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FIRST WALL AND BLANKET DESIGN FOR A HIGH WALL LOADING COMPACT TOKAMAK POWER REACTOR

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ABSTRACT

Among the specific limitations which tend to complicate a compact high wall loading (HWL) tokamak reactor design are high surface and nuclear heating, compactness leading to crowded components, unlikely breeding on the inboard side and frequent first wall/blanket replacement. This paper describes the mechanical, thermal hydraulic and tritium aspects of an improved blanket design for a high β (20%), high wall loading (~ 10 MW/m^2) compact fusion power reactor of 1000 MW_{th} power output.

INTRODUCTION

Recent trends in reactor designs have been toward smaller, higher power density, passively safe fusion devices which can compete more economically in the energy marketplace. Such reactors utilize relatively new physics concepts to access the high β operating regimes which lead to high neutron wall loading. For a tokamak, high β operation requires an indented, or so-called bean shaped plasma. To achieve such a plasma a coil must be placed at midplane on the inboard side of the reaction chamber.

The design of a fusion power reactor blanket is challenging in view of all the restrictions and requirements imposed on it. The task becomes more challenging when the blanket is for a compact high wall loading tokamak reactor. Besides the generic requirements of conventional blankets such as a breeding ratio > 1, high energy multiplication, efficient energy transfer, safety, environmental acceptability and many others, a high wall loading compact reactor blanket has additional limitations which must be accommodated. They are:

1. Very high surface and nuclear heating.
2. Compactness leading to crowded components.
3. Unlikely breeding on the inboard side.
4. Frequent replacement due to radiation damage.

Setting a limit of 1000 MW_{th} while utilizing a high β and requiring a neutron wall loading of 10 MW/m^2 leads to a tokamak with a minor radius of 0.6 m and an aspect ratio of 4.3. The high surface heating makes a separately cooled first wall a necessity. Crowded conditions on the inboard side coupled with the need for a bean shaping coil make it difficult to shield the TF coil and virtually eliminates the possibility of breeding on the inboard side. High neutron wall loading necessitates frequent blanket/first wall replacement. Therefore, minimizing the mass being handled, insuring suitable configurations for removal from the reactor and providing good accessibility and clearance are important considerations.

CONCEPT SELECTION AND GENERAL DESCRIPTION

The concept selected for the HWL blanket utilizes static $Li_{17}Pb_{83}$ contained in ferritic steel HT-9 cylinders and cooled with circulating He gas. Lithium lead has been recognized as a good breeding material for fusion reactors; however, most recent designs have utilized it in self-cooled blankets where it is pumped through the blanket and then taken to a steam generator for energy recovery. This leads to problems from MHD effects, corrosion and radioactive corrosion product transport, tritium containment, particularly in the steam generator, and large costly pipes with a high $Li_{17}Pb_{83}$ inventory in the primary loop.

A separately cooled LiPb blanket gets around most of these problems. Since the breeding material is not circulated, there are no MHD losses, no corrosion product transport and the LiPb inventory is limited to the blanket. Further, if the T_2 diffusing into the helium coolant is in the form of T_2O or DTO, the permeation into the steam generator can be made acceptable.

The Blanket Comparison and Selection Study (BCSS) has not included this concept among its top-rated seven tokamak blanket selections.[1] The critical issues identified at the time were temperature/heat flux on the first wall, corrosion temperature limits and tritium recovery from the He. The need for a separately cooled first wall in the HWL design makes the first issue irrelevant. With respect to the second issue, BCSS set a limit of 475°C for a static $Li_{17}Pb_{83}$ breeder in HT-9 structure, based on a corrosion rate of 25 μm/y, a number which may be relevant for a circulating system where plugging is of concern, but is quite arbitrary for a static system. Structural thinning is inconsequential, since the lifetime of the blanket is only 2 FPY. Further, experimental results of $Li_{17}Pb_{83}$ in HT-9 capsules simulating a static system show evidence of saturation where no loss is reported after 1000 hrs at 500°C.[2]

The issue of T_2 recovery from the He coolant is the most controversial. Recent experimental results indicate that T_2 diffusing through oxidized ferritic steel is predominantly in the form of T_2O.[3] Questions remain, however, of maintaining an oxide layer in the blanket tubes thin enough to allow diffusion but thick enough to provide oxidation. This can only be resolved experimentally. T_2O recovery from He is relatively straightforward with permeation to the steam generator maintained at acceptable levels.

