Spiral Blanket: An Innovative Idea for a Solid Wall Blanket with Potential for High Neutron Wall Loading

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SPIRAL BLANKET: AN INNOVATIVE IDEA FOR A SOLID WALL BLANKET WITH POTENTIAL FOR HIGH NEUTRON WALL LOADING

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ABSTRACT

The Advanced Power Extraction (APEX) program[1] is exploring concepts for blanket designs that can enhance the potential of fusion while using both liquid and solid walls. In that context, an innovative blanket design of dry solid wall configuration, with potential for high neutron wall loading has been proposed. The blanket utilizes nano-composited ferritic (NCF) steel structure (designated 12YWT), which has a maximum operating temperature of 800°C.[2] The cooling/breeding material is Flibe (Li_2BeF_4), a low viscosity version of this molten salt, which has a melting temperature of 465°C and is compatible with ferritic steels up to 700°C.[2] The dimensions for this study have been taken from ARIES-AT.[3] The blanket module, which extends 0.3 m in the toroidal direction at mid-plane, is equipped with spiraling discs ramping from the bottom to the top. The coolant enters on the bottom at 500°C, then travels on the spiral discs, from the rear of the module to the front, then back to the rear, all the way to the top where it exits from the module at 590°C. On its way up, the coolant velocity is amplified at the first wall (FW) by centrifugal action, providing a high heat transfer coefficient for dissipating the high surface heating. The 3 mm thick FW is scalloped with semicircular projections inclined in the flow direction. This facilitates unimpeded smooth flow at the FW, while at the same time stiffening the FW against pressure, obviating the need for welded reinforcements. The discs are made of two halves assembled together with a Be pebble bed enclosed as a neutron multiplier. Preliminary neutronics analysis for the outboard blanket has shown a local tritium breeding ratio (TBR) of 1.36, and an energy multiplication (M) of 1.26 including contribution from a secondary module, using natural Li. The pressure drop of 0.56 MPa in the module produces a modified primary bending stress of 124 MPa where the allowable stress for this material at 750°C is 142 MPa. Without optimization, this blanket is capable of dissipating an average neutron wall loading of 6.4 MW/m^2, with a peak value of 9.6 MW/m^2 and a peak surface heating of 1.3 MW/m^2. The coolant picks up energy in the secondary blanket exiting the reactor at 600°C. Assuming the use of a supercritical steam power cycle, an efficiency of 48% can be expected.

I. INTRODUCTION

The Advanced Power Extraction (APEX)[1] program has been instituted by DOE-OFE for exploring methods of maximizing the neutron wall loading in fusion devices. The program covers liquid protected walls and solid walls. Task IV of the program is exploring solid first walls (FW), where the present design had its origin. Previous solid FW designs have been constrained by severe temperature and stress requirements. The advent of nano-composited ferritic (NCF) steels, which have small amounts of ceramic materials added, has opened up new possibilities for solid wall designs. The steel designated 12YWT can operate at a temperature of 800°C and is compatible with Flibe, a molten salt (Li_2BeF_4) up to 700°C.[2]

Although there have been many solid FW blanket designs performed in the past, they have always been configured for a specific neutron wall loading dictated by the physics of the device. This is the first time that a design has been earmarked to determine the upper limit of neutron wall loading in a solid wall design. This aspect has been aided by the superior structural possibilities of nano-composited ferritic steel. This unconventional innovative design shows that solid wall blankets have a legitimate place along with liquid protected walls as the first line of defense facing an intense neutron plasma. The design takes advantage of two important aspects, namely the accelerated fluid velocity due to the centrifugal action of the flow at the FW and the uniform distribution of the flow around the module, allowing all the coolant to participate in dissipating the high surface heating.

II. GENERAL DESCRIPTION

The dimensions for this study have been adopted from the ARIES-AT design[3] as shown in Figure 1. In this paper only the outboard module will be discussed, since it has the highest neutron loading and surface heating. The area of the outboard module at mid-plane is 0.3 x 0.3 m^2, but at the upper and lower extremities the toroidal dimension is reduced to 0.22 m to allow for the reduced circumference. The heart of the design consists of a spiral that fills the module from bottom to top. The front of the discs about the

