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Abstract— **Neutronics analysis was performed for the reference design of the US dual coolant lead lithium (DCLL) ITER test blanket module (TBM). Detailed CAD models were utilized in the analysis. Relevant nuclear performance parameters were determined. These include tritium breeding, nuclear heating, radiation damage, transmutations, and shielding requirements. The calculated tritium breeding ratio (TBR) in the DCLL TBM is 0.561 and the total nuclear heating is 0.574 MW. For the ITER fluence goal of 0.3 MWa/m^2, the peak cumulative radiation damage and He production in the first wall (FW) are 5.1 dpa and 56 appm, respectively. Mg builds up in the SiC flow channel inserts (FCI) to ~100 appm with possible impact on electrical and thermal conductivities. About 1.2 m thick shield is required behind the TBM to allow personnel access for maintenance.**

Keywords-test blanket; lead lithium; nuclear heating

I. Introduction

In support of the ITER Test Blanket Module (TBM) program [1], the US has been developing a TBM design based on the dual coolant lead lithium (DCLL) blanket concept [2]. The basic idea of the DCLL blanket is to use helium to remove all heat deposited in the first wall (FW) and blanket structure, and a flowing, self-cooled, lead lithium (PbLi) breeder to remove nuclear heat generated in the breeding zone at a high temperature for efficient power conversion [2]. This is the preferred US blanket concept for commercial fusion plants.

The concept consists of PbLi channels contained within a helium-cooled structure made of reduced activation ferritic steel (RAFS). Each PbLi channel is lined with a SiC flow channel insert (FCI) that separates the PbLi from the RAFS structure. This FCI performs two important functions: (a) thermally insulates the PbLi so that its temperature can be considerably higher than the surrounding structure, and (b) provides electrical insulation between the PbLi flow and the thick, load-bearing RAFS walls to reduce the MHD pressure drop. The concept will be tested in one half of a designated test port where it will be mounted inside a water-cooled frame. Many design issues were considered in determining the configuration of the TBM. That includes both PbLi and He manifolding and flow path arrangement.

The design has been evolving over the past several years following several technical reviews and it converged on a reference design for which detailed CAD models were generated. This included changes in the overall dimensions of the TBM to 166 cm height, 48.4 cm width, and 35 cm thickness. These dimensions are smaller than the previous design due to the increased frame thickness. In addition, the reference TBM design utilizes a flat front surface. The PbLi flow was reversed starting at the top flowing downward in the back channel then upward in the front channel. The PbLi concentric pipe concept was replaced by two individual pipes. Helium flow in the FW has two circuits with 7 passes per circuit. Grid plates are used to route the flow radially. Fig. 1 shows the overall DCLL TBM configuration with exploded view showing the internal components.

Neutronics calculations were presented previously for an initial preliminary DCLL TBM design [3]. In this work, detailed nuclear analysis was performed for the reference design configuration based on the detailed CAD drawings. The relevant nuclear performance parameters for the reference DCLL TBM design are given. These include tritium breeding, nuclear heating, radiation damage, and shielding requirements.

II. Calculation Procedure

The neutron wall loading at the TBM is 0.78 MW/m^2. The front surface of the module has a total surface area of 0.8 m^2. A 2 mm thick beryllium layer is utilized as a plasma facing component (PFC) material on the ferritic steel FW. The lithium in the lead lithium ($Li_{17}Pb_{83}$) is enriched to 90% Li-6. 5 mm thick SiC flow channel inserts (FCI) are used at the walls of all PbLi flow channels. The ferritic steel alloy F82H is used for structural material [4]. The PARTISN 4.0 [5] discrete ordinates particle transport code was used to perform the calculations utilizing the FENDL-2.1 [6] nuclear data library. Both the inboard (IB) and outboard (OB) regions were modeled simultaneously to account for the toroidal effects. The IB shielding blanket is modeled with its radial configuration including the Be tiles and Cu heat sink. A separate 316SS/H_2O shield (75% 316SS and 25% H_2O) is used behind the TBM.

