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Overview of design and R&D of solid breeder TBM in China

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ABSTRACT

Testing of breeding blanket module (TBM) is one of ITER's important objectives. China is performing design and technology development of ITER TBMs based on the development strategy of fusion DEMO in China. Solid breeder with helium-cooled test blanket module concept for test in ITER should be the basic option in China. The progress and status of China helium-cooled solid breeder (CH HCSB) TBM since 2004 are introduced briefly. Concept designs of HCSB TBM and ancillary systems, test strategy for their tests in ITER, key R&D issues are summarized in this paper. An international collaboration in R&D, development and testing of TBMs are proposed.

1. Introduction

ITER will play a very important role in first integrated blanket testing in fusion environment. Some of related technologies of DEMO blanket, 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 test blanket module program is an important step towards the development of breeding blankets for China fusion DEMO plant. As one of the options, China plans to develop own solid breeder blanket concept based on the definition and development strategy of Chinese fusion DEMO plant [1]. Therefore, a helium-cooled/solid tritium breeder (HCSB) with the pebble bed concept has been adopted in Chinese TBM modules design. The structure dimension of HCSB TBM design was based on 1/2 ITER test port and vertical partition. The basic future of HCSB TBM is by using the modularized sub-module design to replace the integral-module design.

Under the cooperation of domestic institutes, the preliminary design and performances analysis as well as a draft design description document (DDD) 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 preliminary design and performance analysis of HCSB TBM have been carried out recently. The structure design outline of China helium-cooled solid breeder (CH HCSB) module based on 1/2 port of ITER test port was completed. It is assumed that Chinese two TBM modules will common share the ancillary systems on one port. According to the ITER-FEAT EDA design, the integrated assembly on ITER test port for two TBMs is shown in Fig. 1.

The lithium orthosilicate, Li_4SiO_4 with lithium 90% enriched in ^6Li is used as tritium breeder. To assure an adequate tritium breeding ratio, TBR, beryllium is adopted as neutron multiplier. In order to increase the filling ratio in the neutron multiplier zone, binary breeder sizes of diameters 0.5 mm and 1.0 mm are used in this design. The low-activated ferritic/martensite steel (LAFMS) was used as structure material. The helium gas is used as coolant in the helium-cooling system (HCS) and as purge gas for the tritium extraction, respectively. The pressure of the helium-cooling system and the tritium extraction system are 8 MPa and 0.1 MPa, respectively. Main parameters of the HCSB TBM design are shown in Table 1.

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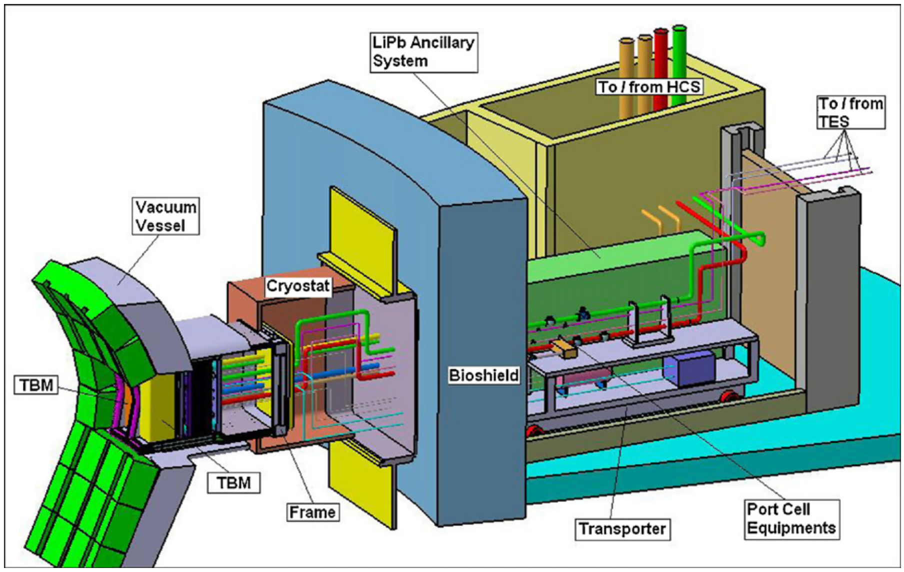

Fig. 1. HCSB TBM integrated assembly on ITER test port.

