A MODULAR FISSION-FUSION HYBRID BLANKET

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Abstract

This paper describes a preliminary helium-cooled hybrid blanket of modular design proposed for a 1000-MW(e) Yin-Yang confinement mirror reactor. The modules are triangular prisms; the first wall of the assembled blanket is similar in shape to a geodesic sphere. The stainless-steel shell of each module encloses five helium-coolant tubes. The lithium within the modules is canned in niobium to reduce the handling problems and minimize the consequences of a helium leak. Tritium is recovered by permeation into evacuated niobium tubes. Liquid lithium is the heat-transfer coupler from the cans to the helium coolant tubes.

Three fuel-region designs are proposed. Design A uses clad, solid-fuel wedges of uranium monocarbide or uranium mononitride held in place against the coolant tubes by tungsten springs; the wedges have a cover gas to enhance the

heat transfer. Designs B and C employ spherical fuel pellets in niobium cans similar to those in the high-lithium region, with lithium acting as the heat-transfer agent to the coolant tubes. The Design B pellets are clad; the Design C pellets are not. Cladding adds about 10 vol% of neutronically unproductive material to the fuel region. Preliminary heat-transfer analyses for a 1-MW/m^2 first-wall flux indicate that the design is limited chiefly by the ability of the helium coolant to remove the heat conducted through the pipe walls in the fuel region.

Approximately 9450 modules are needed for the proposed design. The calculated weight per module is about 386 kg, of which more than 90% is the weight of the uranium fuel. Because the fuel burnup rate is low, refueling probably would not be required oftener than once every five years. A refueling system is proposed that would permit the replacement of single modules.

Introduction

Once the scientific feasibility of CTR has been demonstrated, the next logical step to commercial fusion power is a demonstration fusion reactor. Although scientific feasibility may demonstrate an adequate knowledge of plasma physics to operate a fusion reactor, a demonstration reactor undoubtedly will be needed

for engineering tests of components whose designs, according to current analysis, will reach or exceed the present limits of technological expertise.

In this direction, LLL has been studying hybrid blanket designs—fission-fusion hybrids in which a neutron-emitting fusion source drives a subcritical fission assembly.

The inclusion of fission material can increase the energy-multiplication capability of the blanket from between 1 and 2 to between 10 and 20.¹ Neutron leakage from the blanket may be substantially reduced.² The hybrid may significantly reduce the first-wall energy loading or ease the requirements on the fusion plasma by as much as a tenfold reduction in the Lawson criterion for significant thermonuclear burn.³ Unlike a thermal-fission-lattice design being studied at Battelle-Pacific Northwest Laboratory,⁴ in which the energy and neutron multiplication derive largely from thermal emission of slightly enriched uranium, our designs depend primarily on fast fission of natural or depleted uranium for increased energy multiplication.

Our study has sought to develop a workable set of parameters leading to a totally consistent hybrid reactor design. The constraints have been as follows:

- Reactor geometry is based on a Yin-Yang mirror containment of the same physical dimensions as reported at the Austin fusion conference.⁵
- The power output ranges from 200-2000 MW(e), with a 1000-MW(e) reactor as the objective.
- A helium thermodynamic cycle is used, with a helium exit temperature appropriate for 55% thermal efficiency.
- Ion energy is not directly converted.
- The first-wall radius is 4.8 m.
- Basically, the blanket structure is stainless steel; other materials are permitted if necessary.
- Tritium holdup in the blanket is limited to about 1 kg per 1000 MW(e).

- The tritium breeding ratio is at least 1.10.

- Periodic fuel processing is assumed.

We recognize that the development of an acceptable blanket design will require many iterations and tradeoffs. This report is a first attempt to derive a consistent, workable set of parameters. It should be emphasized that this is a single-point analysis; while reasonable choices among parameters were sought, the choices necessarily were somewhat arbitrary.