Figure 1 is a cross section of the reactor through the center of a TF coil showing the first wall following the contour of the blanket on the outboard side and the shield on the inboard side. The static breeding material is contained in cylinders oriented in the poloidal direction as shown in Fig. 1. Coolant tubes are immersed in the breeding material with the inlet and outlet connections made on the same end of the cylinders. Upon entering the cylinder, the tubes spiral in a helical pattern around the inner surface of the cylinder wall from one end to the other, then turn around and come back through the center. The number of parallel circuits can be varied, depending on the amount of energy in the cylinder. The first wall is also He gas cooled with the same inlet temperature as the blanket.

The blanket and first wall are divided into 12 equal modules, sized for extraction between TF coils without circumferential translation. Each blanket module has its own first wall which wraps around the plasma in the poloidal direction and shadows the blanket from radiant energy.

The two outstanding features of the blanket design are its ability to withstand a He gas leak with ease, making it inherently safe and the ability to entirely drain the breeding material such that a blanket module is light for handling during maintenance.

FIRST WALL DESIGN AND THERMAL HYDRAULICS

The first wall is a membrane type HT-9 structure with built in coolant channels. Figure 2a shows a cross section of the first

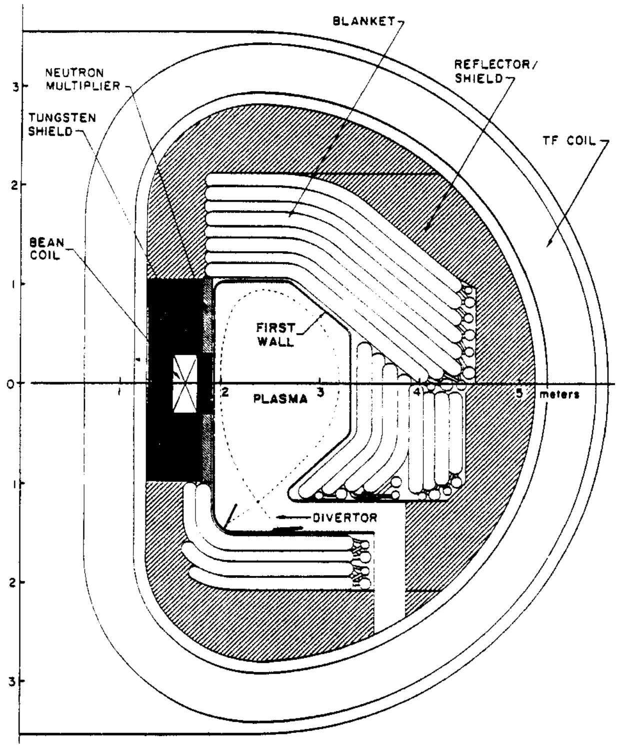

Fig. 1. Cross section of HWL reactor

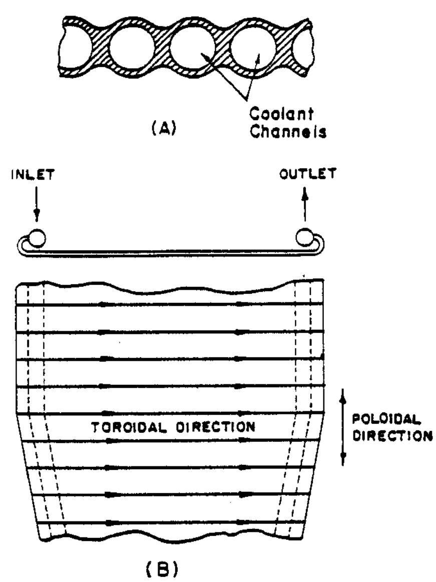

Fig. 2. a. Cross section of membrane first wall.
b. Section of first wall showing flow path.

wall and Fig. 2b a section indicating the inlet and outlet manifolds and the coolant flow direction.