2.1. Structure design

The structure design has been improved based on the neutronics results from 3D MCNP global model. In order to decrease proportion of structure material and breeder material in the SM, arrangement of sub-modules with 3×6 was changed into 2×6 sub-modules. Main purpose of this modification is to enhance breeding tritium rate as well as to reinforce the TBM box.

The schematic structure and 2D calculation model of the HCSB TBM are shown in Figs. 2 and 3. CH TBM has the following overall sizes: 1660 mm (H) × 670 mm (W) × 484 mm (D).

The HCSB TBM consists of the following main components: U-shaped first wall, caps, backplate, grid, breeding sub-modules, and support plate. The sketch map of cross-section of the sub-module is illustrated in Fig. 3. The outer shell of the blanket box is made up from a steel with internal cooling channels bent into U-shaped. The two remaining sides being closed by cap plates. Backside of the blanket box is closed by backplate with all coolant channel for flow distribution. The stiffening grid is welded into the box, and each grid plate is cooled by helium flowing in internal channels that fed in the rear part. Breeding sub-modules (see in Fig. 4) are arranged into space which is divided by grid. Each sub-module has an independent cooling circuit and a purge gas circuit. Tritium breeder and neutron multiplier are divided by cooling channel of the sub-module.

Table 1
Main design parameters of HCSB TBM

Neutron wall loading (MW/m^2)	0.78
Maximum surface heat flux (MW/m^2)	0.5
Tritium production rate (g/day)	0.0123
Tritium breeder	Lithium orthosilicate, Li_4SiO_4
Form	Ø = 0.5–1.0 mm, pebble bed
^{6}Li Enrichment (%)	90
Peak temperature (°C)	664
Neutron multiplier	Beryllium
Form	Binary, Ø = 0.5–1.0mm, pebble bed
Peak temperature (°C)	
Coolant	Helium (He)
Pressure (MPa)	8
T_{in}/T_{out} (°C)	300/500
Pressure drop (Δp)	
Structural material	LAFM
Maximum temperature (°C)	

2.2. TBM ancillary system

Preliminary design of the helium-cooling subsystem has been performed by using ANSYS and RELAP5 codes. This system includes the primary helium heat transport loop with all components and the secondary heat removal loop. The secondary water loop is part of the ITER tokamak cooling water system (TCWS). The thermal power of the test module is removed to the ITER secondary cooling water loop with assumed condition of 35 °C and 75 °C. The pressure of the secondary water loop is lower than 1 MPa.

Main design parameters for the tritium extraction system (TES) are a composition of purge gas of 0.1% H + He, a pressure at the

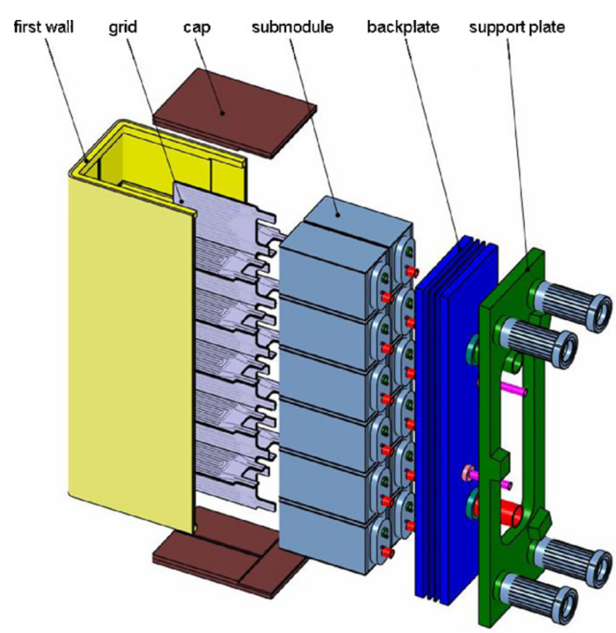

Fig. 2. Schematic view of the HCSB TBM module.

