AN OVERVIEW OF THE US DCLL ITER TBM PROGRAM

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ABSTRACT

Under the US Fusion Nuclear Science and Technology program, we selected the Dual Coolant Lead Lithium (DCLL) concept as our primary Test Blanket Module (TBM) for testing in ITER. The DCLL blanket concept has the potential to be a high-performance DEMO blanket design with a projected thermal efficiency of >40%. Reduced activation ferritic/martensitic (RAF/M) steel is used as the structural material. Helium is used to cool the first wall and blanket structure, and the self-cooled Pb-17Li breeder is circulated for power conversion and for tritium extraction. A SiC-based flow channel insert (FCI) is used as an electrical insulator for magnetohydrodynamic pressure drop reduction from the circulating Pb-17Li and as a thermal insulator to separate the high-temperature Pb-17Li (~650°C to 700°C) from the RAF/M structure, which has a corrosion temperature limit of ~480°C. The RAF/M material must also operate at temperatures above 350°C but less than 550°C. We are continuing the development of the mechanical design and performing neutronics, structural and thermal hydraulics analyses of the DCLL TBM module. Prototypical FCI structures were fabricated and further attention was paid to MHD effects and the design of the inboard blanket for DEMO. We are also making progress on related R&D needs to address key areas. This paper is a summary report on the progress and results of recent DCLL TBM development activities.

1. INTRODUCTION

In support of the ITER Test Blanket Module (TBM) program, we have been focusing on the dual coolant Pb-17Li liquid breeder (DCLL) blanket design, a concept that has been explored extensively in the US [1,2] and the European Union [3]. With the use of reduced activation ferritic/martensitic (RAF/M) steel as the structural material, we are limited to a maximum steel structure temperature of <550°C and a corrosion limit of ~480°C. At the same time, we need to remove the first wall (FW) heat flux, breed adequate tritium for the D-T fuel cycle and achieve high coolant outlet temperature for high power conversion efficiency when the design is used for DEMO. A preliminary DCLL TBM design was completed in 2005 [1]. A key element in the approach is the use of the SiC flow channel insert (FCI). This FCI element performs a key function of reducing the magnetohydrodynamic (MHD) pressure drop associated with the circulating self-cooled Pb-17Li breeder and thermally insulating the high temperature Pb-17Li from the low-temperature RAF/M steel structure, which is cooled by helium. The Pb-17Li is flowing in the poloidal channels at about 10 cm/s to achieve high outlet coolant temperature. This paper presents the continuing development of the DCLL blanket concept, with focus on its validity for the DEMO design and addressing the following topics: improving the design of the DT phase DCLL TBM module, addressing key MHD-related issues for DEMO and FCI development, and with support from basic research and development of key areas.