Since the material configuration inside the TBM varies significantly in the vertical direction, the TBM is divided into seven vertical (poloidal) layers. These sections are determined in order to maintain a uniform vertical configuration in each vertical layer. For each poloidal layer the radial zones are homogenized by determining the volume fractions of each material in the zone. The material volume fractions for each zone were then used in the neutronics calculations to determine the nuclear parameters throughout the corresponding poloidal layer. Results for the 7 layers were combined using their heights to determine the overall integrated parameters for the TBM. These layers with their heights are shown in Fig. 1. Fig. 2 shows the configuration of the mid-plane layer (layer 5) that represents 77% of the TBM and Table I gives the radial build for that layer.

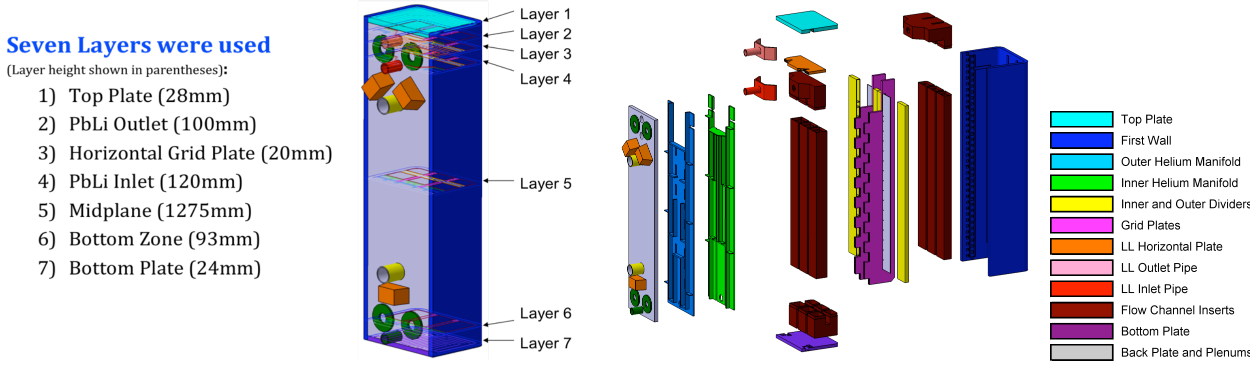

Figure 1. DCLL TBM configuration with exploded view.

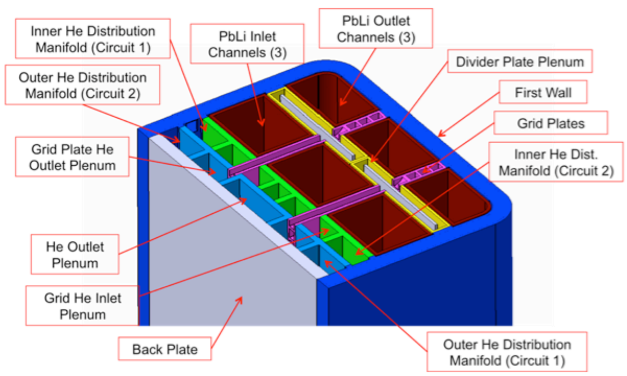

Figure 2. Configuration at mid-plane of TBM.

TABLE I. RADIAL BUILD AT MID-PLANE OF TBM

Zone Description	Thick (mm)	% Be	% FS	% LL	% SiC	% He
PFC Layer	2	100	0	0	0	0
Front wall of FW	4	0	100	0	0	0
FW cooling channel	20	0	17	0	0	83
Back wall of FW	4	0	100	0	0	0
FCI layer 1	7	0	6.3	24.7	55.4	13.6
Front breeding channel	66	0	6.3	73.9	6.2	13.6
FCI layer 2	7	0	6.3	24.7	55.4	13.6
Front wall of divider	4	0	86.4	0	0	13.6
Divider gap 1	10	0	31	0	0	69
Plenum layer	4	0	82.1	0	0	17.9
Divider gap 2	10	0	31	0	0	69
Back wall of divider	4	0	86.4	0	0	13.6
FCI layer 3	7	0	6.3	24.7	55.4	13.6
Back breeding channel	86	0	6.3	73.9	6.2	13.6
FCI layer 4	7	0	6.3	24.7	55.4	13.6
Inner He manifold	10	0	86.4	0	0	13.6
Inner He channel	30	0	12.6	0	0	87.4
Outer he manifold	10	0	88.4	0	0	11.6
Outer He channel	30	0	10.2	0	0	89.8
Back plate	30	0	90.4	0	0	9.6

