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Abstract— **Three blanket design concepts were evaluated for the High Average Power Laser (HAPL) chamber. These include a self-cooled lithium blanket, a helium-cooled solid breeder blanket, and a dual-coolant lithium lead blanket. The nuclear features of the three candidate blankets were compared. The Li blanket seems to be the preferred option based on neutronics considerations. However, other considerations should be included for selection of the reference blanket.**

Keywords-Laser fusion; lithium blanket; solid breeder; lithium lead; tritium breeding; nuclear heating

I. INTRODUCTION

The High Average Power Laser (HAPL) program led by the Naval Research Laboratory is carrying out a coordinated effort to develop Laser Inertial Fusion Energy (Laser IFE) based on lasers, direct drive targets and a dry wall chamber [1]. A primary focus is the development of a tungsten-armored ferritic steel (FS) first wall (FW) that must accommodate the threat spectra from the fusion micro-explosion. Only a thin region of the armor (10-100 μm) will experience the highly cyclic x-ray and ion energy deposition transients. The FW structure behind the armor as well as the blanket will operate under quasi-steady-state thermal conditions, similar to MFE conditions. This allows for the possible use of blanket designs that are being developed for MFE and allows maximum utilization of information available from the large international MFE blanket R&D effort. We carried out scoping studies of possible blanket designs that can be integrated with the FW protection scheme.

Three blanket design concepts were evaluated. One of the concepts is a self-cooled lithium blanket. The absence of magnetic field in IFE allows taking advantage of the high heat transfer capability of lithium without the MHD issue of concern in MFE. The second concept consists of a number of Li_4SiO_4 solid breeder (SB) and Be multiplier packed bed layers separated by helium cooling plates and arranged in parallel to the FW. The third design is based on the dual coolant lithium lead (DCLL) blanket concept where He is used to cool the FW and blanket structure. SiC inserts are used in the LiPb flow channels to thermally isolate the high temperature LiPb from the low temperature structure. All concepts utilize low activation FS F82H structure, constrained to a maximum temperature of 550°C [2]. In this paper, we discuss the main design features of the blanket concepts and compare the expected nuclear performance parameters.

Common chamber parameters were used in the neutronics analyses of the three blanket concepts. The chamber radius at mid-plane is 6.5 m. The ONEDANT module of the DANTSYS 3.0 discrete ordinates particle transport code system [3] was used to perform the neutronics calculations utilizing the FENDL-2 nuclear data library [4]. The chamber is modeled in spherical geometry with a point source at the center emitting neutrons with a softened energy spectrum resulting from interactions between fusion neutrons and the dense target materials. 70.5% of the target yield is carried by neutrons with an average energy of 12.4 MeV. For a target yield of 150 MJ and 12 Hz repetition rate (fusion power of 1.8 GW), the peak neutron wall loading at mid-plane is 2.4 MW/m^2. A 1-mm thick tungsten armor is bonded to the inner surface of the FW in all design concepts. A helium-cooled steel vacuum vessel/shield is utilized behind the blanket.

II. SELF-COOLED LITHIUM BLANKET

One of the concepts considered is a self-cooled lithium blanket. Figure 1 is a cross-section of the chamber showing a cylindrical vacuum vessel (VV). This blanket concept has similarities with the one used in ARIES-AT [5]. There are 12 side blanket modules in the reactor, each subtending 30° of circumference. At mid-plane the major radius is 6.5 m but at the ends, it tapers down to 2.5 m. Each module is made up of 13 sub-modules, which vary in width and depth to accommodate the reduction in radius. The sub-modules consist of two concentric rectangular tubes separated by a constant gap as shown in Fig. 2. The blanket module containing the ports has a special sub-module in the center, which contains the ports. Unlike the other sub-modules, it has a constant width from top to bottom, in order to provide adequate room for the ports without compromising the FW coolant flow. Lithium coolant enters the sub-module at the bottom, then flows at a high velocity in the gap between the tubes to cool the FW. Vanes are provided to allow the coolant to spiral around the tubes in order to even out the temperature. At the top, the coolant makes a 180° turn, then travels back at a very low velocity through the large central channel of the inner tube exiting at the bottom. By this action, the fluid is allowed to pick up heat from neutrons, but the poor heat transfer allows the channel walls to stay at a lower temperature.