The use of the Yin-Yang magnet coil results in an ellipsoidal, nearly spherical plasma; thus a spherical first wall within the magnet coils can be assumed for this analysis. Surrounding the magnets are two joined spherical shells (see Fig. 1) that carry the forces from the faces of the magnets. Each separable, hemispherical steel shell contains one of the two magnet coils, a neutron shield, and half the blanket and coolant manifold. This arrangement allows direct access to the first wall; it also allows the construction of two identical components (one section the same as the other but rotated 90° in its mounting). The coolant enters and leaves through the coolant ports.

For our coolant, we have chosen helium, which has the special virtue of being flowable across magnetic field lines. Other relevant characteristics — both favorable and unfavorable — are as follows:

- Tritium removal is not difficult if some oxygen impurity is allowed in the helium.⁷

Fig. 1. Magnet and structural support vessel.⁶

- Coatings applied to the coolant tubing should be relatively stable because of helium's chemical inertness.

- Helium is essentially a neutronic void.⁸

- Helium is optically clear, allowing direct visual inspection of the interior of the coolant piping.

- Because helium-4 is generated from D-T fusion reactions, this source (slightly contaminated by tritium) could be used as makeup.⁹

- Mechanical seals are feasible in the coolant system.

- High pressure (higher than 30 atm) is necessary for acceptable heat transfer.⁸

- High helium flow rates are required for good heat transfer (this could result in unacceptably large pumping-power losses).

- A direct helium gas-turbine cycle might greatly reduce tritium losses through the energy-conversion equipment.⁷

- The tubing thickness required to contain the high-pressure helium adds neutronically detrimental material to the blanket.

- The volume occupied by the helium requires an increase in the total blanket volume over that which would be required

if all the materials were neutronically productive.

- A helium-tube rupture could cause a severe pressure transient in the blanket.

- A small leak in a helium tube could cause a large void in the lithium due to lithium's low mass.

Modular Design

Traditional blanket designs usually have required the coolant to flow in paths parallel to the plane of the first wall. In our design, the flow of the helium coolant is in U-tubes aligned normal to the plane of the first wall. Uniform cooling of the first wall is achieved, and each coolant tube experiences an equal thermal loading.

A cross section of the blanket — following the zoning of fissionable material first suggested by Lee¹ — is shown later in this report. As soon as a radial cooling scheme is chosen, the possibility of blanket modularization becomes obvious. For the individual modules, we selected the triangular prism shape shown in Fig. 2; these are assembled into a geodesic sphere* presenting a nearly spherical first wall (see Fig. 3). We call this assembly a geodesic blanket. All modules are identical, and the centerline of each module points directly to the center of the plasma.

Flowing lithium tends to leach the nickel from austenitic stainless steels (such as 304 SS and 309 SS), and opera-

tion may not be feasible at temperatures greater than about 975 K.¹⁰ Hence, we assume stagnant lithium for compatibility with stainless steel and can it in sealed containers to reduce the liquid-metal handling problems. Canning also reduces the probability of lithium blowout from the blanket in case of a helium leak.

Because the lithium is canned, tritium recovery by permeation is indicated. Minimizing the tritium holdup precludes stainless steel as the permeation material (stainless steels have permeation rates about a thousand times less than

Fig. 2. Exploded view of a blanket module.

* We once considered an icosahedral plasma chamber, with each face of the icosahedron composed of many triangular prism modules. However, we abandoned this in favor of the geodesic sphere design.

Fig. 3. Geodesic blanket assembly.

niobium). The alloy Nb-1%Zr is commonly suggested for a permeation material in contact with lithium. Our design calls for tritium recovery by permeation into evacuated small-diameter niobium tubes that penetrate the lithium cans.