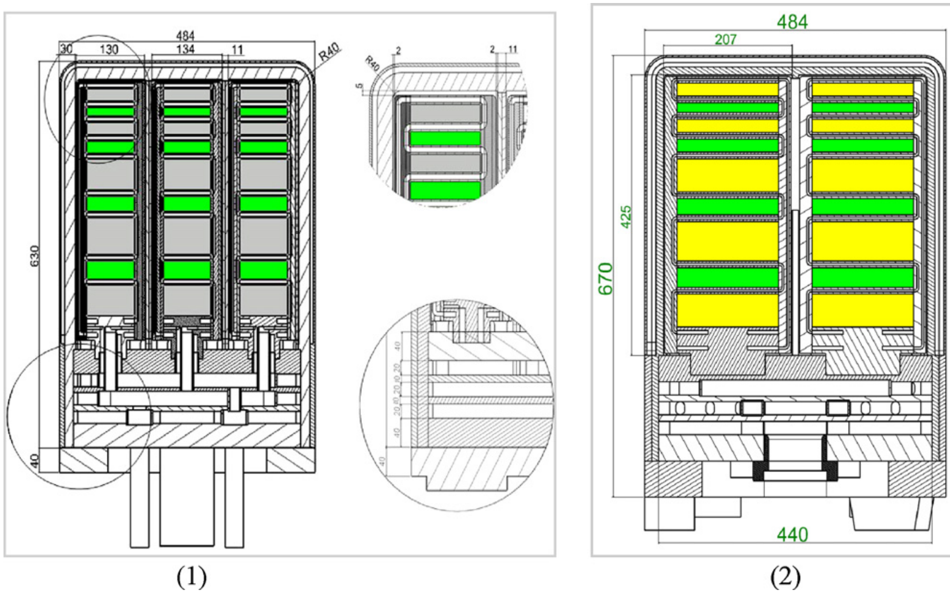

Fig. 3. 2D model of the HCSB TBM blanket module: (1) 3 × 6 arrangement of sub-modules and (2) modified 2 × 6 arrangement of sub-module.

inlet of TBM blanket of 0.12 MPa, extracted amount of tritium of 0.1 g/day, helium mass flow of 0.65 g/s, a tritium extraction efficiency of ≥95%. Tritium extraction system will be located in ITER tritium building. The TES design has the following features: first, a hot Mg bed is used to decomposed HTO released during regeneration of the molecular sieve bed (MSB) for tritium recovery in HT; secondly, a small size isotope separation subsystem (ISS) is designed to separate HT produced in the TES for H2 recycle, thus greatly reducing the amount of the discharged H_2 waste.

3. Performance analyses

Using ANSYS and FLUENT codes preformed the thermal–hydraulic and stress calculation. The results shown that the peak temperature at the interface of solider breeder and structural material can meet the requirement of engineering design. Preliminary safety analyses were completed by using FDKR, PELAP5 codes. The tritium exaction system, helium-cooling system, and the coolant purification system (CPS) have been designed. Relevant R&D on the key issues, the development of the solider breeder, the tritium permeation barrier, structure material, etc., will be preformed step by step as scheduled in China.

3.1. Neutronics analysis

The neutronics calculations for the HCSB TBM have been performed with the Monte Carlo code MCNP/4C [2], and nuclear cross-section data from the data library FENDL-2.0 [3]. A calculation model with 20° torus sector model based on the ITER-FEAT design is shown in Fig. 5. A platform of integrated HCSB TBM is shown in Fig. 6. Typically two million source neutron histories were tracked in the MCNP calculations to ensure a sufficient statistical accuracy for the calculated nuclear responses. The power density distribution has been calculated in the poloidal zones of the TBM. Fig. 7 shows the power density distribution calculated along the radial direction. Total energy deposition in HCSB TBM is 0.587 MW. A peak power density of 6.26 MW/m^3 under an average neutron wall loading of 0.78 MW/m^2 occurs at the end of first wall. The tritium generated amounts to about 0.0123 g/day under the ITER standard operation condition. In order to improve the distribution of power density in the blanket module, the arrangement of the Be

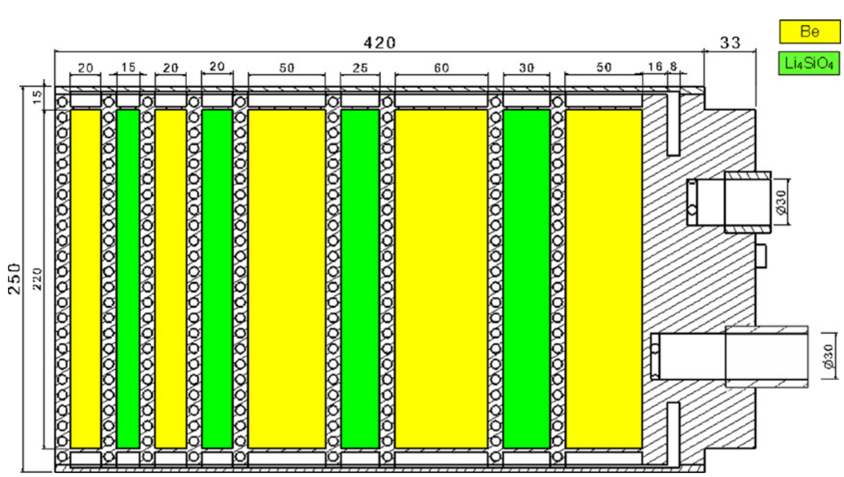

Fig. 4. Cross-section of HCSB TBM sub-module.