Neutronics calculations were performed to determine the relevant nuclear performance parameters for the blanket. The tritium breeding ratio (TBR) was found to maximize at 20% ^{6}Li enrichment but with a TBR gain enhancement of only 2.5% and an order of magnitude increase in Li cost. Hence, natural

lithium is used. Li enrichment can be used as a knob in the design allowing for adjustment of the TBR and shielding if needed. The lifetime of the plant is assumed to be 40 full power years (FPY). For the VV to be a lifetime component with cumulative end-of-life radiation damage <200 dpa, it was determined that the side blanket thickness should be at least 47 cm. The peak damage and helium production rates in the FW steel at mid-plane are 19.2 dpa/FPY, and 184 appm/FPY, respectively. The peak FW dpa rate implies that the blanket lifetime is expected to be ~10 FPY. It is interesting to note that at the W/FS interface, atomic displacements and helium production in the FS are higher than those in W by factors of 3 and 38, respectively. Possible differential swelling at that interface needs to be assessed. The large helium production at the inner surface of the VV implies that rewelding will not be possible. We determined that the thickness of the VV cooled by 15% He should be at least 50 cm to allow for rewelding at its back anytime during the plant lifetime. The peak end-of-life helium production at the back of the 50 cm thick VV is 0.67 appm.
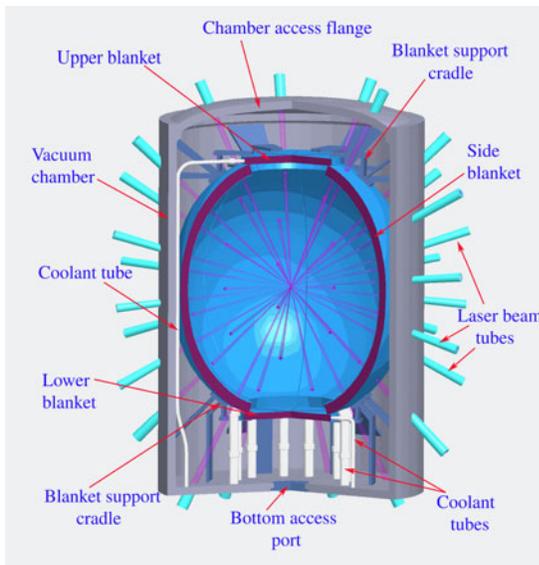

Figure 1.   Cross-section of HAPL chamber.

Moving away from mid-plane towards the top and bottom of chamber, the blanket thickness increases but the blanket sub-module width decreases resulting in an increased volume fraction of side walls. Only modest breeding is required from the top and bottom blankets that have a small coverage of ~5.8%. The top/bottom blankets are only 30 cm thick and include 20% Li. The overall TBR was determined to be 1.124 with only 0.024 contributed by the top/bottom. Therefore, tritium self-sufficiency can be achieved. The solid angle fraction subtended by the 60 beam ports is ~0.4% with minimal impact on the overall TBR. Nuclear heating profiles in the blanket components were determined and used in the thermal hydraulics analysis. Figure 4 shows the radial distribution at mid-plane. The peak power densities in FS, Li, and W are 14, 7, and 39 W/cm^3, respectively. The total thermal power is 2103 MW with 12.5% of it carried by the helium coolant of the VV. Only 112 MW of the thermal power is contributed by the top/bottom blanket. The total nuclear heating (deposited by