High pressure helium is used to cool the first wall entering at 80 atm and 275°C, and leaving at 77.5 atm and 400°C. The inlet conditions are selected to match those for the blanket coolant while the exit coolant temperature is selected so that the maximum wall temperature at the "hot spot" is limited to about 500°C. This temperature is well below the point when the yield stress for HT-9 begins to drop perceptibly with increasing temperature.

Each first wall module contains 754 coolant channels with an inside diameter of 0.8 cm and a flow length in the poloidal direction varying between 1.02 m and 1.73 m for the inboard and outboard vertical regions, respectively. Since all the coolant channels in a given module are fed by and discharged into common headers, it is necessary to use "orificing" at the channels' inlet, except for the longest channels. Orificing allows the flow rate to be tailored to nearly match the power removed in each channel which is proportional to the channel length.

The coolant supply and discharge headers for the twelve wall segments are arranged so that six similar units are formed consisting of two segments connected in series. The helium coolant enters the supply header of the first segment at 80 atm and 275°C, and discharges into the supply header of the second segment. The helium coolant leaves the discharge header of the second segment at 400°C and 77.5 atm and is combined with that from the other six segment discharge headers before being supplied to the power cycle. This arrangement allows the coolant velocity in the channels to be increased to about 100 m/s as recommended for high heat flux gas-cooled systems[4] so that reasonably high heat transfer coefficients can be obtained. The maximum coolant velocity in the hot segments varies between 90.2 m/s and 110.8 m/s corresponding to heat transfer coefficients of 0.722 and 0.873 W/cm^2°K, respectively.

A two-dimensional finite-element computer code, FEM2D, has been used to determine the wall temperature distribution.[5] Axial conduction in the flow direction and heat losses (or gains) from the outside tube surface on the blanket side have been ignored. The analysis has been performed parametrically for different values of the plasma-side surface heat flux, and coolant heat transfer coefficient. A uniform value of the volumetric heat generation rate, q''' = 52 W/cc, has been used, consistent with a divertor efficiency of 70%. The results are shown in Fig. 3. For all the cases examined, the maximum wall temperature occurs at the crest of a coolant channel.

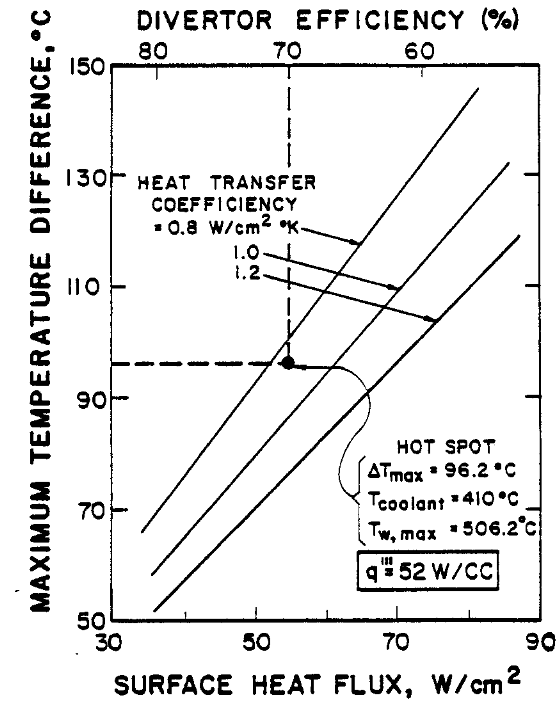

Fig. 3. Parametric results showing variation of the maximum first wall temperature rise with the surface heat flux and local heat transfer coefficient (the maximum temperature rise is defined as the difference between the maximum wall temperature and the local coolant bulk temperature).

For each coolant channel in the hot segment, upon knowledge of the total power removed and coolant flow rate, an energy balance is used to determine the coolant exit temperature. Variation of the coolant exit temperature with poloidal angle is shown in Fig. 4. These results, together with the local heat transfer coefficient values and the parametric results, are used to determine the variation of the maximum wall temperature with poloidal angle at the coolant channels' exit (Fig. 4). These results show that the hottest point in the first wall will be at the exit section of the coolant channels in the outboard vertical region of each of the six hot segments and is equal to 506°C. Table I summarizes the physical and thermal hydraulic parameters of the first wall.

