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Preliminary Design for an ITER Test Blanket Module*

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Abstract: Preliminary design and analysis for China helium-cooled solid breeder (CH HC-SB) test blanket module (TBM) for testing in ITER(International Thermonuclear Experimental Reactor) device have been carried out recently. In this paper, the design description, the performance analysis and the related ancillary systems for CH TBM are introduced. The key features of the design are based upon the breeder-outside-tube (BOT) concept, on the use of solid breeder ceramic material, of helium as coolant and tritium purge gas, of ferrite-martensite steel as structural material, and of beryllium as neutron multiplier. Results show that the proposed TBM concept has the advantages of higher tritium breeding ratio (TBR), simple structure design and engineering feasibility.

Key words: ITER; test blanket module; helium cooled solid breeder blanket; TBM
CLC number: TL64 **Document code**: A

1 Introduction

ITER will play a very important role in testing for the first time the blanket modules integrated in a fusion environment. Some of the DEMO blanket related technologies, such as tritium self-sufficiency, the extraction of high-grade heat, design criteria and safety requirements and environmental impacts, will be demonstrated in ITER test blanket modules (TBMs).

China has planned to develop independently ITER TBM modules for testing during ITER operation period. Although different concepts for ITER TBMs have been proposed by other parties[1], the He-cooled solid breeder blanket(HC-SB) with ferritic/martensitic steel (FMs) is still the main stream for the fusion DEMO blanket design and has foundation of the worldwide R & D database. Therefore, a helium-cooled solid pebble bed concept has been adopted, as an option, in Chinese TBM modules design. Under the cooperation of domestic institutes, the preliminary design and performances analysis as well as a draft Design Description Document (DDD)[2], based on the definition and the strategy of DEMO fusion reactor in China, have been carried out recently. Preliminary design and analysis have shown that the proposed TBM module concept is feasible within the existing technologies.

2 Design Description

The schematic structure and 2-D calculation model of the HC-SB TBM are shown in Figs. 1 and 2. CH TBM has the following overall size: 890 mm (H) × 664 mm (W) × 630 mm (D). The ferritic/martensitic steel EUROFER is chosen as reference structural material for the first wall and the main components. The lithium orthosilicate, Li_4SiO_4, is selected as tritium breeder. In order to increase the filling ratio in the tritium breeding zone, binary breeder sizes of diameters

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0.5 mm and 1.0 mm are used in this design. The helium gas is used as coolant in the helium cooling system (HCS) and as purge gas for tritium extraction. To assure an adequate tritium breeding ratio (TBR), beryllium pebble bed is adopted as neutron multiplier in the breeding zone; Li_4SiO_4 with lithium 90% enriched in ^6Li is used as tritium breeder. The pressure of the helium cooling system and the tritium extraction system are 8 MPa and 0.1 MPa, respectively. Main parameters of the HC-SB TBM design are shown in Table 1.

Table 1 Main parameters of HC-SB TBM

Parameters	Values
Neutron wall loading/(MW/m^2)	0.78
Max. surface heat flux/(MW/m^2)	0.5
Tritium breeding rate, TBR	1.29
Tritium breeder	Lithium orthosilicate, Li_4SiO_4
Form	D = 0.5—1.0 mm, pebble bed
^6Li Enrichment(%)	90
Max. temperature(℃)	664
Neutron multiplier	Beryllium
Form	Binary, D = 0.5—1.0 mm, pebble bed
Max. temperature(℃)	518
Coolant (He)	
Pressure/MPa	8
Temperature (inlet / outlet) (℃)	300 / 500
Pressure drop, ΔP	0.38
Structural material	Ferritic steel, Eurofer 97
Max. Temperature(℃)	541

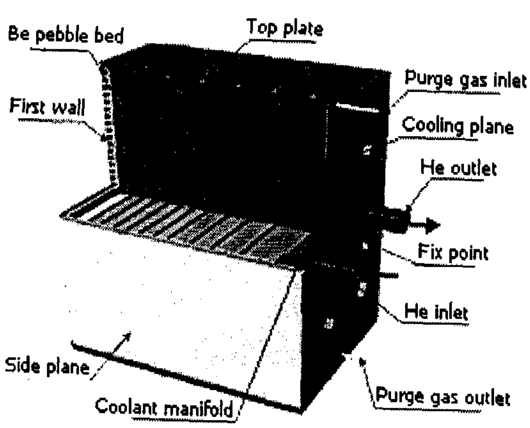

Fig. 1 Schematic view of the HC-SB TBM module.