neutrons and gamma photons) in the blanket and VV is 1572 MW implying that the overall nuclear energy multiplication is 1.24. The blanket is coupled to a Brayton power cycle through a Li/He heat exchanger with an efficiency in the range of 42-45%.
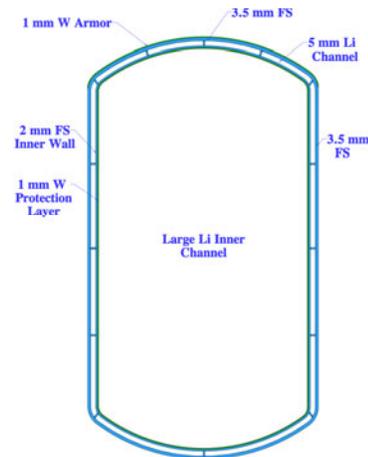

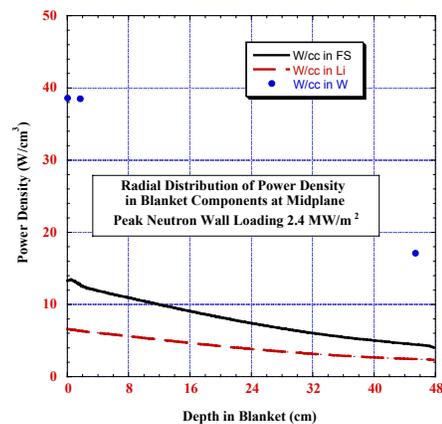Figure 2.   Schematic of self-cooled Li blanket sub-module.

Figure 3.   Radial variation of nuclear heating in components of Li blanket.

III. HELIUM-COOLED SOLID BREEDER BLANKET

This concept is based on a static solid breeder blanket design that is entirely cooled with He gas at 8 MPa. The design is similar to the EU He-cooled pebble bed blanket developed in Europe at FZK [6] and the blanket developed in the U.S. for the ARIES-CS compact stellarator power plant [7]. The blanket consists of a number of solid breeder (SB) and Be multiplier packed bed layers separated by cooling plates and arranged in parallel to the FW, as illustrated in the cross-section view shown in Fig. 4. Single-sized pebbles are assumed in both cases with a packing fraction of ~62%. Lithium ortho-silicate (Li$_4$SiO$_4$) is selected as the SB. The He gas, after cooling the FW, enters the breeding region and cools the SB and Be layers before exiting the blanket and going to the heat exchanger. The blanket cooling plates consist of 4 mm x 4 mm channels between two 1-mm thick plates. The distribution of the 60 beam ports is the same as in the Li-cooled blanket. The laser

beam tubes terminate at the VV and from there the laser light travels to the target without tubes. There are 48 side blanket modules with four beam ports in every fourth module. There is an upper and lower blanket with six beam ports in each.
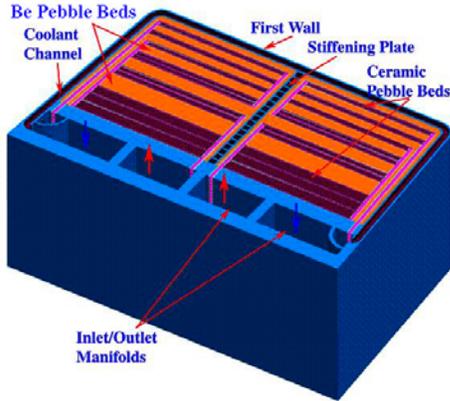

Figure 4.   Schematic of solid breeder blanket.