BLANKET DESIGN AND THERMAL HYDRAULICS

The blanket consists of 15 cm OD, 14.5 cm ID ferritic stainless steel HT-9 cylinders containing static $Li_{17}Pb_{83}$, oriented poloidally as shown in Fig. 1. The cylinders cover the out-

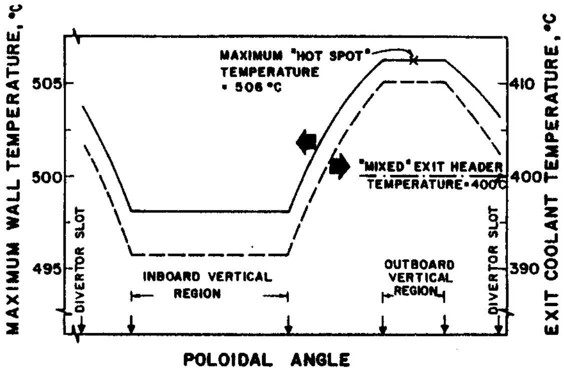

Fig. 4. Coolant exit temperature and maximum wall temperature for the various coolant channels.

Table I. Physical and Thermal Hydraulic Parameters of the First Wall

Material of construction	HT-9
Coolant	He gas
Overall thickness (cm)	1.0
Tube ID (cm)	0.8
Tube wall thickness (cm)	0.1
Tube center to center (cm)	0.9
First wall surface area (m^2)	106
First wall mass/module (tonnes)	0.30
First wall volumetric heating (MW_{th})	17.4
Surface heating (70% divertor eff.) (MW_{th})	61.5
Heat generation rate (W/cm^3)	52
Avg. plasma side surface heat flux (W/cm^2)	55
Coolant inlet temp. (°C)	275
Coolant exit temp. (°C)	400
Coolant inlet pressure (atm)	80
Coolant outlet pressure (atm)	77.5
Coolant flow rate (kg/hr)	4.38×10^5
Pumping power (70% comp. eff.) (MW_e)	7.00
Maximum coolant velocity (m/s)	110
Maximum first wall temp. (°C)	506.

board side, the upper and lower side and a small fraction of the inboard side of the plasma chamber. The breeding material is cooled with 2.1 cm OD, 0.1 cm wall thickness tubes immersed in it with 80 atm helium coolant circulating through at an inlet temperature of 275°C. The coolant tubes enter and exit the cylinder from one end making the manifolding relatively easy. After entering the cylinder, the coolant spirals helically around the inner cylinder wall from one end to the other, then turns around and travels back through the center of the cylinder as shown in Fig. 5. The large variation in nuclear heating from the first row to the last is controlled by providing coolant passages in parallel. The cylinders are stacked on a

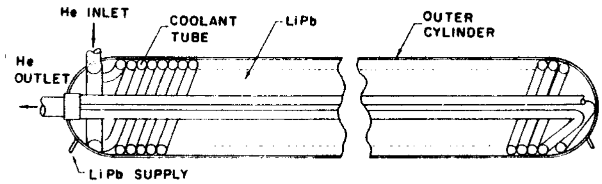

Fig. 5. Cross section of a blanket cylinder.

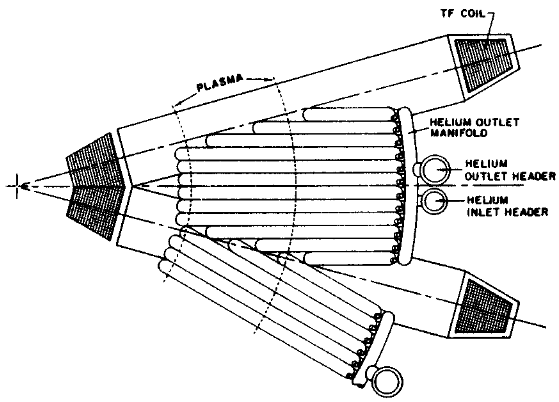

Fig. 6. Top view of first row of cylinders in the blanket.

triangular pitch forming eight rows with an overall blanket thickness of 106 cm. They are all interconnected for easy filling and draining of the breeding material.