FW, which consists of 3 mm thick semicircular scallops, is inclined in the direction of coolant flow. The spiral is made up of discs mounted on a shaft individually, and the shape of the discs varies from near rectangular at the extremities to near square at the mid-plane. Figure 2 is a side and front view of the module. The side view shows the FW on the left with a cutaway showing two discs. The front view, which faces the plasma, shows the progression of scallops in the direction of flow. Each disc is fabricated separately, consisting of two perforated halves, an upper and a lower half, which when assembled, enclose a bed of Be balls. After the halves are diffusion bonded together, the disc is placed on the shaft, interfacing with the previous disc in a parting line at the rear of the module as shown in Figure 2. The shape of the discs tapers from the shaft to the FW, being 5 cm thick at the shaft and 1.0 cm thick at the FW. This is done to minimize the impact of the disc on the flow at the FW. The Be packing fraction is 65%; however, additional space can be provided on the top of the discs for swelling if needed.

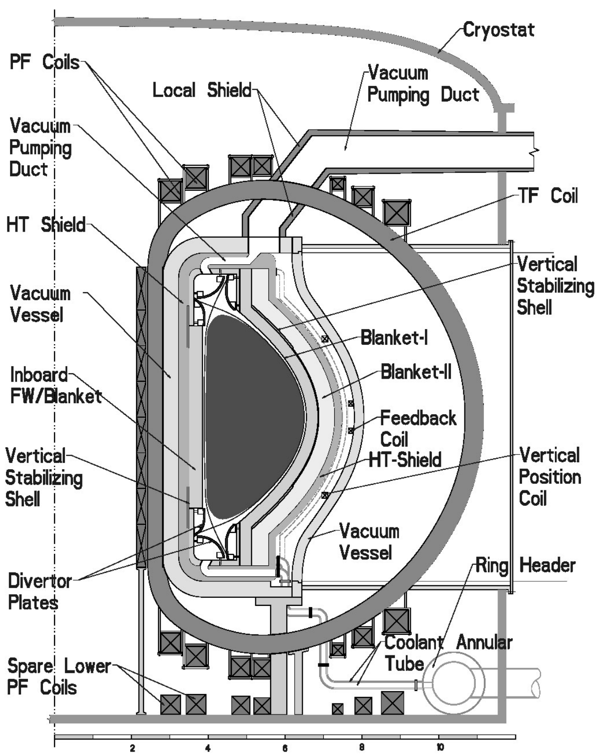
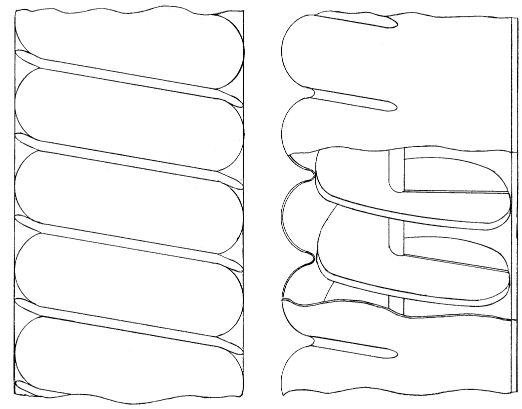


Figure 1. Cross-Section of ARIES-AT.



Front View from Plasma Side View Cut-away

Figure 2. Front and Side Views of Module Section.

III. THERMAL HYDRAULICS

The thermal hydraulics for the spiral blanket are based on maximizing the neutron wall loading and the resulting surface heating while achieving limits on material properties, and a coolant temperature which can yield a high power conversion efficiency. Thus, all the coolant flow participates in heat dissipation at the FW, by circulating on the discs from the rear of the module to the front, then back to the rear, all the way to the top. The centrifugal action of the coolant accelerates the velocity at the FW increasing the heat transfer coefficient. However, the pressure drop is figured on the average velocity in the channel, which is on the order of 5 cm x 13.5 cm. Thus the pressure drop is low and the absolute pressure in the module is low. Figure 3 shows the neutron wall loading and surface heating distribution in the poloidal direction. It shows that they peak around the mid-plane and that is where the FW temperature peaks as well. Figure 4 shows the FW inner and outer surface temperature, as well as the Flibe temperature as a function of the vertical distance along the blanket. The coolant enters the module on the bottom at 500°C and exits at the top at 590°C for the case of 6.4 MW/m^2 average neutron wall loading. The maximum FW external surface temperature is 800°C while the interface temperature with Flibe is 692°C. The coolant picks up another 10°C on the way down in the secondary blanket and is piped to the steam generator at 600°C. Assuming a supercritical steam cycle with double walled steam generators, an efficiency approaching 48% is possible. Table I gives the physical parameters of the spiral blanket and Table II provides the thermal hydraulic parameters.