2. DCLL TBM DESIGN AND ANALYSIS

We continue the development of the DCLL-TBM FW and blanket design for the D-T testing phase module [Fig. 1(a)] for ITER based on R&D results and with the goal that the concept can provide adequate performance when it is extrapolated to DEMO [4]. DCLL TBM design parameters are given in Table 1. Since 2005, design changes were made on the mechanical design to minimize the natural convection impact from the Pb-17Li flow [5]. The revised DCLL TBM vertical section view through the Pb-17Li channels is shown in Fig. 1(b). It has a two-pass poloidal Pb-17Li flow. The Pb-17Li enters from the upper part of the module from the back, flows down the inside Pb-17Li channel and turns at the bottom of the module before flowing up the outside channel to mitigate the risk of "hot spots" due to reverse fluid currents caused by the flow opposing buoyancy forces. Separate inlet and outlet pipes are used for both Pb-17Li and helium coolants circulating in and out of the TBM module for convenience. For the DEMO design in order to minimize thermal and tritium losses, concentric pipes are still the recommendation [4]. As shown in Fig. 1(c), the helium coolant is designed to pass the FW multiple times with two streams coming in from the bottom of the module and flowing left and right towards the FW in counter flow while cooling the FW in the toroidal direction before leaving the first wall panel to cool the rest of the RAF/M steel structure. In order to demonstrate the applicability of the FW design to the DEMO design, we continue to recommend that the first wall cooling channel surface on the plasma side should be designed and fabricated with a one-sided 2-D roughened surface in order to enhance the local heat transfer coefficient and minimize the helium coolant pressure drop. At the same time, detailed changes were made on the routing of the helium for the cooling of the internal steel structure.
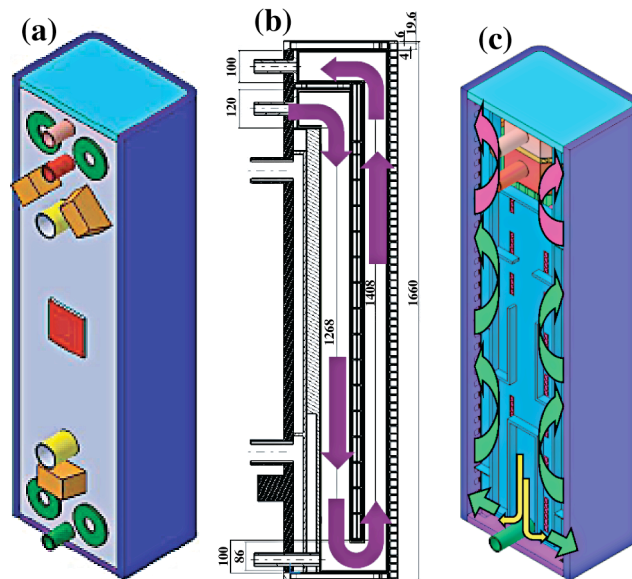

Fig. 1. (a) DCLL TBM module. (b) Poloidal Pb-17Li flow, inlet and outlet channels located at the top (c) Helium flow distribution at the back to the first wall, inlet located at the bottom and outlet located at the top of the module.

3-D neutronics calculations were performed to determine the important nuclear performance parameters, including volumetric nuclear power distributions in the module. A summary of this work is presented in the next section. With the 3-D neutronics results, we performed thermal hydraulics analysis to determine the suitable design for uniform helium and Pb-17Li flow distributions. We continue to perform the FW heat transfer and structural analysis. We have been applying the ANSYS-CFX code for our fluid flow distribution assessment. Concurrently, we are applying the CRADLE code for comparison. For thermo-structural analysis, the TBM CATIA model was used to create a detailed solid model. TBM materials, loads and back support boundary conditions are used. Initial thermal analysis has identified hotspots along the top TBM edge and a total displacement of ~10.4 mm under the set of TBM operating parameters. Minimization of displacement and identified stress concentrations will be addressed with the next design iteration. At the same time, safety analysis is being performed as an ITER requirement.

2.1. 3-D NEUTRONICS

We completed the detailed 3-D neutronics analysis for the US DCLL TBM design. Calculations were performed for the exact CAD model using the DAG-MCNP code [6] with all geometrical details of the DCLL TBM preserved. Detailed 3-D analysis of the TBM with the surrounding massive water-cooled frame and representation of exact source and other in-vessel components yields lower TBM nuclear parameters than for the 1-D estimates. Results were generated for volumetric nuclear heating, tritium breeding and generation, and material radiation damage (helium production and atomic displacement). In summary, the tritium generation rate in the Pb-17Li is 4.19×10^{-7} g/s during a D-T pulse with the 500 MW fusion power of ITER. For the planned 3000 pulses per year the annual tritium production in the TBM is 0.53 g/year with only 1.04×10^{-3} g/year produced in the Be plasma facing surface. Total nuclear heating in the TBM is 0.374 MW and the peak cumulative end-of-life damage parameters are 3.7 dpa and 51 He appm [7].