A 1.5-cm coolant tube and a 5.07-MPa (50-atm) helium pressure were selected on the basis of information in Ref. 8. The required tube-wall thickness is based on the simple hoop stress in the tube:

$$t = \frac{DP}{2\sigma}, \quad (1)$$

where t is the coolant-tube wall thickness, D is the inside diameter of the coolant tube, P is the helium pressure, and σ is the stress in the pipe wall. The maximum allowable tube-wall temperature is

taken to be 900 K. The structural material is 304 stainless steel (although 309 stainless steel might be more compatible with lithium¹⁰). The 0.00001% creep rate stress for 304 stainless steel at 922 K is 44.79 MPa (6500 psi).¹¹ To allow for stress concentrations and provide a safety margin, a tube-wall thickness of 0.1 cm is assumed; the resulting hoop stress is about 34.5 MPa (5000 psi).

The blanket composition near the first wall is assumed to be similar to that described in Ref. 5; that is, the first wall is 0.1 cm of stainless steel followed by 0.9 cm of lithium. The helium coolant U-tubes approach to within 1 cm of the face of the first wall. Following the data in Ref. 5, the maximum energy deposition in the lithium is taken to be 3.6 W/cm³; the spacing of the coolant tubes is

estimated by analysis of a unit cell shown in Fig. 4.

The following equation describes conductive heat transfer in which R corresponds to the radius of the circle circumscribed about the center of an equilateral triangle whose vertices are located at the center of the coolant tubes:

$$T_R - T_0 = (T_R - T_p) + (T_p - T_i) + (T_i - T_0)$$

$$= q \left\{ \frac{R^2}{4k_f} \left[\left(\frac{r_p}{R} \right)^2 - 2 \ln \frac{r_p}{R} - 1 \right] \right.$$

$$+ \frac{r_p^2}{4k_f} \left[\left(\frac{r_i}{r_p} \right)^2 - 2 \ln \frac{r_i}{r_p} - 1 \right]$$

$$\left. + \frac{R^2 - r_i^2}{2r_i h} \right\}, \quad (2)$$

where

T_R = temperature at unit cell boundary,

T_p = temperature at outer surface of tube,

T_i = temperature at inner surface of tube,

T_0 = helium coolant temperature,

r_i = inner radius of tube,

r_p = outer radius of tube,

R = radius of unit cell,

k_p = thermal conductivity of tube wall,

k_f = thermal conductivity of region R to r_p ,

q = energy deposition rate.

Using only the first terms of this equation and assuming a lithium k_f of 585 W/m·K (from Ref. 10), one finds that $R = 3.4$ cm results in a $T_R - T_i$ value of 307 K.

Within the limitation of a triangular module cross section, many different coolant-pipe arrangements are possible, the simpler ones containing 3, 5, 14, 18, or 33 U-tubes. A five-tube arrangement is chosen as a reasonable tradeoff between the required number of modules and a manageable module size. The result is a module having an edge length of 26.6 cm at the first wall surface and a first-wall surface area of 306.38 cm². For the 4.8-m-radius plasma chamber, approximately 9450 modules are needed.

Using the unit-cell heat-transfer analysis, we can determine the relative severity of the first-wall heating. A 0.1-cm steel shell is added at the unit's boundary; R is taken to be 4 cm. A wall

Fig. 4. Heat transfer analysis unit cell.

loading of 1 MW/m^2 is assumed. By linearly scaling the Ref. 5 data, the total energy-deposition rate in the first wall is determined to be 129 W/cm^3 . Using

Eq. (2), the temperature rise from the inside of the helium tube to the outside of the first wall is 309 K, indicating the feasibility of adequate first-wall cooling.

Lithium Region

As shown in Fig. 5, the interior of each module is packed with cans of non-flowing lithium. Integral to each can is a 0.2-mm-wall niobium U-tube for tritium recovery.

To attain a desirable minimum tritium inventory in a blanket of approximately 1 kg, the tritium concentration (averaged over the volume of the module) is about 15 ppm. If we use the approach suggested by Hickman,¹² the maximum change in concentration as a function of distance from the permeation tube is approximately 3.22 ppm. Assuming a

linear gradient, the peak tritium concentration is about 16.5 ppm, ranging down to 13.2 ppm next to the permeation tubes. Using the tritium-permeation relationship discussed in Refs. 9 and 13 to calculate the required tritium permeation area, the required tube diameter is 0.5 mm for a downstream tritium partial pressure of $10.64 \mu\text{Pa}$ (8×10^{-8} Torr) and an operating temperature of 900 K.