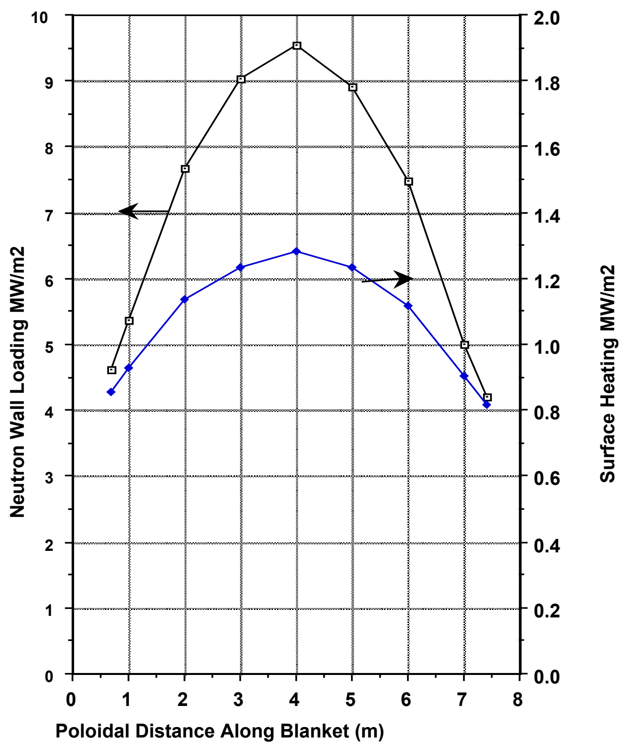


Figure 3. Poloidal Distribution of NWL and Surface Heating.

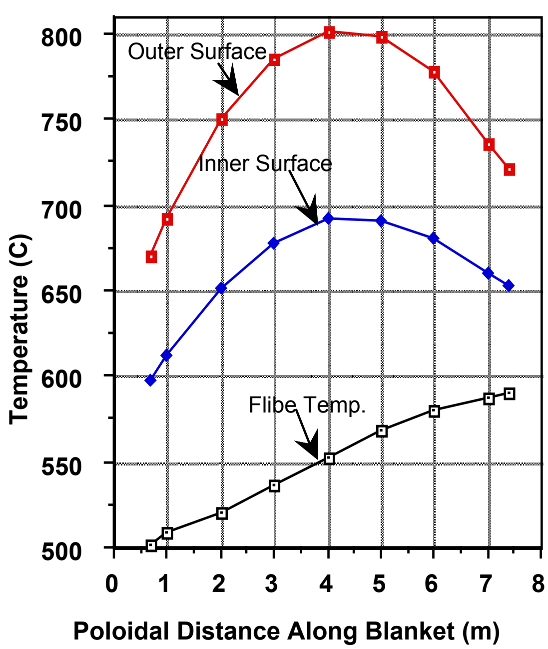


Figure 4. Temperature Profiles as a Function of Poloidal Distance Along the Blanket.

Table I. Physical Parameters of Blanket

Poloidal height along module (m)	6.7
Mid-plane module cross-section (cm^2)	30 x 30
Cross-section at extremities (cm^2)	30 x 23
First wall thickness (cm)	0.3
Number of discs	70
Disc thickness at first wall (cm)	1.0
Disc thickness at shaft (cm)	5.0
Disc cladding thickness (cm)	0.2
Dimension of a scallop (cm)	6.7
Average channel width (cm)	5.4
Average fluid path length (m)	33.0

Table II. Thermal Hydraulic Parameters for the Case of 6.4 MW/m^2 Average NWL

Maximum NWL (MW/m^2)	9.54
Average NWL (MW/m^2)	6.4
Maximum surface heating (MW/m^2)	1.3
Mid-plane, mid-channel velocity (m/s)	3.3
Range of velocities at FW (m/s)	5.5-6.3
Coolant inlet temperature (°C)	500
Coolant exit temperature from module (°C)	590
Coolant temperature to power cycle (°C)	600
Maximum steel temperature (°C)	800
Maximum interface with Flibe (°C)	694
Pressure drop through module (MPa)	0.56

IV. PRELIMINARY STRESS ANALYSIS

The corrugated first wall is modeled with a toroidal span of 30 cm and was analyzed using beam theory. The bending shape factor is 1.47. Stresses were calculated for a range of wall thicknesses from 2-6 mm, a pressure of 0.56 MPa and a temperature of 750°C, which is the maximum average temperature of the steel. Figure 5 shows the P_b/K_{eff} stresses as a function of FW thickness for two NCF materials, 12YWT and MA957. These stresses include time effects for the estimated lifetime of the primary blanket. As can be seen, the 12YWT material qualifies below the S_t limit for a 3 mm thick wall with margin to spare. The modified primary bending stress is 124 MPa, while the allowable stress is 142 MPa. The second material, MA957, does not qualify.