2.2. MHD EFFECTS

Details of the Pb-17Li flow design were analyzed for the DCLL DEMO blanket and can be represented in four key areas. The first one is on the buoyancy-assisted and buoyancy-opposed flows. Results show that buoyancy opposed flows can lead to instability of the fluid flow and flow reversal near the "hot" wall, this leads to the selection of the present flow configuration of the Pb-17Li in the TBM. The second is on the impact from non-perfect wetting and associated interfacial slip between the SiC and Pb-17Li. Two fundamental MHD problems (Hartmann flow and fully developed flow in a rectangular duct) have been solved analytically for poor-wetting conditions. A strong effect of poor-wetting on the flow structure and MHD pressure drop reduction can be predicted in the blanket flows even though the slip length is small (~1–10 μm) [8]. The third area is on the design of the SiC FCI. In order to avoid excessive thermal stress on

the 0.5 cm thick single layer flow channel insert, two FCI layers called "nested FCI" design were introduced as shown in Fig. 2 [9]. The outer layer with unbonded SiC plates will handle most of the thermal insulation with minimized thermal stress, the inner FCI channel is for the reduction of MHD pressure drop of the flowing Pb-17Li.   The fourth area is the modeling of hydrodynamic instability due to mixed convection in the two Pb-17Li poloidal channels. Results show that the fluctuation of the downward flow is much higher than that in the upward flow, but both fluctuations seem to be acceptable for the design.
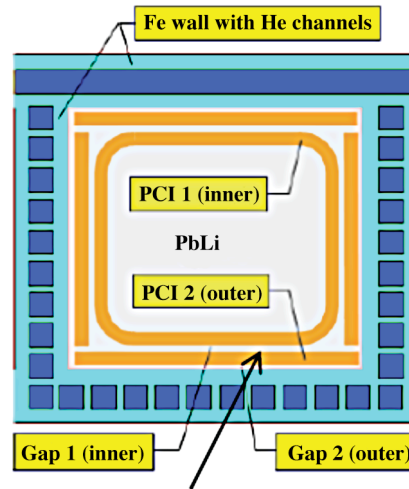

Fig. 2. Two layer SiC FCI (nested FCI).

At the same time, experiments [10] and 3-D MHD simulations [11] demonstrated that nearly uniform flow rates of liquid metal can be reached among DCLL parallel channels when a strong magnetic field is applied in the direction in which the parallel, insulated channels are stacked. The mechanism responsible for this effect is the strong diffusion of the velocity component parallel to the magnetic field leading to some momentum redistribution and, ultimately, to a quasi two-dimensional flow distribution when the channel walls are poorly conducting or insulated. This finding has positive implications for the DCLL DEMO blanket and TBM design as it simplifies the design and helps to eliminate any strong non-uniformity in flow that can lead to poor heat transfer and locally overheated channels. Future work is needed to discover how effective or sensitive this MHD process is when the field is inclined to the walls, the insulation is not perfect, or upstream conditions are not uniform between parallel channels, such as could result from imperfections in the channel insulation.

3. DESIGN OF THE INBOARD COOLANT ROUTING

With our continuing approach for evaluating the DCLL blanket concept for DEMO application, we analyzed the inboard (IB) DCLL blanket (Fig. 3) design, which would operate under twice the magnetic field strength of the outboard blanket and with focus on the impacts from helium and Pb-17Li flow channels of the IB side of the tokamak [12]. Results show that the MHD pressure drop in the IB DCLL blanket under DEMO conditions is ~1.43 MPa. Most of the pressure drop originates from the 3-D flows in the blanket inlet/outlet manifolds and those in the access ducts due to the strong magnetic field gradients. However, the correlations used in the calculations of the 3-D MHD pressure drop are based on empirical data and included in our subjective choice of the empirical coefficient k for the manifolds, typically, $0.25 < k < 2$. Therefore, the calculated MHD pressure drop has an uncertainty of ±30%. More accurate assessments will be needed in the future.
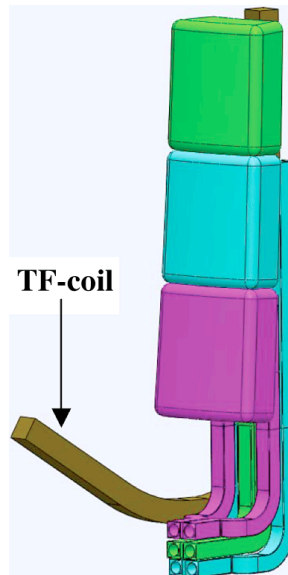

Fig. 3. Schematic of three inboard DCLL blanket modules and corresponding concentric inlet and outlet channels for He and Pb-17Li coolants.