The shape of the lithium cans is such that replacement can be accomplished without removing the helium coolant tubes. The cans contact the module

Fig. 5. Cross section of a blanket module.

walls at regular intervals; their walls form a secondary honeycomb structure. Between the cans and the coolant tubes is a region filled with a liquid-lithium heat-transfer coupler that acts as a heat-transfer medium, provides an escape path for the helium in case of a coolant-tube failure, and allows for thermal expansion of the module and its components. The coupler also permits faulty modules to be identified, through external monitoring of the coupler's apparent "level." A sudden change in the coupler level indicates an internal problem, allowing prompt corrective action.

Lithium has about 20% volumetric expansion between 300 and 1273 K. The liquid coupler should be kept at approximately 400 kPa (50 psi) at operating temperatures to prevent vaporization.

Thermal expansion can be compensated by an expansion bellows comprising part of the canning. On heatup, lithium expansion inside the can compresses the bellows, forcing some of the heat-transfer coupler fluid out of the module; on cooldown, the bellows expands, drawing back the fluid.

Lithium canning is not suggested for the first-wall region primarily because of the complicated shapes required and the desirability of minimizing structure in this region. The helium coolant tubes near the first wall are positioned by a tubesheet that functions also as the second wall. Perforations in this sheet allow lithium to pass from the first-wall coolant region into the heat-transfer coupler passages; they permit passage of the tritium-recovery U-tubes into the first-wall coolant region.

Fuel Region

Potential fissionable fuels for the hybrid blanket are uranium metal, uranium monocarbide (UC), uranium dioxide (UO_2), uranium mononitride (UN), thorium metal, thorium dioxide (ThO_2), plutonium metal, and plutonium dioxide (PuO_2). Selected physical properties of some of these materials are given in Table 1. Thorium metal and thorium dioxide have lower energy multiplications than other fissionable fuels,^{1,2} hence are not optimum for a hybrid blanket if high energy multiplication is desired. Plutonium metal is too difficult to work with, due to its chemical activity. Uranium dioxide and plutonium dioxide, because of their low thermal conductivity, produce unacceptably high peak tempera-

tures (higher than their melting points); they will not be considered further. Solid uranium metal must operate in its gamma phase, hence by itself is unsatisfactory; however, a slurry of uranium pellets in a lithium bath appears quite acceptable. The fuels we consider are a uranium-lithium slurry, uranium monocarbide, and uranium mononitride. (Unfortunately, many physical properties of UN — a most promising fuel developed for the discontinued SNAP-50 reactor program — are given only in classified documents.)

Fissionable fuel must be loaded in high volume percentages to attain a high energy multiplication. Based on Lee's results,¹ the fuel volume must be 50% or

Table 1. Physical properties of hybrid blanket fuels.^a

Fuel	Thermal conductivity at 973 K (W/m·K)	Density (Mg/m ³)	Melting point (K)	Comments
Thorium metal	41.8	11.72 ^b	2023	(1) Phase change at 1673 K (2) Isotropic properties (3) Resistant to Li at 873 K
Uranium metal	36.6	18.33 ^c	1407	(1) α phase 938 K β phase 938-1043 K γ phase 1043-1403 K (2) Highly unisotropic properties (3) Resistant to Li at 1273 K
Uranium dioxide	3.47	10.96 ^b	3073	k_f decreases by 60% at 366 K after 4000 MW-day/ton burnup
Uranium monocarbide	2.04	13.63 ^b	2653	(1) Good compatibility with Li (2) May react with SS at 1273 K (3) k_f reduces by 58% on irradiation at 1273-1773 K and 7500-10 000 MW-day/ton burnup
Uranium mononitride	2.5	14.31 ^b	2903+50	Very good compatibility with Li