V. NEUTRONICS FEATURES

Neutronics calculations have been performed for the spiral blanket to determine the key nuclear performance parameters. The ONEDANT module of the DANTSYS 3.0 discrete ordinates particle transport code system[4] was used to perform the calculations utilizing the FENDL-2 data library.[5] The model includes a 30 cm thick spiral blanket composed of 74.4% Flibe, 18.67% Be and 6.93% NCF structure. A 45 cm thick secondary breeding blanket is also included in the outboard (OB) side with 95% Flibe and 5% NCF structure. The local tritium breeding ratio (TBR) is 1.36 in the OB region and 1.16 in the inboard (IB) region using natural lithium. The overall TBR is estimated to be 1.33 excluding tritium breeding in the divertor region that could add ~0.05 to the overall TBR. Although enriching the lithium in ^6Li is expected to enhance the TBR, the relatively high TBR implies that natural lithium can be used. The energy multiplication (M) is 1.12 in the spiral blanket, and when the secondary blanket is included, it rises to 1.26.

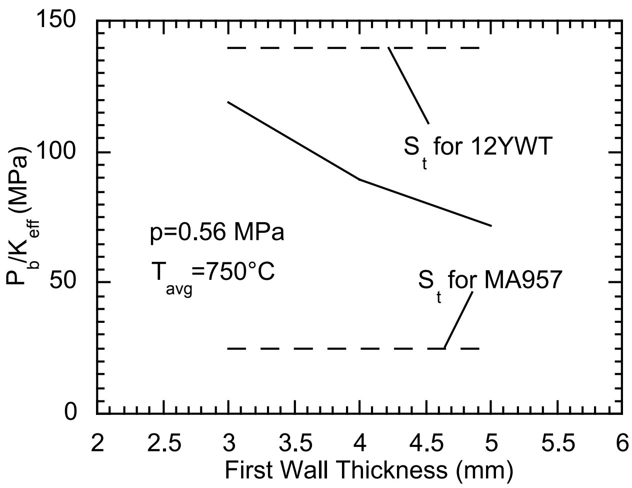


Figure 5. Variation of a Modified Primary Bending Stresses P_b/K_{eff} as a Function of FW Thickness.

Nuclear heating in the constituents of the spiral blanket was calculated and used as input in the thermal hydraulics analysis. The radial variation of power density in the NCF structure, Flibe and Be is shown in Figure 6 normalized to a neutron wall loading of 5 MW/m^2. The average power density in the spiral blanket at the poloidal location with this neutron wall loading is 18 W/cm^3. The poloidal variation of power density was determined by scaling the results with the poloidal distribution of the neutron wall loading. The peak dpa rate in the first wall is 110 dpa/FPY at the peak neutron wall loading of 9.54 MW/m^2. For a 200 dpa damage limit this corresponds to a lifetime of ~1.82 FPY for the spiral blanket. On the other hand, the peak dpa rate in the secondary blanket is only 6.1 dpa/FPY implying that it will be a lifetime component. The peak helium production rate in the Be spiral is 34,960 appm/FPY. This corresponds to total He production that represents ~6% of the Be atoms at the end of the spiral blanket life. This is an important parameter for assessment of expected irradiation swelling in the Be.

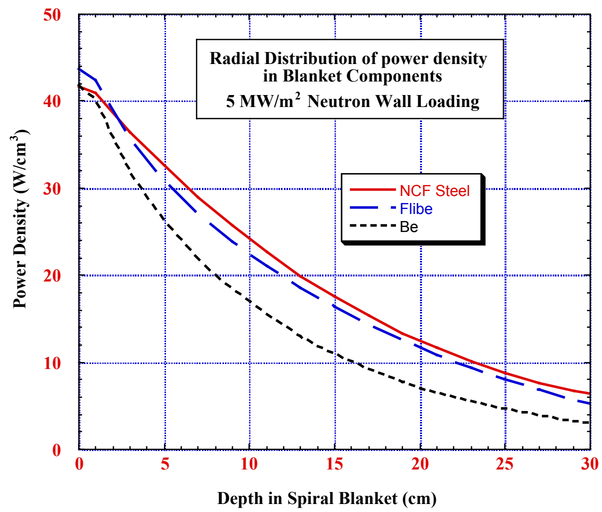


Figure 6. Radial Distribution of Power Density in Blanket Components.