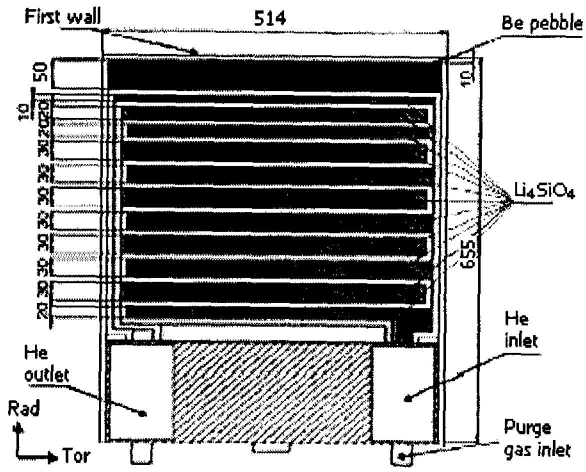

Fig. 2 2-D model of the HC-SB blanket module.

3 Performance Analysis

3.1 Neutronics analysis

The neutronics calculations have been performed using the neutron transport codes: 1-D ONEDANT[3], 2-D TWODANT[4] and 3-D MCNP[5]. The 3-D results from MCNP are definitely selected as input data for other systems design. 1-D and 2-D calculations are mainly used in optimization calculation for geometry and materials. The data library is based on FENDL 2.0[6]. The results of 1-D, 2-D and 3-D neutronics transport calculation yield local tritium breeding ratio (TBR) of 1.29, 1.23 and 1.15, respectively. The power density distribution has been calculated by using MCNP code. A peak power density of 9.71 W/cm^3 under an average neutron wall loading of 0.78 MW/m^2 occurs at the end of first breeding zone of Li_4SiO_4. Fig. 3 shows the power density distribution calculated along the radial direction.

In order to improve the distribution of power density in the blanket module, the arrangement of the Be neutron multiplier in the breeding zone has been optimized. Binary Be pebbles with diameters 0.5 mm and 1 mm were chosen for the bed.

Fig. 4 shows the tritium production rate in the module as a function of the distance from the first wall. The tritium generated amounts to about 0.022 g under the ITER standard operation condition[7]. The generated tritium amount is also a basis of the tritium extraction system and coolant purification system design.

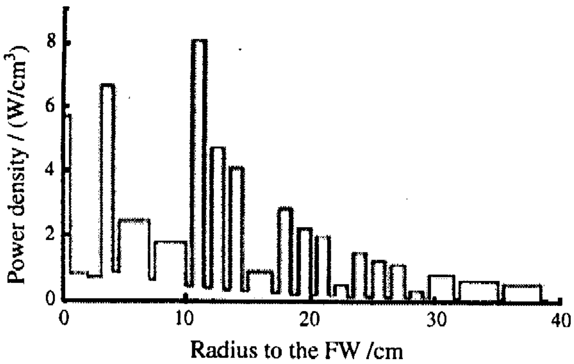

Fig. 3　Radial power density distribution in the blanket module as a function of distance from the first wall in the radial direction.

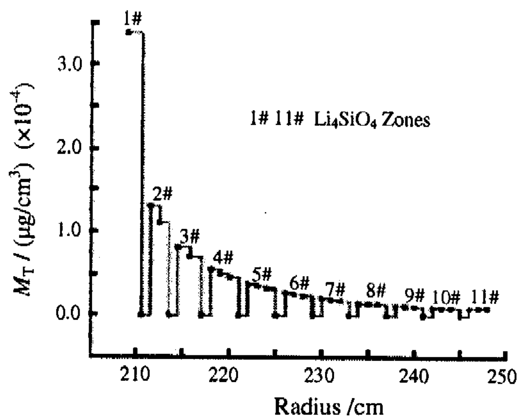

Fig. 4　Tritium distribution vs. the blanket thickness.

