cooled blanket (UC-SS-He); and 2) a UO_2 fueled, zircaloy clad, stainless steel structure, boiling water cooled blanket (UO_2 -Zr-BW). The motivation for investigating the UO_2 -Zr-BW blanket was that it utilized many features of proven light water fission reactor technology and also was similar to a concept studied previously^{2,4}. The UC-SS-He blanket concept offers the potential for improved neutronic performance relative to the UO_2 concept, but at the expense of employing more advanced, less well developed technology and materials. Two different tritium breeding media, Li_2O and LiH , were evaluated for use in both blanket concepts. The use of lead as a neutron multiplier or reflector and graphite as a reflector was also considered for both blankets.

A series of neutronics studies was performed to scope the various material zone thicknesses required for different geometric configurations, with the constraint that the arrangement be self-sufficient in breeding tritium. The neutronic performance was calculated for a sufficient irradiation exposure time to achieve about 3% fissile fuel enrichment in the blanket. An important aspect of the analysis was to investigate the neutronic benefits of full blanket wall coverage in a tokamak fusion hybrid device. Due to accessibility, maintenance, and blanket refueling considerations in a tokamak, it may be desirable to restrict the blanket on the inward portion of the torus to lithium bearing regions only, and use fertile fueled zones only on the outer portions.

The results of the comparison of full (inner and outer) fertile blanket coverage versus outer fertile blanket coverage only indicate that increases in

fissile fuel production of up to about 20% would be possible with the full blanket coverage. The difference occurs because of increased neutron multiplication with the inner fertile zone, but is somewhat smaller than might be expected because of the non-uniform tokamak neutron source distribution which results in about 70% of the D-T fusion neutrons being directed toward the outer blanket region. Also, for an outer fertile blanket only, the inner region can be designed with a material such as lead which provides some additional neutron multiplication. Another factor in considering the benefits of fertile material in the inner region of a tokamak hybrid is a possible decrease in the plant availability due to increased reactor downtime to service and refuel the inner blanket. If this increased plant outage becomes too large, the neutronic benefit of using full blanket coverage will be negated.

Other conclusions from the neutronic analyses indicated certain advantages in the use of LiH as compared to Li_2O for the tritium breeding material. A blanket using LiH can be thinner than one using Li_2O due to the excellent moderating properties of the hydrogen. Also, the same moderating effect with the LiH decreases parasitic absorption in the structure in the lithium material zone compared to the use of Li_2O . An investigation of the use of lead in the blanket indicated that significant benefits could result by using it both as a reflector behind the fertile zone and as a neutron multiplier in the inner region for the case of outer fertile blanket coverage only.

For each of the blankets a conceptual mechanical design was developed and the thermal-hydraulic performance was evaluated. High pressure coolants were used in the blankets to obtain high power conversion efficiency and low pumping power requirement. The steam pressure at the turbine throttle valve for the UO_2 blanket is about 6.20 MPa (900 psia) and that for the UC blanket is about 8.62 MPa (1250 psia). With these pressures the present day nuclear turbine technology for economical commercial applications can be utilized. For the axial flow boiling water cooled blanket the water density varies along the length of the blanket fuel rods. This causes nonuniform axial fissile production. One remedy is to reverse the fuel section periodically. Due to the hydrodynamic limitation, the water mixture quality must be kept below

40%. Studies on the variation of the water coolant density indicate decreased fissile fuel production and increased thermal power production with increasing water density. For both coolant concepts, it was determined that pumping power limitations restrict the length of flow in the fertile fueled front zone of the modules to 1.5 m for the UC-He concept and 2.5 m for the UO₂-boiling water concept.

A list of important performance parameters comparing two blanket concepts with full blanket coverage and lead reflector regions is presented in Table 8-1. The values given are the average over a four year irradiation period, or 4.8 MW-yrs-m⁻² of exposure, the approximate time to reach three percent enrichment. Both are self-sufficient in tritium, and produce about the same thermal power. However, the UC, He cooled blanket produces about 75% more fissile ²³⁹Pu fuel due to both the increased uranium density with the UC compared to the UO₂ and also because of the harder neutron spectrum with the helium compared to the boiling water. In both blankets the power production increases by approximately 70% over the four year irradiation cycle. Means of levelizing the thermal power require further study once a reference concept is selected.

The determination as to which of these blankets is most desirable for use in the CTHR will be decided mainly on economics, i.e., minimizing the cost of electricity for the total system of the hybrid plus the fission reactors it supports. A companion study to this work that deals with these economics questions is reported in Reference 16. In addition, material lifetime and radiation damage considerations may be an important factor. While most of the analyses in this report were for a four year irradiation period, sufficient to reach ~3% enrichment in the blanket, economics studies will be required to determine the optimum exposure time. It should be noted that the results presented in this report represent an initial assessment. Further iterations between neutronic, thermal-hydraulic, structural, and mechanical design are required to arrive at an optimum design configuration. Additional considerations such as fuel shuffling and operation on an equilibrium cycle will be important. Although not addressed in this report, such blanket fuel management studies will be a key area for future work on the CTHR blanket design. Overall, both of these blanket concepts appear feasible and attractive for use in a tokamak fusion hybrid reactor.

TABLE 8-1

SUMMARY OF IMPORTANT PERFORMANCE PARAMETERS FOR
TWO BLANKET CONCEPTS WITH FULL BLANKET WALL COVERAGE

	<u>UO₂ Fueled^(a)</u>	<u>UC Fueled</u>
Fertile Material	UO ₂	UC
Clad Material	Zr-4	SS-316
Structural Material	SS-316 ^(b)	SS-316
Coolant	Boiling Water	Helium
T-Breeding Material	Li ₂ O	LiH
Reflector Material	Pb	Pb
T, Tritium Breeding Ratio ^(c)	1.13	1.20
F, $\frac{\text{Net } ^{239}\text{Pu Atoms}}{\text{Fusion Neutron}}^{\text{(c)}}$	0.54	0.97
M, Blanket Energy Mult. ^(c)	7.5	7.9
P _f , Fusion Power, MW	1200	1200
Plant Availability x Plasma Duty Cycle	0.6	0.6
Wall Load at Pulse, MW m ⁻²	2.0	2.0
Fissile Production, $\frac{\text{kg } ^{239}\text{Pu}}{\text{yr}}^{\text{(c)}}$	1524	2730
Thermal Power, MWt ^(c)	5840	6100
Fissile Enrich. After Four Years, %	2.96	2.89
Blanket Inventory, tonnes U + Pu	255	410
Max. Fuel Temperature, °C	810	660
Max. Clad Temperature, °C	325	550
Max. Pressure Vessel Temp., °C	500	500
Coolant Inlet Pressure, psia	1000	1250
Coolant Outlet Temperature, °C	280	450
Power Conversion Turbine Inlet Pressure, psia	900	1250
Gross Power Output, MWe ^(c)	2082	2257

(a) Values for case with constant 30% water quality.

(b) Although Zr was assumed in the neutronic analysis for simplicity.

(c) Average over four year irradiation time.

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