4. SiC FLOW CHANNEL INSERT DEVELOPMENT

Another key element in the DCLL concept is the SiC FCI, with the possibility of tailoring the electrical and thermal properties of the SiC material by controlling the manufacturing process of the composite. For the SiC FCI material, the goal is the nearly zero thermal leakage condition with thermal conductivity of 2 W/m.K across the plate, and to reduce the MHD pressure drop in the poloidal ducts by a factor of 50–100 times compared to non-insulated ducts. The corresponding recommended electrical conductivity is $\sigma < 20 - 50$ 1/Ω-m [13]. In order to reach these values, two development paths are implemented. One is the use of fiber-reinforced SiC composite material. Correspondingly, we are looking into radiation effects with very high temperatures and high dose, including mechanical properties, creep, and microstructure properties. We are continuing the modeling for composite properties, fracture, and life prediction, including the study of corrosion between SiC and Pb-17Li [14]. The other development path is the use of foam-based SiC as shown in Fig. 4 [15]. Samples and mockup FCI components were fabricated with closed and open-cell foams. Corresponding thermal, electrical and mechanical properties were measured. Measured thermal conductivities of ~1–2 W/m.K in the temperature range of 100°C to 800°C were achieved [16]. Figure 4 shows an assembly of two FCI segments with a total height of 750 mm and a width of 130 mm [17], and electrical conductivities of less than 20 1/Ω-m were measured as well [14].
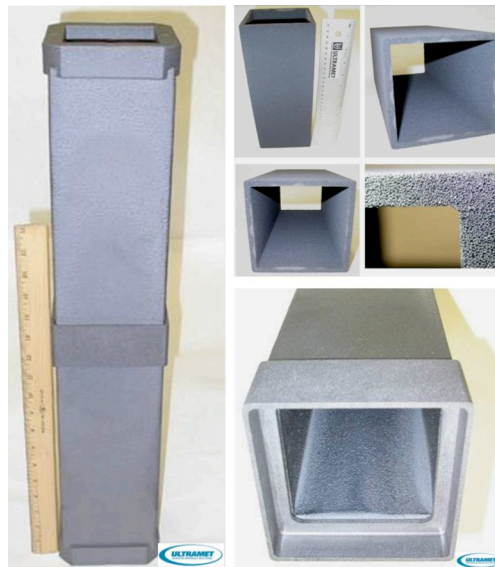

Fig. 4. FCI Prototype consisting of an assembly of two FCI segments totaling a height of 635 mm, 116 mm wide, and 116 mm deep; the SiC-foam was infiltrated by CVD SiC to ~10–20 vol % (Ultramet, Inc.).

5. ONGOING R&D ACTIVITIES

For the development of the DCLL blanket design, we continue research and development in the following areas, which are not presented in this paper:

- Tritium permeation and recovery

- Safety analysis and modeling

- Irradiation effects in RAF/M steels

- Integrated modeling/Virtual TBM

- Diagnostics

Some of these areas of research are enhanced with international collaborations under the IEA, TITAN and ITER-TBM programs and further collaborations are encouraged.

6. CONCLUSIONS

We continue the design and development of the DCLL blanket concept as the FW and blanket concept for the DT phase of the ITER-TBM program. TBM mechanical design and analysis and 3-D neutronics analysis were presented. Key results of MHD analysis were shown including the impact from buoyancy effect, effects of non-perfect wetting between SiC and Pb-17Li, a new two-layer SiC-based FCI design along with flow fluctuation in the two parallel Pb-17Li flow channels. Experimental and simulation results of Pb-17Li manifold were summarized. The design of the IB flow design in a DEMO plant was also assessed and an acceptable total MHD pressure drop of ~1.43 MPa was estimated, which is mostly due to the flow manifolds and the strong magnetic field gradient. SiC flow channel inserts are being developed based on fiber-reinforced SiC composite architectures and the alternative of closed and open-cell foams. Other fundamental R&D activities are on-going, including tritium extraction and permeation, safety analysis, irradiation effects on RAF/M steel, virtual TBM and diagnostics.

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