3.2　Activation analysis

The Be neutron multiplier, the structural material and the Li_4SiO_4 tritium breeder can be seriously activated in high energy D-T neutron field and yield many radioactive materials. Activation analysis has been performed assuming a continuous irradiation over one year at full fusion power (500 MW). Neutron fluxes are provided in 46 energy groups by one-dimensional neutron transport code, BISON1.5[8] for each specified material zone. Activation calculations are performed by means of computation code FDKR[9].

Variations of the radioactivity and the residual afterheat after shutdown for one year's operation are shown in Fig. 5. Results show that the radioactivity and afterheat are 1.87×10^{16} Bq and 5.06×10^{-3} MW at shutdown time after one year of operation, respectively. Results also show that the structure materials dominate the activity and residual afterheat in the TBM module design.

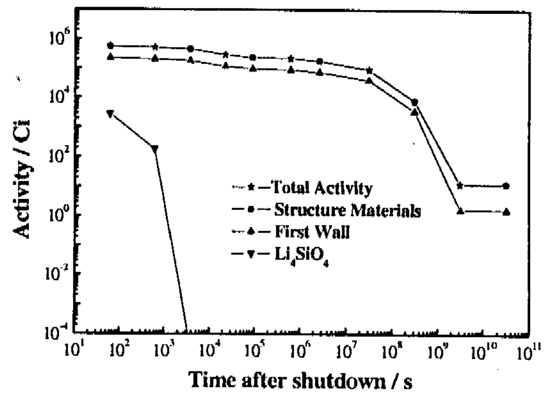

Fig. 5　Activity as a function of time after shutdown in HC-SB module.

3.3　Thermal hydraulic analysis

The thermal-hydraulic and stress calculations for the first wall, cooling tube and cooling plate and backplate, were performed by means of computer codes ANSYS[10] and FLUENT. Calculation results show that the peak temperature at the interface of solid breeder and structural material amounts to 636 ℃ with a fusion power of 500 MW and a surface heat flux of 0.5 MW/m^2. It can be found that the peak temperature of the structure materials is 541 ℃, which is located at the first wall. The peak temperature of the beryllium armor on the first wall is 574 ℃. A total heat power of 0.84 MW is deposited in the blanket module. The inlet and outlet temperatures of the helium coolant are 300 ℃ and 500 ℃, respectively. The temperature distributions profile obtained for the test blanket module and cooling plate are shown in Figs. 6 and 7. The results show that the temperature of different zones is in the permissible range of different materials (790℃ for beryllium pebble beds, 550 ℃ for ferritic steel, and 900 ℃ for ceramic Li_4SiO_4). Especially, the temperature of the lithium silicate pebble bed is in the range of 420—640 ℃, which is the best temperature windows for extracting tritium in the lithium silicate pebble bed.

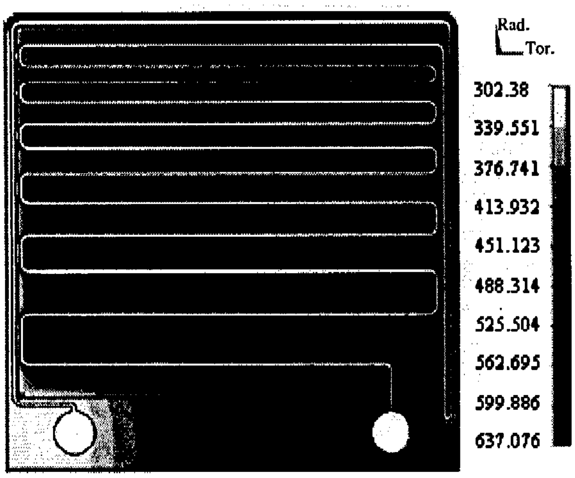

Fig. 6 Temperature distribution of the blanket module(℃).

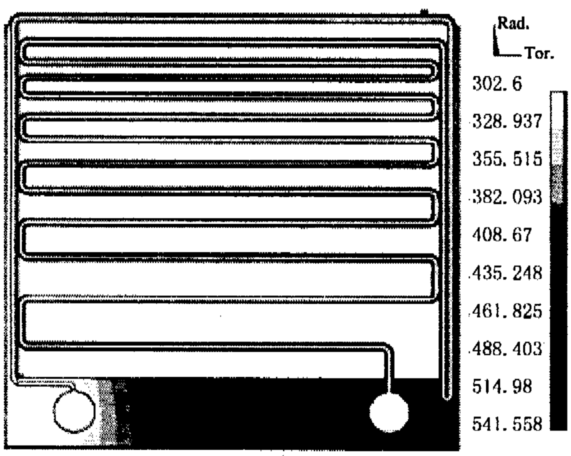

Fig. 7 Temperature distribution of the cooling plate(℃).