III. TRITIUM BREEDING

The calculated local tritium breeding ratio (TBR) in the TBM is only 0.561 because of the relatively small thickness used. During a D-T pulse with 500 MW fusion power, tritium is produced in the TBM at the rate of 7.73×10^{-7} g/s. The peak tritium production rate in PbLi is 2.3×10^{-8} kg/m^{3}s. For a pulse with 400 s flat top preceded by 20 s linear ramp up and 20 s linear ramp down, the total tritium generation in the TBM is 3.25×10^{-4} g/pulse. For the planned 3000 pulses per year the annual tritium production is 0.97 g. The tritium inventory in the TBM at any time will be much smaller since tritium will be continuously extracted from the PbLi. Tritium production rate in the Be PFC is only 1.54×10^{-9} g/s during the 500 MW D-T pulse with total annual generation of 1.94×10^{-3} g representing only 0.2% of the total tritium production in the TBM.

IV. NUCLEAR HEATING

Nuclear heating radial profiles in the different constituent materials at the 7 poloidal layers were determined for use in thermal hydraulics analysis. The results are shown in Fig. 3 for PbLi, SiC, and ferritic steel at mid-plane. Table II compares the peak power densities in the constituent materials. The nuclear energy multiplication in the TBM is 0.92. The neutron power incident on the TBM front surface is 0.624 MW during the 500 MW D-T pulse. This results in total nuclear heating of 0.574 MW in the TBM. The breakdown of nuclear heating and tritium breeding in the 7 vertical layers is provided in Table III.

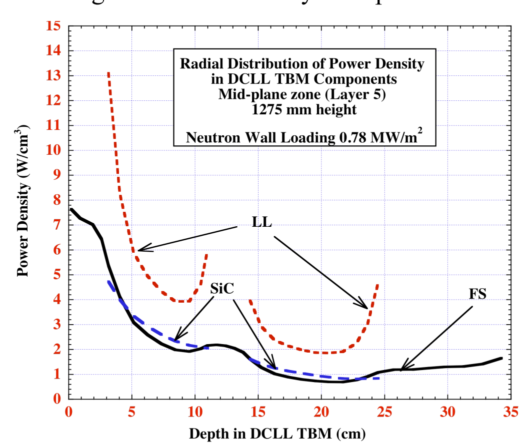

Figure 3. Radial distribution of power density at mid-plane of TBM.

V. STRUCTURE RADIATION DAMAGE

The radial variations of the atomic displacement (dpa), helium production, and hydrogen production rates in the ferritic steel structure of the DCLL TBM at mid-plane are shown in Fig. 4. The results are given for the 0.78 MW/m^2 neutron wall loading corresponding to the 500 MW D-T pulse. For the average ITER neutron wall loading of 0.57 MW/m^2 and the total fluence goal of 0.3 MWa/m^2, the total full power lifetime is 0.526 FPY. The peak cumulative end-of-life radiation damage in the FW is 5.1 dpa and the peak end-of-life helium production is 56 He appm.

TABLE II. PEAK POWER DENSITIES (W/CM^3) IN TBM MATERIALS

Constituent Material	Peak Power Density (W/cm^3)
Be PFC (layer 2)	8.14
Ferritic Steel (layers 1,7)	9.20
PbLi (layer 3)	13.20
SiC (layer 2)	4.79

TABLE III. BREAKDOWN OF NUCLEAR HEATING AND TRITIUM BREEDING

Vertical Layer	Height (mm)	Nuclear Heating (MW)	Local TBR
1	28	0.009	0
2	100	0.035	0.773
3	20	0.007	0.338
4	120	0.042	0.662
5	1275	0.441	0.560
6	93	0.032	0.625
7	24	0.008	0
Total	1660	0.574	0.561