VI. TRITIUM INVENTORY IN Be

A critical issue associated with using Be in fusion blankets is the amount of tritium produced and retained in the beryllium. Tritium production was calculated in the Be for the expected spiral blanket lifetime. The total spiral blanket volume is 104.2 m^3 (70.8 m^3 OB and 33.4 m^3 IB) resulting in a total Be inventory in the spiral blanket of 36 tonnes. The total tritium production in the Be of the spiral blanket at its end-of-life is 1.5 kg. This total tritium production is smaller than in solid breeder designs due to the smaller Be inventory and distributing the Be over the blanket thickness that results in a lower average neutron flux. In the EU Demo, Be inventory is ~300 tonnes and total tritium production is 22 kg. The tritium inventory will be much lower than the tritium production due to tritium permeation out of Be. It is expected that tritium will come out if we heat up the blanket to 700°C.[6,7] Based on available experimental data, the temperature at which most of tritium is released is in the range 500-700°C depending on the density and fluence level. Lower density and higher fluence result in lower temperature for release of tritium. The Be temperature in the spiral blanket varies over the spiral. The Be temperature distribution indicates that the temperature at the bottom of the spiral is 580-770°C and at the top is 680-870°C. Therefore, we expect that tritium inventory in the Be spiral will not be an issue.

VII. NCF STEEL COMPATIBILITY WITH Be

Ferritic steel in contact with Be forms a discontinuous brittle superficial layer adherent to the steel with a concomitant generation of holes in the Be. The thickness of the layer depends on the Be/steel interface temperature and the length of time they are in contact. Estimated for the

1.82 FPY at a maximum temperature of 875°C, the layer thickness is 1100 μm where it is in contact with the shaft of 5000 μm thickness.[8] All the other contact places are at lower temperatures resulting in thinner layers. It does not appear that this is an insurmountable issue.

VIII. FABRICATION

One of the advantages of the spiral blanket is that its fabricability appears to be possible given the limitation on forming and joining NCF steels. The front end of the module can be formed from a single sheet of steel, including the FW scallops starting in the center of the side wall and all the way to the opposite side wall. The FW is reinforced by these scallops, thus obviating the need for welding any additional structure. Strengthening of NCF steels is dependent on the direction of stretching, or forming. Thus the module forming process has to take this into account, to insure that all parts of the module are equally strengthened. The next step is to assemble the discs on the shaft and insert them into the module chassis. Minor adjustments and additions have to be made to insure that the discs are well restrained against shifting and the shaft is well supported. Finally, the rear plate is attached to the module using diffusion bonding. In effect, forming and diffusion bonding are the only processes needed to fabricate the spiral blanket, both of which have been demonstrated with NCF steels.

IX. CONCLUSIONS

The spiral blanket is an innovative design of a solid FW concept utilizing NCF steel in combination with Flibe coolant. Thermal hydraulic evaluation has shown that it is capable of dissipating an average neutron wall loading of 6.4 MW/m^2, a peak neutron wall loading approaching 10 MW/m^2 and surface heating of 1.3 MW/m^2, while satisfying the temperature limits ascribed to NCF steel of 800°C maximum and 700°C interface with Flibe. The pressure drop is a modest 0.56 MPa and the stresses for 12 YWT NCF steel are well within limits with time effects taken into account. The overall TBR is 1.33 using natural Li and the energy multiplication is 1.26. The amount of tritium bred in the Be over the estimated blanket lifetime is 1.5 kg. However, at the operating temperature of the Be, most of the tritium will diffuse out. Fabrication of the blanket using forming techniques and diffusion bonding appears to be possible. Using an inlet Flibe temperature of 500°C, an outlet from the primary blanket of 590°C and after traversing the rear blanket, exiting at 600°C, and assuming a supercritical steam power cycle with double wall heat exchanger, an efficiency of near 48% can be expected. Finally, this unconventional innovative design shows that solid wall blankets have a legitimate place along with liquid protected walls as a first line of defense facing an intense neutron plasma.

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