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November 2006

UWFDM-1311

FUSION TECHNOLOGY INSTITUTE

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NEUTRONICS ANALYSIS OF A SELF-COOLED BLANKET FOR A LASER FUSION PLANT WITH MAGNETIC DIVERSION

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A blanket concept made of the low electrical conductivity SiC$_f$/SiC composite and utilizing Li$_{17}$Pb$_{83}$ as coolant and tritium breeder has been developed and integrated with the magnetic diversion system. Neutronics issues related to tritium breeding adequacy particularly with the area lost to the dump plates at the ring and point cusps were addressed. Radiation damage and lifetime considerations for the SiC$_f$/SiC structural material were also addressed. Another issue of concern is providing adequate shielding for the superconducting cusp magnets. Detailed neutronics analyses show that tritium self-sufficiency can be achieved. A 0.5 m thick water-cooled steel shield that doubles as the vacuum vessel is a reweldable lifetime component and will provide adequate shielding for the magnets.

I. INTRODUCTION

The High Average Power Laser (HAPL) program is carrying out a coordinated effort to develop laser inertial fusion energy (IFE) based on direct drive targets and a dry wall chamber.[1] The dry wall must accommodate the ion and photon threat spectra from the fusion micro-explosion over its required lifetime. The current HAPL strategy assumes as baseline a chamber with no protective gas and tungsten and ferritic steel as armor and structural materials, respectively. This results in a large chamber (~10.5 m radius) to ensure armor survival.[1] A parallel effort is underway to explore the option of using magnetic diversion to steer the ions emanating from the target away from the chamber wall.[2] A cusp magnetic field is imposed on the chamber and the ions from the micro-explosion are trapped within the magnetic field and are directed to more readily accessible and replaceable dump plates at the equator and poles. A large fraction of the magnetic energy can be dissipated in the chamber walls if an electrically resistive structural material is used.

An example blanket concept made of the low electrical conductivity SiC$_f$/SiC composite (required for dissipating the magnetic energy resistively) and utilizing Li$_{17}$Pb$_{83}$ eutectic as coolant and tritium breeder has been developed and integrated with the magnetic diversion system.[3] In this paper, we address the neutronics issues associated with the blanket design in the HAPL chamber with the magnetic diversion system.

II. CHAMBER AND BLANKET DESCRIPTION

The shape of the chamber differs from the earlier designs to better facilitate the directions of the diverted ions and the areas where they impact. Instead of being spherical, it consists of an upright cone on top of an inverted cone, with the mid-plane space reserved for a toroidal ring cusp dump. The apex of each cone has a polar cusp armored dump, which is exposed to some of the diverted ions. The blanket is divided into two segments on the top, and two segments on the bottom as shown in Fig. 1. One segment extends 5.8 m from the mid-plane, and is called the upper/lower mid blanket. It is followed by the second segment, which is called the upper/lower blanket and extends to the polar dump.
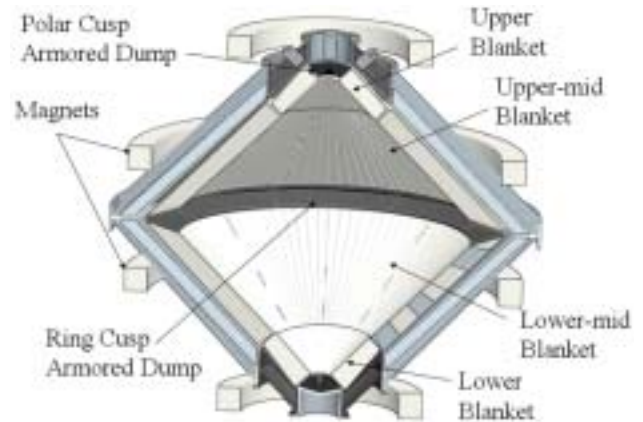

Fig. 1. Chamber configuration with magnetic diversion.

Each mid blanket consists of 16 modules, which in turn, consist of five sub-modules. The sub-modules are 47 cm wide and 70 cm deep at mid-plane, and 19.6 cm wide and 106 cm deep at the ends. The geometry consists of two concentric conduits forming an annular channel and a large inner channel. The first wall, annular channel and inner wall thicknesses are each of the order of 1 cm, the exact values are to be set from the results of the final integrated analysis including flow and stress considerations as well as on magnetic energy dissipation requirements. The coolant is admitted at the mid-plane and flows at a high velocity through the concentric channel, then makes a U-bend at the end of the sub-

module, and travels at a lower speed through the large inner channel at the end of which it exits. In this way, it cools the first wall at a high velocity, but on the return trip, at a lower velocity, absorbs nuclear energy and increases in temperature. The second segment of the blanket is similar to the first with slightly different dimensions. It is fed with coolant at the junction between the two segments.