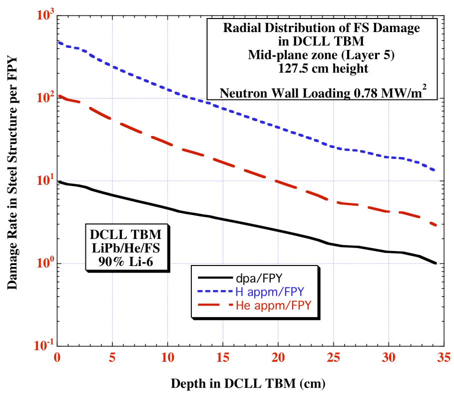

Figure 4. Radial variation of damage rates in steel structure at mid-plane.

VI. FCI RADIATION DAMAGE

Neutronics calculations were performed to determine the radiation damage parameters for the SiC fiber/matrix and the candidate interface materials in the SiC_f/SiC composite FCI. The leading interface material candidates are graphite and multilayer SiC. The radiation damage parameters were calculated for both the carbon and silicon sublattices. We used the recommended average displacement energies for the Si and C sublattices of 40 and 20 eV, respectively [7]. 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.

Table IV gives the peak damage parameters in the front layer of the FCI. The results indicate that the dpa rate in the C sublattice is 17% 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. The 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 due to the (n,n′3α) reaction. The average He production rate in the graphite interface is 60% higher than that in the SiC fiber/matrix. Significant hydrogen production occurs in the silicon with a negligible amount produced in the carbon. The burnup rate of the Si sublattice is twice that for the C sublattice of the SiC fiber/matrix and graphite interface material. The nonstoichiometric burnup of Si and C is expected to be worse than stoichiometric burnups and could be an important issue for lifetime assessment. The damage parameters drop as one moves deeper in the TBM as illustrated in Fig. 5. The peak cumulative end-of-life damage parameters in the FCI are 4.8 dpa, 409 He appm, 153 H appm, and 0.073% burnup.

TABLE IV. PEAK DAMAGE PARAMETERS IN SiC_F/SIC FCI

	C Sublattice	Si Sublattice	SiC	Graphite Interface
dpa/FPY	9.86	8.41	9.14	6.57
He appm/FPY	1235	320	777	1235
H appm/FPY	0.2	583	291	0.2
% Burnup/FPY	0.05	0.09	0.14	0.05

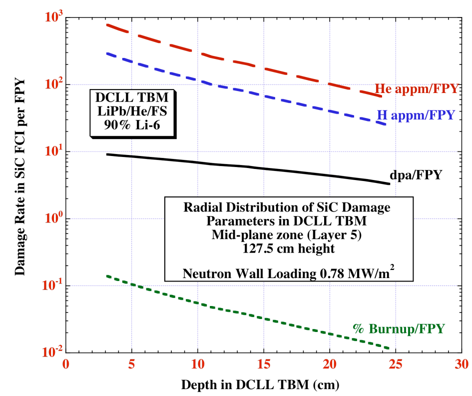

Figure 5. Radial variation of damage rates in the SiC FCI.

Since the FCI is not a structural component, the main concern is degradation in its primary role as electrical and thermal insulator. Degradation in resistivity results from the introduction of transmutation products. We performed transmutation calculations using the ALARA code [8] to determine the rate of buildup of the different transmutation products. Fig. 6 shows the concentration of the different metallic transmutation products in the front layer of the FCI as a function of operation time. The dominant metallic transmutation product is Mg that builds up to ~100 appm at end-of-life of ITER. Other transmutation products with significant buildup are Be and Al. Although these are small concentrations for the low ITER fluence, we estimate that Mg will build up to a significant amount of ~0.5 at% at the expected lifetime (~20 MWa/m2) of a DCLL blanket in a fusion power plant. It is essential to assess the impact of these levels on electrical and thermal conductivities of the FCI to determine if it will restrict the lifetime of the DCLL to less than that determined by radiation damage to the ferritic steel FW.

