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Structural design and preliminary analysis of liquid lead–lithium blanket for China Fusion Engineering Test Reactor

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ABSTRACT

China Fusion Engineering Test Reactor (CFETR) has been proposed as an option in China to bridge the gaps between ITER and fusion power plant. Since one major goal of CFETR is to demonstrate long pulse or steady-state operation with duty cycle time ≥0.3–0.5, easier maintenance of the in-vessel components is emphasized in the design process. In this contribution, a kind of liquid lead–lithium tritium breeder blanket concept focus on the remote maintenance has been designed for CFETR. To make the pipes and mechanical connections at the rear of the blanket accessible from vacuum vessel, two kinds of guide tubes were adopted to provide passageways for remote handling tools. In order to evaluate the effects of the guide tube installation on the structural performance of the blanket, as a preliminary stage, thermal-hydraulic analysis of first wall was carried out based on the heat load obtained from 3D modeled neutronics calculations. In addition, thermal stress analysis of the first wall under normal condition was performed to evaluate the thermomechanical behavior. The preliminary analysis results validated the performance of current blanket design.

1. Introduction

China Fusion Engineering Test Reactor (CFETR) [1] was proposed as one of the options in China to bridge the gaps between ITER and fusion power plant. Currently two tokamak options (superconducting and water-cooling copper magnets) for CFETR are under study. Since one major goal of CFETR is to demonstrate long pulse or steady-state operation with duty cycle time ≥0.3–0.5, to insure the duty cycle of CFETR, easier maintenance of the in-vessel components is emphasized in the design process. Several vacuum vessel (VV) schemes and the corresponding remote handling procedure for the in-vessel components replacement were envisaged and investigated for improving the remote maintenance efficiency. However, the specific replacement scheme for the internal components, such as blanket, remains to be further research.

Based on the previous works on liquid lead–lithium (Pb-17Li) tritium breeder blanket by Institute of Nuclear Energy Safety Technology, Chinese Academy of Sciences. FDS Team [2–9] and the remote maintenance of the in-vessel components for ITER [10–12], a kind of liquid blanket concept, focus on its remote installation and removal was proposed for the superconducting magnets CFETR. In this contribution, the basic concept, structural design and cooling scheme of a typical outboard blanket on the equatorial plane were presented. For the purpose to estimate the structural performance of the blanket, thermal-hydraulic analysis of first wall (FW) has been carried out based on the heat load obtained from 3D real model neutronics calculations. And then, thermal stress analysis of the FW under normal condition has been performed according to the heat load obtained from the thermal-hydraulic analysis.

2. Design description

2.1. General concept

As the dual functional lithium–lead test blanket module for ITER, in this blanket concept, high pressure (8 MPa) helium gas is used as coolant, liquid PbLi is selected as tritium breeder and coolant, and Reduced Activation Ferritic/Martensitic (RAFM) steel, e.g. China Low Activation Martensitic (CLAM) steel [13], is selected as blanket structure material. Two operating modes, helium-cooled quasi-static lead–lithium (SLL) mode and helium/PbLi dual-cooled lead–lithium (DLL) mode, were designed as the initial and advanced stage of the blanket, respectively. In the SLL mode, PbLi is only used as breeder, flows quasi-statically in tritium breeding zones (TBZs) and the outlet temperature (450 °C) is below the compatibility temperature limit for PbLi/CLAM steel. Coating (e.g. Al_2O_3)

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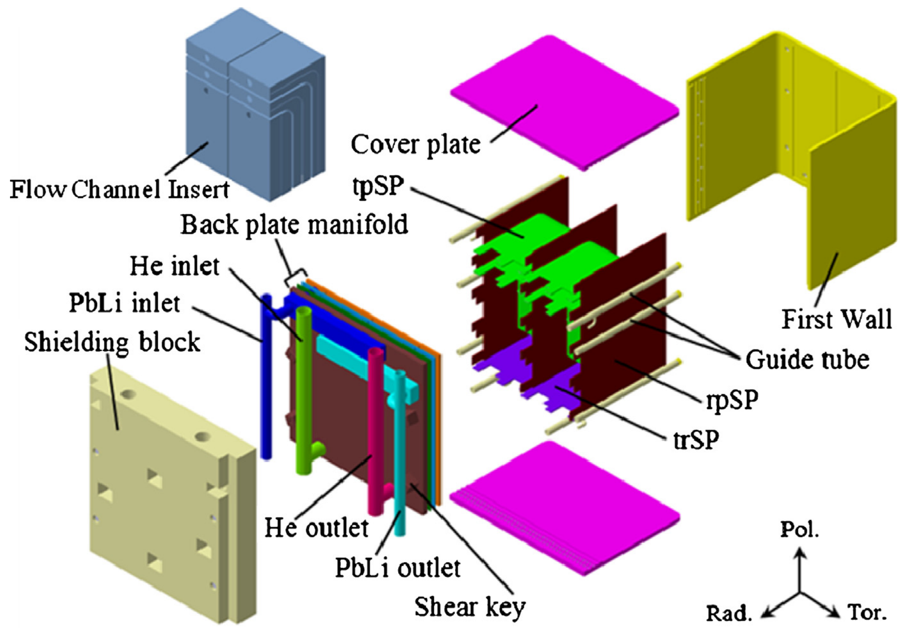