III. CALCULATION PROCEDURE

The ONEDANT module of the DANTSYS 3.0 discrete ordinates particle transport code system[4] was used to perform the neutronics calculations utilizing the FENDL-2 nuclear data library.[5] 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. The reference HAPL target yield is 367.1 MJ.[6] For a repetition rate of 5 Hz, this corresponds to a total fusion power of 1836 MW. The target emits 1.4×10^{20} neutrons per shot with an average energy of 12.3 MeV. In addition, 1.7×10^{16} gamma photons with an average energy of 6.1 MeV are emitted from the target.

The neutron wall loading varies significantly along the first wall (FW) due to the large change in distance from target and incidence angle of source neutrons. The neutron wall loading, calculated analytically, peaks at a polar angle of 45°. For a 6 m chamber radius at mid-plane, the peak neutron wall loading is 6 MW/m^2 and the average value is 4.3 MW/m^2, as shown in Fig. 2. The neutron wall loading variation is accounted for in the neutronics results presented here.
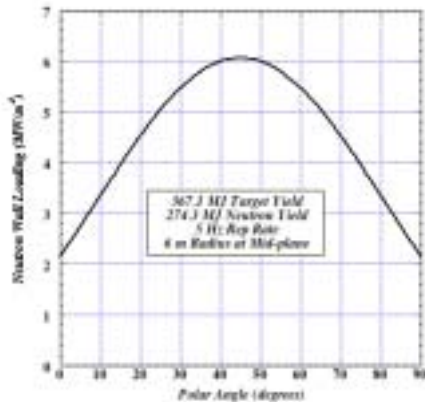

Fig. 2. Neutron wall loading variation.

IV. TRITIUM BREEDING

In order to ensure tritium self-sufficiency, the overall tritium breeding ratio (TBR) should exceed unity by an adequate margin to compensate for losses and radioactive decay of tritium, supply inventory for startup of other plants, provide reserve storage inventory, and account for uncertainties in nuclear data and modeling.[7] In this study, we require an overall TBR >1.1. The full angle subtended by the ring cusp and each of the point cusps is ~8.5°. As a result, the breeding blanket coverage lost by the ring cusp is 7.4% and that lost by the two point cusps is 0.3%. In addition, the 40 laser beam ports coverage is 0.7%. This leads to a total breeding blanket coverage loss of 8.4%. Therefore, the local TBR should be at least 1.2.

Figure 3 shows the effect of FW thickness on the local TBR for both cases with natural and enriched lithium. It is clear that tritium self-sufficiency cannot be achieved with natural lithium. With highly enriched Li, up to 2.5 cm FW can be used. With 90% ^{6}Li and ~1 cm thick SiC FW, the overall TBR is estimated to be ~1.25. While this exceeds the required value of 1.1, we selected that design for conservatism with the Li enrichment used as a knob to control the TBR if needed.
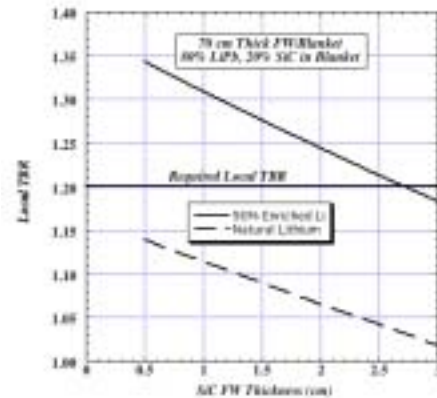

Fig. 3. Impact of FW thickness on TBR.

V. NUCLEAR HEATING

Nuclear heating profiles in the blanket components were determined and used in the thermal hydraulics analysis. Figure 4 shows the distribution as a function of depth in the blanket at the polar location with highest neutron wall loading. The peak power densities in LiPb and SiC are 89 and 31 W/cm^3, respectively. Figure 5 gives the SiC power density profiles at different polar locations. It is clear that significant variation in nuclear heating occurs in both the radial and polar directions. The blanket nuclear energy multiplication is 1.185.