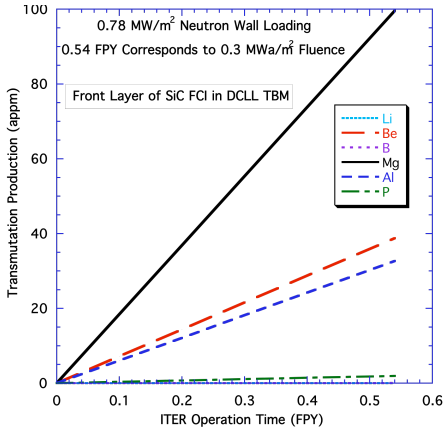

Figure 6. Buildup of transmutation products in the SiC FCI.

VII. SHIELDING REQUIREMENT

The required size of the shield behind the TBM is determined primarily by the need to have hands-on access behind it after shutdown for disconnecting components. The criterion used in ITER is that the dose rate should not exceed 100 μSv/h at 10^6 s after shutdown. Rules of thumb can be used to relate the dose after shutdown, from decay gamma of activated shield and outlying components, to the fast neutron flux during operation. Formulas used in the ITER Nuclear Analysis Report, G 73 DDD 2 W 0.2, July 2004, relate the dose rate at 10^6 s after shutdown to the fast neutron flux during operation as DR(μSv/h) ~1-3x10^{-5} FF(n/cm^{2}s). To be conservative, we used the factor of 3x10^{-5} and the flux with E>0.1 MeV for the fast flux. Hence, the fast neutron flux should not exceed ~3x10^6 n/cm^{2}s. Fig. 7 gives the effect of shield thickness on the fast neutron flux behind it. Based on these results we estimate that ~1.2 m thick shield is required behind the DCLL TBM. This needs to be confirmed by performing detailed activation analysis that accounts for streaming in the shield penetrations.

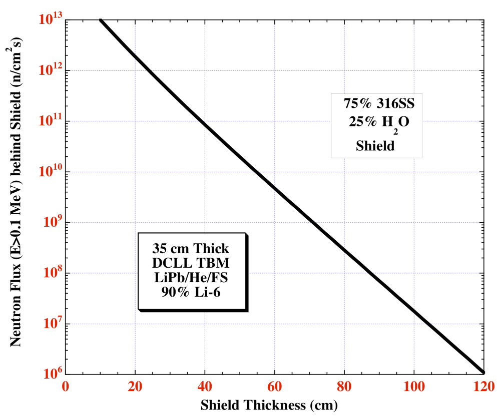

Figure 7. Variation of fast neutron flux with shield thickness.

VIII. SUMMARY AND CONCLUSIONS

Neutronics calculations were performed to determine the relevant nuclear performance parameters for the reference US DCLL ITER TBM. These include tritium breeding, nuclear heating, radiation damage and transmutations, and shielding requirements. The neutron wall loading at the TBM is 0.78 MW/m^2. The front surface area of the module is 0.8 m^2 and the radial depth is 35 cm. The detailed CAD model was utilized to divide the TBM into 7 vertical layers and perform calculations for each with the appropriate radial build. Results for the 7 layers were combined using their heights to determine the overall integrated parameters for the TBM.

The calculated tritium breeding ratio (TBR) in the DCLL TBM is only 0.561 because of the relatively small thickness used. For the planned 3000 pulses per year the annual tritium production in the TBM is 0.97 g. The total nuclear heating in the TBM is 0.574 MW. For the ITER fluence goal of 0.3 MWa/m^2, the peak cumulative radiation damage and He production in the FW are 5.1 dpa and 56 appm, respectively. The corresponding end-of-life values for the SiC FCI are 4.8 dpa and 409 He appm. The dominant metallic transmutation product in the FCI is Mg that builds up to ~100 appm at end-of-life of ITER and its impact on electrical and thermal conductivities of the FCI need to be assessed particularly at elevated fluences in a fusion power plant. We estimated that ~1.2 m thick shield is required behind the DCLL TBM to allow personnel access for maintenance.

Detailed 3-D analysis using the DAG-MCNP code is in progress. The overall ITER configuration with correct source distribution is used to generate a surface at the front surface of the TBM port that is used in subsequent calculations with the detailed TBM and frame CAD model.

ACKNOWLEDGMENT

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