^aData from Refs. 14, 15, and 16.^bTheoretical density.^cDensity at 922 K.

more. Two different approaches were used in the fuel region: (1) specially shaped fuel wedges held in place against the tubing containing the helium, with heat transfer enhanced by a low-pressure cover gas, and (2) small fuel pins or pellets in a lithium bath loaded into a can of the same construction as the lithium canning. Figure 6 illustrates the first approach: the spaces between the fuel

wedges, which allow for thermal expansion and burnup swelling of the fuel, are partly filled by a tungsten foil spring that holds the wedges against the tube. Figure 5 illustrates the second approach, in which a uranium-lithium slurry replaces the lithium. If the fuel is clad (for either approach), the cladding occupies about 10 vol% of the space otherwise usable for the fuel. Cladding almost surely is

Fig. 6. Cross section of the solid fuel region.

needed for the first approach but might be omitted for the second if we could assume good compatibility between the fuel and the

liquid lithium (the can acting as the cladding). It remains to be seen whether safety considerations will allow the latter approach.

Some Proposed Designs

Volumetric compositions for three different fuel-region designs (labeled A, B, and C) were computed using a 12-region model (see Fig. 7). The first seven regions in each design are:

- Region 1: First wall
- Region 2: First-wall coolant
- Region 3: First-wall coolant region containing helium tubing
- Region 4: Tubesheet/second wall
- Region 5: Bottom of lithium can
- Region 6: Lithium region (see Fig. 6)

Region 7: Top of lithium can.

The last three regions in each design are:

- Region 10: Bottom of canned thermal insulation
- Region 11: Canned thermal insulation
- Region 12: Top of canned thermal insulation.

Canned, evacuated thermal insulation is consistent with the modular approach and fits within each module.

Fig. 7. Axial zones of blanket module.

Design A embodies the first approach described in the previous section (fuel wedges in a low-pressure cover gas).

Designs B and C embody the second approach. In Design B, the fuel loaded into the can is in the form of pellets 1 cm in diam clad with 0.25-mm Nb-1%Zr. Design C assumes pellets of a "convenient" size. The composition of these designs is given in Tables 2 and 3.

In Designs B and C, the fuel volume percentage in Region 8 is calculated assuming hexagonal, close-packed spherical pellets. This results in 88.86 vol% pellets and 11.14 vol% lithium in the cans. If cladding is required, higher fuel-volume fractions can be achieved by using large pellets, because cladding thicknesses tend to be constant.

In Design A, Region 8 is a barrier to prevent liquid lithium from entering the fuel region. Radial swelling of the fuel is accommodated by the 1-mm void between the wedges; axial swelling is accommodated by the eight 1-mm voids that contain tungsten foil springs for axial fuel positioning. This results in 10-cm-long fuel wedges and allows for a 0.1% axial swelling of the fuel due to burnup.

In all designs, the volume percentages are calculated for a hot module.

The fuels for Design A could be either uranium monocarbide or uranium mononitride. As derived from the data in Ref. 8, the upper limit on the heat-removal capability of the helium coolant under the conditions of this analysis is about 96 W per cm^2 of tube-wall area. A module 50 cm from the first wall has a cross-sectional area of 376.86 cm^2 . The helium, tritium-recovery void, and swelling voids together account for 9.45% of this cross section (which does not provide heat); therefore, the area available for

Table 2. Composition of Regions 1-6 and 10-12 of proposed blanket designs.