In order to determine the total thermal power, the energy partitioning of the target as well as the blanket coverage fraction should be accounted for. Out of the total 367.1 MJ target yield, neutrons carry 274.3 MJ, gamma photons carry 0.017 MJ, x-rays carry 4.94 MJ, and ions carry 87.84 MJ.[6] Magnetic diversion of the ions results in 70% of the ion energy dissipated resistively in the blanket with the rest deposited at the cusp dump surfaces. The results were scaled by the 91.6% breeding blanket coverage to yield a total blanket thermal power of 1819

MW. This consists of 1489 MW volumetric nuclear heating, 307 MW volumetric ion energy dissipation, and 23 MW x-ray surface heating. The thermal power in the water-cooled 50 cm thick shield is only 11 MW. With 7.7% coverage, the total cusp dump thermal power is 240 MW including 106 MW volumetric nuclear heating, 132 MW ion surface heating, and 2 MW x-ray surface heating. If the energy deposited in the cusp dumps and shield is included in the power cycle, the total plant thermal power will be 2070 MW.
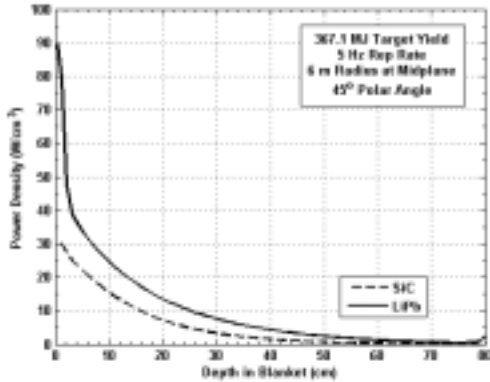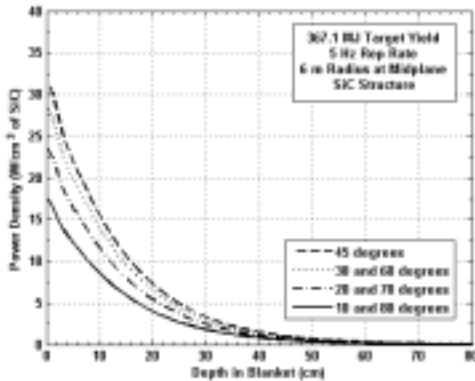

Fig. 4. Nuclear heating at 45° polar angle.

Fig. 5. Nuclear heating in SiC at different polar locations.

VI. RADIATION DAMAGE IN SiC STRUCTURE

The lifetime of the SiC_f/SiC composite material in the fusion radiation environment has been a major critical issue. The radiation effects in the fiber, matrix, and interface components of the composite material represent an important input for lifetime assessment. Neutronics calculations were performed to determine the radiation damage parameters for the SiC fiber/matrix and the candidate interface materials. The radiation damage parameters were calculated for both the carbon and silicon sublattices. The damage parameters calculated are the atomic displacement rate, the helium production rate, the hydrogen production rate and the total transmutation or burnup rate. The displacement energies of materials are dependent on the bonding. We determined the dpa cross

sections using the recommended average displacement energies for the Si and C sublattices of 40 and 20 eV, respectively.[8] The leading interface material candidates are graphite for near-term applications, and multilayer or porous SiC for longer-range applications. The damage parameters for the SiC interface material are identical to those for the SiC fiber/matrix. The damage parameters for the graphite interface material are the same as those for the C sublattice of SiC except for the dpa due to the higher (30 eV) displacement energy of C in graphite.[8] The SiC_f/SiC damage parameters were determined at the FW and as a function of depth in the blanket.

Table I gives the peak radiation damage parameters in the Si and C sublattices at a polar angle of 45°. It is interesting to note that the dpa values are ~10% lower than in magnetic fusion reactors with the same neutron wall loading and gas production and burnup values are lower by a factor of 2. This results from the softer energy spectrum and perpendicular incidence of source neutrons on the FW.[9] The results indicate that the dpa rate in the C sublattice is larger than in the Si sublattice of the SiC fiber/matrix. The dpa rate in the graphite interface material is 33% lower than in the C sublattice of the SiC.