The configuration and thicknesses of the Be and SB regions (arranged in parallel to the FW) were optimized to ensure tritium self-sufficiency and maintain temperatures less than 750°C in Be and 950°C in SB. The total radial thickness of the blanket is 65 cm including a 20 cm thick zone for manifolds and support structure behind the breeding zone. A uniform enrichment of 40% ^{6}Li was found to be adequate leading to an adequate TBR of 1.18. The peak power densities in W, FS, Be, and SB are 67, 20, 16, and 48 W/cm^3, respectively. The radial variation of nuclear heating in the blanket components is given in Fig. 5. Moderate power densities from nuclear heating exist in the front layers of Be and SB ensuring that temperatures will not exceed the specified limits even if a uniform enrichment of 90% ^{6}Li is used. The total thermal power is 2302 MW with 48 MW of it generated in the VV. The total nuclear heating (deposited by neutrons and gamma photons) in the blanket and VV is 1723 MW implying that the overall nuclear energy multiplication is 1.36. The design has the potential for a power cycle conversion efficiency of 30-35% using the Brayton cycle. The peak damage and helium production rates in the FW steel at mid-plane are 20.1 dpa/FPY, and 183 appm/FPY, respectively. The peak FW dpa rate implies that the blanket lifetime is expected to be ~10 FPY. The peak end-of-life helium production at the back of a 30 cm thick VV is 0.4 appm allowing for rewelding.

IV. DUAL-COOLANT LITHIUM LEAD BLANKET

In the DCLL blanket design helium cools the ferritic steel FW and structure and is used for FW/blanket preheating and possible tritium control. The lithium lead (LiPb) eutectic $Li_{17}Pb_{83}$ is circulated at low speed for power conversion and tritium breeding. The concept is used in several MFE designs. It was first utilized in the ARIES-ST design [8]. It was then extensively studied by FZK as a candidate for the EU DEMO blanket [9]. Recently, it is being considered for the ARIES-CS power plant design [10]. The U.S. is developing a DCLL blanket test module to be tested in ITER [11]. The basic approach of the DCLL concept is shown in Fig. 6 [10]. A key element in the approach is the use of the SiC$_f$/SiC composite

(SiC-composite) flow channel insert (FCI) [9]. This FCI element performs the key function of providing thermal insulation to decouple the LiPb high temperature in the main channel from the low temperature FS structure, which is cooled by helium. In HAPL, the DCLL blanket is designed to cover the entire vertical length of the chamber with 12 modules. LiPb is admitted at the bottom of the blanket module, travels vertically upwards in a large channel behind the FW, then makes a U turn at the top, and travels down exiting the module on the bottom. He coolant connections are also made on the bottom.
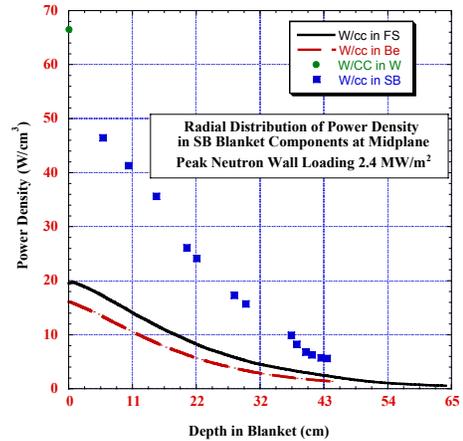

Figure 5.   Radial variation of nuclear heating in components of SB blanket.
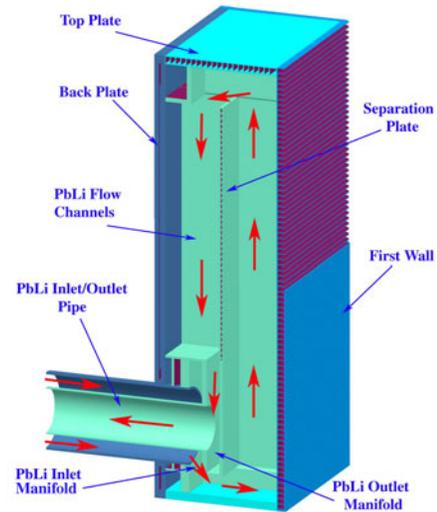

Figure 6.   Schematic of DCLL blanket concept.