Fig. 1. Exploded view of the liquid lead–lithium tritium breeder blanket for CFETR.

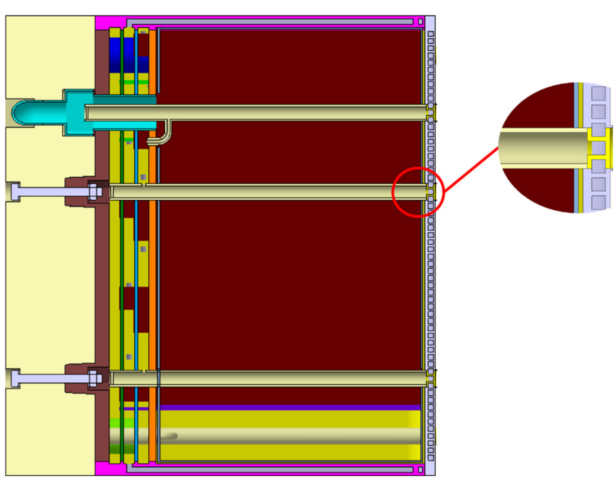

Fig. 2. Cutaway view of the blanket across the center of the GTs close to the PbLi outlet side.

is considered to prevent tritium permeation and protect the steel structure against corrosion of PbLi. In the DLL mode, helium is used to cool the blanket structure. PbLi is acts as self-cooled tritium breeder and flows fast in the TBZs. To reduce magnetohydrodynamic (MHD) pressure drop and obtain a high outlet temperature up to 700 °C, flow channel inserts (FCIs) made of Si_fC/SiC composite serve as thermal and electrical insulators are fixed in PbLi channels. Besides, coating is also needed in the DLL mode.

2.2. Structural design

In view of the space limitation in the VV and the desire to facilitate maintenance, the dimensions of the blanket were designed to be 1280 mm (poloidal) × 1200 mm (radial) × 1300 mm (toroidal). In the radial direction, the blanket is divided into two parts: tritium breeding module and shielding block. The tritium breeding module is designed as a replaceable component, which is connected with the shielding block by mechanical connections (e.g. flexible

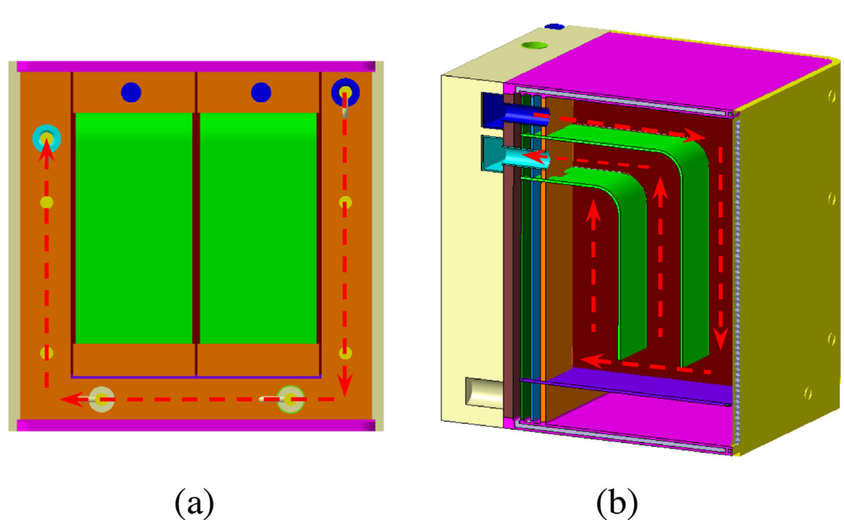

Fig. 3. Liquid PbLi flows in the (a) “U” shaped and (b) “7” shaped TBZs.