TABLE I. Peak Damage Parameters in SiC_f/SiC

	C Sublattice	Si Sublattice	SiC	Graphite Interface
dpa/FPY	92	70	81	61
He appm/FPY	7844	2174	5009	7844
H appm/FPY	5	3900	1953	5
% Burnup/FPY	0.32	0.60	0.92	0.32

He production rate in the C sublattice of the SiC fiber/matrix and the graphite interface material is about a factor of 4 higher than in the Si sublattice of the SiC fiber/matrix. This is dominated by the (n,n´3α) reaction. The average He production rate in the graphite interface is 60% higher than the average He production rate in the SiC fiber/matrix. Significant hydrogen production occurs in the silicon with a negligible amount produced in the carbon. The H production rate in the graphite interface material and the C sublattice of the SiC fiber/matrix is more than three orders of magnitude lower than in the Si sublattice of the SiC fiber/matrix. This is due to the very high threshold energy of 13.6 MeV for the (n,p) reaction with carbon compared to 4 MeV for silicon. The burnup rate of the Si sublattice is twice that for the C sublattice of the SiC fiber/matrix and graphite interface material. The burnup is equivalent to introducing impurities in the sublattices of the SiC. Property degradation depends on the kind of impurities introduced. Transmutation of Si produces primarily Al with a smaller amount of Mg. The main transmutation product for C is Be with some Li produced from multiple neutron reactions. The damage parameters drop as one moves deeper in the blanket as illustrated in Figs. 6 and 7 for dpa and burnup rates.
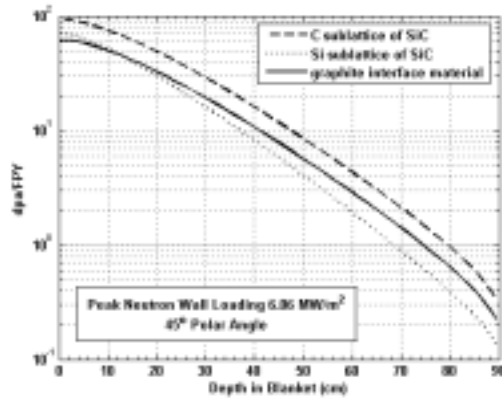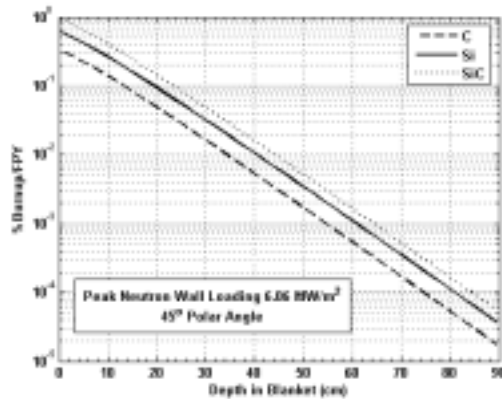
Fig. 6. Radial variation of dpa rate in the SiC.

Fig. 7. Radial variation of burnup rate in the SiC.

Lifetime considerations of SiC$_f$/SiC composite structure in fusion reactors have been addressed in a recent paper.[10] The useful lifetime of SiC$_f$/SiC composites in a fusion neutron environment can now only be speculated. Lifetime depends primarily on effects of He and metallic transmutants such as Al, Be, and Mg. The presence of free silicon degrades strength and enhances creep at high temperatures. The results presented here show that transmutation of silicon occurs at about twice the rate of carbon, producing excess carbon in the crystal. Unlike free silicon, carbon is not expected to segregate to grain boundaries and degrade properties. However, the other metallics produced by transmutation still may be problematic. Helium is the likely life-limiting factor for SiC. While it is clear that bubble formation due to helium will produce swelling in SiC, it is not known what the magnitude of the swelling will be, though it will certainly scale with burnup. Until this is known, an actual estimate of lifetime can only be a guess. If we consider an optimistic 3% burnup limit (corresponding to 260 dpa, 16300 He appm, and 6370 H appm), blanket lifetime is 3.26 FPY. However, a determination of the effect of fusion-neutron transmutations on the thermomechanical properties of SiC will be required for better assessment of SiC lifetime in the HAPL chamber.