A blanket module consists of three cylinder assemblies, the upper, the middle and the lower, all structurally connected such as to act as a unit. Figure 6 is a top view of the first row of the upper cylinder assembly. It can be seen that the cylinder lengths are varied to accommodate the wedge shape of the module. Further, adjacent modules can have cylinder lengths adjusted to fill in the voids at the interface between them. Subsequent rows of cylinders are also interwoven to virtually eliminate any direct neutron streaming through the blanket. Table II summarizes the physical parameters of the blanket.

In order to determine if the structural design limits are satisfied one must identify the region of peak energy deposition and the corresponding peak temperature. In the current tokamak design the complex geometrical arrangement of the plasma, first wall, blanket and shield causes the wall loading due to neutrons and

Table II. Physical Parameters of HWL Blanket

Cylinder OD (cm)		15
Cylinder ID (cm)		14.5
Cylinder length (cm)		82-290
Coolant tube OD (cm)		2.1
Coolant tube ID (cm)		1.9
Distance between coils (cm)		0.1-15
Number of modules in reactor		12
Number of cylinders per module		202
Mass of module full of LiPb (tonnes)		49
Mass of module empty of LiPb (tonnes)		5.7
Material Fractions		
	1st row	8th row
$Li_{17}Pb_{83}$ (%)	51.3	75.5
HT-9 (%)	10.8	6.2
He + void (%)	37.9	18.2

alpha particles to have a large degree of spatial variation. In addition, the peak structural temperature will be influenced by the helium gas flow path relative to the energy wall loading near its peak value (~ 12 MW/m^2). To evaluate the location of the hot spot in the blanket, it is necessary to determine the cylinder with the highest energy deposition, then determine the maximum coolant temperature near the structural cylinder wall and finally, determine the maximum structure temperature at that location due to the local heat flux.

Based on plasma and neutronic analyses, the peak wall loading occurs in the outboard side cylinder at the midplane and is equal to an average value of 11 MW/m^2. The maximum structural temperature in this "worst case" cylinder depends on the bulk coolant temperature profile. To determine this, we write a steady state, one-dimensional energy balance for the helium coolant and the LiPb cylinder along its length, neglecting axial conduction and natural convection which is suppressed by the high magnetic field. The solution to this set of simultaneous coupled ordinary differential equations yields the temperature profile in the outer tubes, the inner tubes and the breeder material.

Figure 7 gives a qualitative bulk helium temperature distribution Tc_1 in the outer tube, Tc_2 in the inner tube and T_f, the breeder material temperature. Using an inlet temperature of 275°C the maximum coolant temperature occurs at the closed end and is equal to 525°C. The peak structural temperature can now be calculated using the assumption that the blanket outer boundary is adiabatic. Volumetric heating of 39 W/cm^3 in the cylinder wall and coolant tubes closest to the plasma when transferred to the coolant results in a peak structural temperature of 535°C.

If a coolant line ruptures, the cylinder will be pressurized to 80 atm and the hoop stress in it will be 239 MPa. The yield

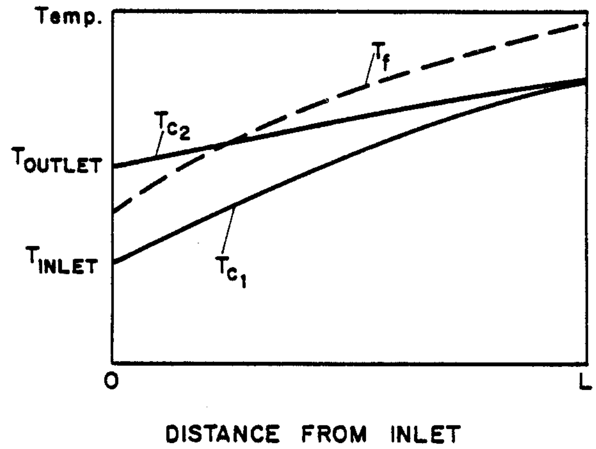

Fig. 7. Qualitative temperature profile in the He gas coolant and the breeding material as a function of the distance from the inlet.

strength for HT-9 at the "hot spot" temperature of 535°C is 356 MPa. Table III gives the thermal hydraulic parameters of the blanket.