Neutronics calculations were performed for the DCLL design with a total radial thickness of 52 cm at mid-plane. The Li in LiPb is enriched to 90% ^{6}Li leading to an overall TBR of 1.18. The peak power densities in W, FS, SiC, and LiPb are 44, 16, 12, and 34 W/cm^3, respectively. The radial variation of nuclear heating in the blanket components is given in Fig. 7. The total thermal power is 2096 MW with 123 MW of it generated in the VV. About 40% of the thermal power is removed from the blanket and VV by the He coolant with the rest being carried by LiPb. The total nuclear heating (deposited

by neutrons and gamma photons) in the blanket and VV is 1565 MW implying that the overall nuclear energy multiplication is 1.23. The design has the potential for a thermal efficiency of 40-45% using the Brayton cycle. The peak damage and helium production rates in the FW steel at mid-plane are 26.3 dpa/FPY, and 174 appm/FPY, respectively. The peak FW dpa rate implies that the blanket lifetime is expected to be ~7.6 FPY. The peak end-of-life helium production at the back of a 30 cm thick VV is 0.5 appm allowing for rewelding.
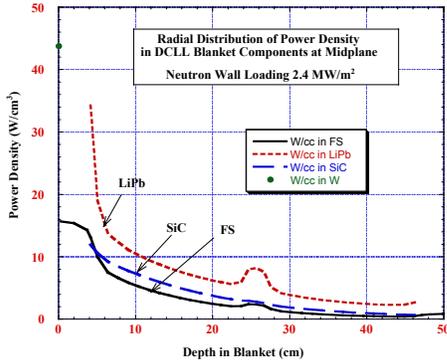

Figure 7. Radial variation of nuclear heating in the DCLL blanket.

V. COMPARISON OF BLANKET NUCLEAR FEATURES AND CONCLUSIONS

The main neutronics performance parameters for the three candidate HAPL blanket designs are compared in Table I. The three blankets have comparable TBR values >1.1 ensuring tritium self-sufficiency. There are design flexibilities to allow adjusting the TBR if needed. A thicker SB blanket with a significant amount of Be is required to achieve the required TBR. This is due to the poor tritium breeding capability of solid breeders and the large amount of structure required for the cooling plates between the many layers of SB and Be. While no lithium enrichment is required for the Li blanket, the SB and DCLL blankets require Li enrichment. The large amount of Be used in the SB blanket results in the highest nuclear energy multiplication yielding ~10% more thermal power. However, this is offset by the much lower thermal efficiency. In addition, the power density in the FW of the SB blanket is 20-40% higher than for the other blanket designs, which adds a burden on the FW cooling. While all of the thermal power is carried by the He in the case of the SB blanket, only 12%, and 40% is carried by He in the Li and DCLL blankets, respectively, with the rest being carried by the breeder. While the FW radiation damage (dpa) rates are similar for Li and SB, it is ~30% higher for the DCLL blanket resulting in a shorter blanket lifetime. Notice that the FW damage rate in IFE chambers are lower than those in MFE chambers with the same neutron wall loading. This is due to the fact that the source neutrons from the target impinge perpendicularly on the FW, resulting in less damage at the FW and larger damage at the back of the blanket compared to those in MFE chambers. Due to the poor shielding

capability of Li, a thicker VV is required with the Li blanket to allow rewelding at the back of the VV. For the three blanket designs, the VV is expected to be a lifetime component. Based on the neutronics results, the Li blanket seems to be the preferred option. However, other considerations should be accounted for in the blanket selection. Examples of issues to be considered are material compatibility, safety, tritium retention/control, thermal efficiency, design complexity, fabrication, weight, cost, development risk, and R&D cost.

TABLE I. NUCLEAR FEATURES OF CANDIDATE BLANKETS

	Li Blanket	SB Blanket	DCLL Blanket
Overall TBR	1.12	1.17	1.17
Blanket thickness (cm)	47	65	52
% ^{6}Li	natural	40%	90%
Total thermal power (MW)	2103	2302	2096
Power density in FW (W/cm^3)	13	20	16
Peak FW damage rate (dpa/FPY)	19	20	26
Peak dpa in VV @ 40 FPY	170	19	58
Blanket lifetime (FPY)	10	10	7
Required VV thickness (cm)	50	30	30
Thermal efficiency	~45%	~30-35%	~40-45%

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