Region	Thickness (cm)	Material composition (vol%)				Voids
		SS	Li	He	Nb	
1	0.10	100.00	—	—	—	—
2	0.90	—	100.00	—	—	—
3	3.55	4.72	88.94	6.14	0.13	0.07
4	0.10	94.94	0.59	4.34	0.06	0.07
5	0.05	4.04	0.59	4.34	90.96	0.07
6	24.20	4.04	87.41	4.34	4.14	0.07
10	0.05	95.10	0.45	4.34	0.06	0.07
11	0.85	6.10	0.45	4.34	0.06	89.05
12	0.05	95.10	0.45	4.34	0.06	0.07

Table 3. Composition of Regions 7-9 of proposed blanket designs.

Region	Design	Thickness (cm)	Material composition (vol%)						W
			SS	Li	He	Nb	Voids	Fuel	
7	A	0.05	4.04	0.59	4.34	90.96	0.07	—	—
	B	0.10	4.04	0.59	4.34	90.96	0.07	—	—
	C	0.10	4.04	0.59	4.34	90.96	0.07	—	—
8	A	0.10	95.93	—	4.34	0.06	0.07	—	—
	B	70.00	4.04	10.27	4.34	15.13	0.07	66.15	—
	C	70.00	4.04	10.27	4.34	4.14	0.07	77.14	—
9	A	70.10	4.04	—	4.34	10.32	5.11	75.87	0.32
	B	0.05	4.04	0.59	4.34	90.96	0.07	—	—
	C	0.05	4.04	0.59	4.34	90.96	0.07	—	—

energy deposition is 341.25 cm^2 . Based on five U-tubes, the total coolant-tube wall area in the fuel region is 2860 cm^2 , resulting in a heat-removal capacity of 274 kW per module. The resulting average energy deposition that matches the heat-removal capacity is 11.5 W/cm^3 . If we assume a 3.7-cm-radius unit-cell heat-transfer analysis with $T_i = 850 \text{ K}$, the peak fuel temperature for uranium monocarbide is 1236 K and for uranium

mononitride is 1165 K. In both cases the peak temperatures are about one-third the melting temperature of the fuel. Under conditions of burnup-reduced thermal conductivity (see Table 1), the peak temperature for uranium monocarbide increases to only 1768 K, still well below its melting point (2653 K).

For Designs B and C, the thermal conductivity of the lithium uranium mixture should be similar to that of liquid lithium.

Even if the thermal conductivity of the mixture were 50% of that for liquid lithium, the peak temperature would be only 1119 K. Therefore, this blanket design evidently is limited by the ability of the helium to remove the deposited energy rather than by any peak-temperature constraint in the fuel region that results from conductive heat transfer to the coolant tubes.

The typical module weighs about 386 kg. This module contains about 13 kg

of stainless steel, 18 kg of niobium, 6.44 kg of lithium, and 349 kg of uranium. Thus uranium constitutes more than 90% of the module's weight; however, the shielding effect of uranium against 14-MeV fusion neutrons may allow a thinner neutron shield region.² The total weight of this hybrid blanket (not including shielding) is about 3.6×10^6 kg, which poses some interesting structural problems not considered to date.

Refueling

Continuous refueling of the hybrid blanket might be desirable, in that the problems associated with fission-product buildup could be avoided. Although the solid fuel in Design A seems to preclude this possibility, Designs B and C might allow continuous replenishment if the lithium-uranium slurry could be pumped. We merely note this option, since our burnup calculations indicate that refueling will be required only every five to ten years.

We anticipate complete module replacement during refueling. Because of the blanket's nearly spherical shape, one way to replace modules would be to index the blanket within the main structural shell to an opening through which one or more of the modules could be removed by an extractor mounted outside the shell. This fundamentally difficult scheme needs further investigation.

Summary

The modular approach to blanket design has many advantages. Preliminary calculations indicate that a satisfactory design is achievable. Although this study is concerned with a hybrid blanket, the modular approach should be equally applicable to a pure fusion blanket. A limitation to the described design seems to be

the heat-removal capability of the helium coolant. Processing of the modules for fuel charge, etc., may be difficult. If the design were used for a laser, inertially confined plasma, it would be more tractable from an assembly/disassembly standpoint than for the magnetic system we have chosen to study.

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