3.4 Stress analysis

According to the thermal-hydraulic calculations, the stress analysis for different components has also been completed by using the ANSYS code. The following assumptions were used for calculation: 1) Irradiation and creep effects were not taken into account; 2) The loads from electromagnetic forces have been ignored; 3) Stress was obtained by means of elastic approach. The thermo-mechanical properties of structural material used in calculations are: Young's modulus of 181.5 GPa, Poisson ratio of 0.3, thermal expansion coefficient of 11.9×10^{-6}/K, and thermal conductivity of 29 W/mK. As shown in Figs. 8 and 9, the max. equivalent stress of the first wall is 182 MPa. Max. equivalent stress of the cooling plate in the breeding zone amounts to 142 MPa. Results show that all stresses are below permissible limits for the requirements of structure strength regulations according to the $3S_m$ rules of ASME code[11] for the boiler and pressure vessel.

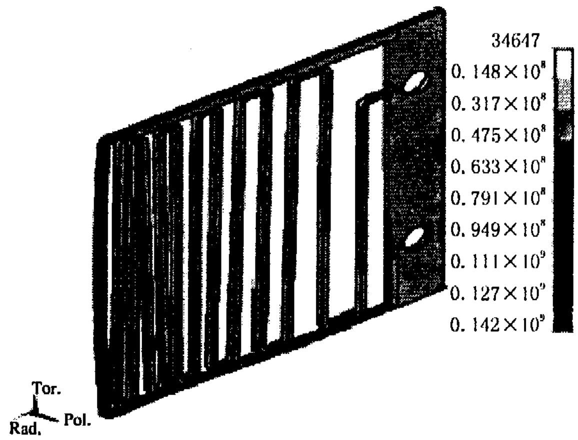

Fig. 8 3-D stress distribution of cooling tube(Pa).

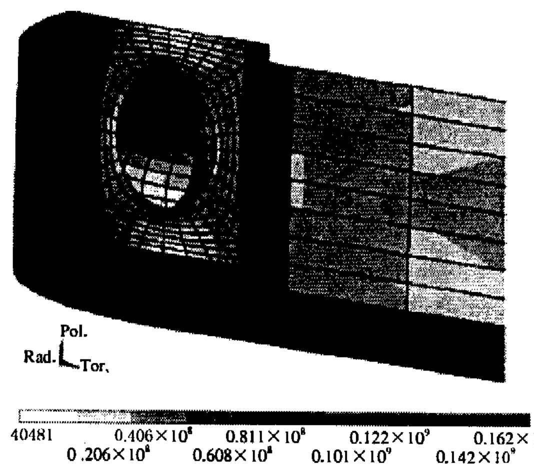

Fig. 9 Stress distribution in the first wall(Pa).

4 Summary

A preliminary design concept for the CH ITER HC-SB TBM has been proposed. Preliminary design and performance analysis for the TBM module have been performed. The results show that the proposed TBM design is feasible within the existing domestic technologies. It is characterised by a simple structure design, and the high TBR achieved can meet the design requirement. The design description document (DDD) will be carried out as scheduled within this year. The further HC-SB TBM design works will update and optimize the structure design as well as ancillary subsystem parameters.

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ITER 实验包层模块初步设计*

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摘　要： 中国国际热核聚变实验堆(ITER)氦冷固态氚增殖剂实验包层模块(CH ITER HC-SB TBM)设计已经完成。给出了 HC-SB TBM 的总体设计、性能分析和相关辅助系统的设计。HC-SB TBM 氚增殖区的设计采用 BOT 概念，锂陶瓷做氚增殖剂，氦气做冷却剂和载氚介质，铁素体马氏体钢做结构材料，铍做中子倍增材料。设计和分析结果表明，所提出的设计具有高氚增殖率、结构简单和工程上可行的特点。

关 键 词： 国际热核聚变实验堆；实验包层模块；氦冷固态氚增殖剂包层；实验包层模块

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