TRITIUM INVENTORY, EXTRACTION AND PERMEATION

Tritium extraction in this blanket concept depends on its diffusion into the He gas coolant and through the cylinder walls to the plasma chamber. For a 1000 MW_{th} reactor with a breeding ratio of 1.1, the steady state T_2 permeation is 28 moles/d. Tritium diffusing through the cylinder walls is processed with the plasma exhaust. That diffusing into the He gas coolant must be extracted such that the T_2 pressure is $< 10^{-9}$ torr in order to limit losses to the steam cycle to 10-100 Ci/d. This is possible if the T_2 is oxidized to T_2O.

To insure an oxide layer on the He side of the coolant tubes, a small quantity of O_2 is mixed into the He gas. This oxide layer reduces the T_2 permeation rate by an order of magnitude. Tritium permeation depends on the surface area wetted by the LiPb and the thickness and temperature of the material being permeated. Cylinders in the back of the blanket require fewer cooling tubes, thus exposing more cylinder surface for direct T_2 diffusion into the plasma chamber. Taking this into account, the T_2 pressure in the blanket is calculated to be 23-33 torr and the T_2 inventory in the 520 tonnes of LiPb, 35-42 g.

There is a certain amount of controversy on the state of T_2 diffusing through an oxidized tube. Although somewhat crude, a recent experiment performed at EG&G Idaho indicates that all

Table III. Thermal Hydraulic Parameters of the "Worst Case" Cylinder in the Blanket

Peak power density (W/cm^3)	87
Avg. power density (W/cm^3)	47
Avg. linear power in cylinder (kW/cm)	5.56
Total power in cylinder (MW)	0.97
Inlet bulk He gas temp. (°C)	275
Inlet He gas pressure (atm)	80
Inlet He gas velocity (m/s)	68
Max. He gas velocity (m/s)	96
Outlet He gas temp. (°C)	500
Max. He gas temp. (°C)	525
Max. HT-9 structural temp. (°C)	535
Estimated pumping power for the whole blanket (MW_e)	25-50

the T_2 diffusing through an oxidized ferritic steel tube is in the form of T_2O.[3] The major criticism of this experiment is that it was of very short duration. Clearly, a careful balance has to be found which will maintain an oxide layer capable of supplying adequate O_2 to oxidize the T_2 but not impede the diffusion process. These questions can only be answered by carefully planned experiments of longer duration than that performed at EG&G.

If the diffusing T_2 is in the form of T_2O, then it can be readily separated with molecular sieves keeping $PT_2 \ll 10^{-9}$ torr. However, if a fraction is in the form of T_2, then there are several possibilities for alleviating the problem. The first is called D_2 swamping and works on the principle of adding D_2 to the He gas coolant. This scheme, proposed by D.K. Sze of ANL, utilizes the reaction $D_2 + T_2 = 2DT$ to reduce the T_2 pressure. To be effective, however, the D_2 pressure must be on the order of 1000 torr and this may have undesirable consequences. The second possibility is more radical and should be only thought of as a last resort. It depends on a slowly circulated liquid lithium barrier between double walled steam generator tubes. Tritium is captured via the reaction $2Li + T_2 = 2LiT$ which proceeds very rapidly at these temperatures. The LiT is then separated from the Li for T_2 recovery. The penalty on the power cycle will be minimal and the Li inventory will be small enough so as not to pose a safety hazard.

Finally we have estimated the T_2 implantation into the first wall from the plasma and eventual diffusion into the He gas coolant. Using a D&T ion energy of 200 eV, a flux of 2.5 x 10^{16} $cm^{-2}s^{-1}$, a wall area of 100 m^2, a wall thickness of 0.1 cm and an average temperature of 414°C, the permeation is determined to be 4 g/d. This, together with the 28 moles/d from the blanket, gives a maximum of 172 g/d of T_2 diffusing into the He gas coolant.

CONCLUSIONS

A He gas cooled LiPb blanket presented in this paper retains most of the desirable features of a self-cooled LiPb blanket while eliminating many of its drawbacks. Variations of this design can be adapted to all fusion reactor concepts and therefore, deserve further development.

ACKNOWLEDGEMENT

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