VII. SHIELD RADIATION DAMAGE

A 50 cm thick shield that doubles as a vacuum vessel (VV) is used behind the blanket. The shield is made of steel and is cooled by 25% water. Two types of steel were considered; the austenitic steel 316SS and the low activation ferritic steel F82H.[11] The shield radiation damage parameters were determined at the location with highest neutron wall loading (45° polar angle and 90 cm thick blanket) and at the location with thinnest blanket (85° polar angle and 75 cm thick blanket). Damage parameters are slightly higher at 85° polar angle. Figures 8 and 9 give the radial variation of end-of-life (after 40 FPY operation) dpa and He production in the 316SS and ferritic steel shields, respectively. The peak end-of-life radiation damage in the shield is only ~5 dpa implying that it will be a lifetime component. He production in the 316SS shield is ~2 orders of magnitude higher than in the ferritic steel due to the large Ni content in 316SS. For the shield/VV to be reweldable, the helium production should not exceed 1 appm. It is clear that back of the shield/VV is reweldable. If ferritic steel is used, rewelding is possible at locations at least 5 cm deep in the shield. On the other hand, if 316SS is used, rewelding will be possible only at locations at least 30 cm deep in shield. If rewelding is required near the front of the shield/VV, use of ferritic steel is recommended.
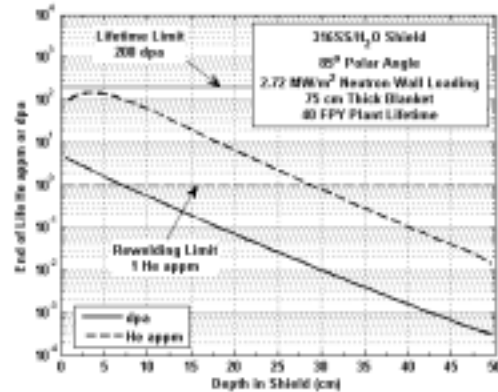

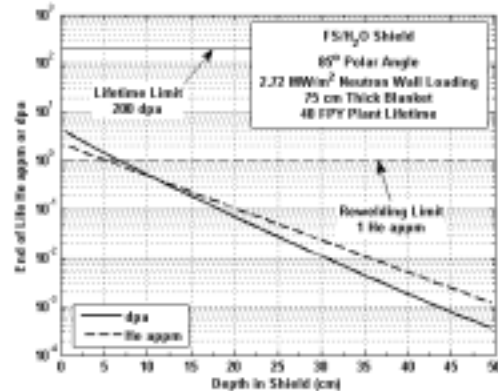Fig. 8. Radial variation of dpa and He in 316SS shield.

Fig. 9. Radial Variation of dpa and He in F82H Shield.

VIII. DAMAGE PARAMETERS IN CUSP COILS

The peak fast neutron (E>0.1 MeV) fluence in the superconducting cusp coils is limited to 10^{23} n/m^2 to avoid degradation in the critical current density of the Nb$_3$Sn superconductor. In addition, the peak dose in the polyimide insulator is limited to 10^8 Gy due to degradation of mechanical properties.[12] Calculations were performed to determine the peak damage parameters in the cusp coils at 45° and 85° polar angles with ferritic steel and 316SS shields. Using 316SS provides slightly better magnet shielding. The largest magnet damage occurs at 85° polar angle with ferritic steel shield/VV. At this location, the peak end-of-life (40 FPY) fast neutron fluence is 7.04×10^{21} n/m^2 and the end-of-life insulator dose is 2.30×10^7 Gy. The results indicate that the cusp coils are well protected with the 50 cm shield/VV and no restrictions should be imposed on the location of the coils from the shielding point of view.

IX. SUMMARY AND CONCLUSIONS

A blanket concept made of the low electrical conductivity SiC$_f$/SiC composite and utilizing Li$_{17}$Pb$_{83}$ as coolant and tritium breeder has been developed and integrated with the magnetic diversion system. Neutronics issues related to tritium breeding adequacy particularly with the area lost to the dump plates at the ring and point cusps were addressed. With 90% ^{6}Li in LiPb and ~1 cm thick SiC FW, the overall TBR is estimated to be ~1.25, ensuring tritium self-sufficiency. Li enrichment can be used as a knob to reduce the TBR if needed. At the 6 MW/m^2 peak neutron wall loading, the peak power density is 89 W/cm^3 in LiPb and 31 W/cm^3 in SiC. The total plant thermal power is 2070 MW if energy deposited in the dumps and shield is included in the power cycle. For a 3% SiC burnup limit (corresponding to 260 dpa, 16,300 He appm, and 6,370 H appm), the blanket lifetime is 3.26 FPY. However, a determination of the effect of fusion-neutron transmutations on the thermomechanical properties of SiC will be required for better assessment of the SiC lifetime in the HAPL chamber. The peak end-of-life radiation damage in the shield/VV is only ~5 dpa implying that it will be lifetime component. Although the back of the 0.5 m thick shield/VV is reweldable, it is recommended to use ferritic steel to allow rewelding near the front of the shield/VV. The superconducting cusp coils are well protected.

ACKNOWLEDGMENTS

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