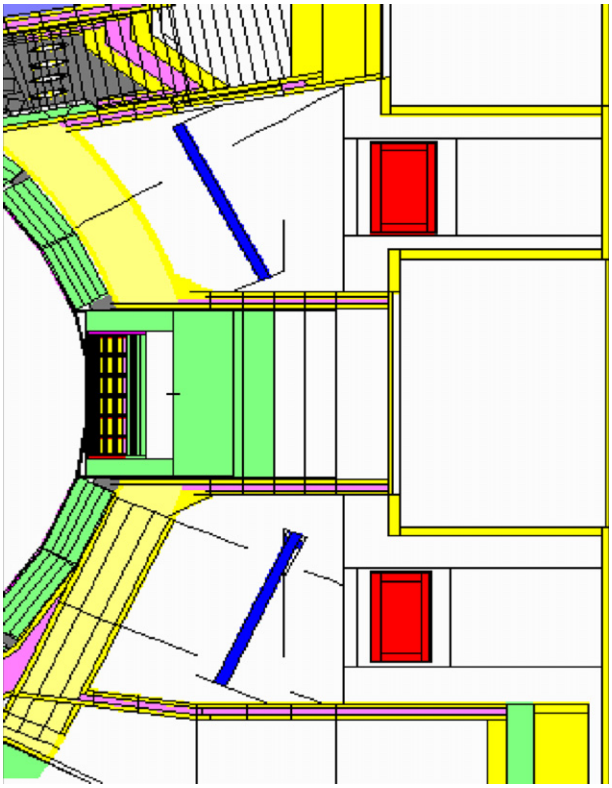

Fig. 5. Poloidal cross-section of ITER showing the HCSB TBM model.

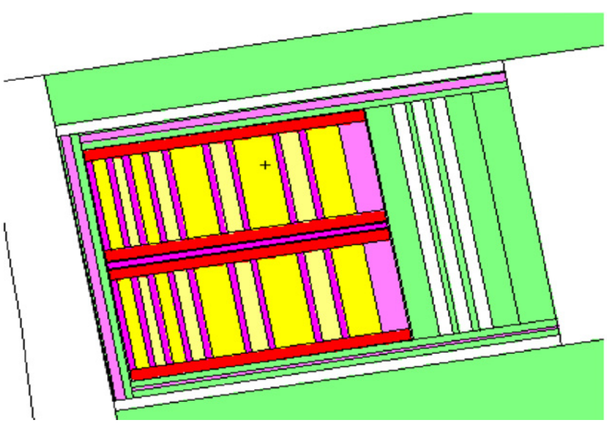

Fig. 6. Vertical cut of the MCNP model of HCSB TBM.

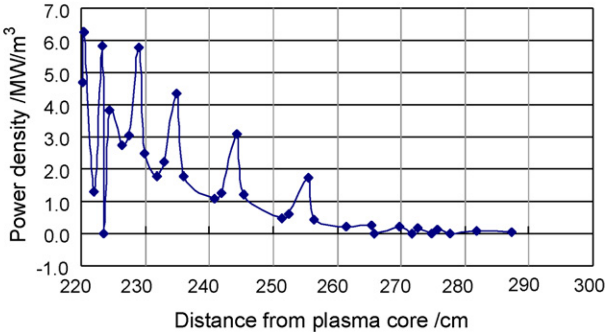

Fig. 7. Power density distribution in the poloidal zones of the TBM.

Table 2
Thermal–hydraulic parameters

Neutron surface loading (MW/m^2)	0.78
Surface heat flux (MW/m^2)	0.5
Total power (includes surface heat flux) (MW)	0.95
Helium pressure (MPa)	8
Helium inlet/outlet (°C)	300/500
First wall	300/360
Sub-module	360/500
Mass flow rate of helium (kg/s)	
First wall	1.34
Sub-modules	0.48
Equivalent diameter of flow channel (mm)	
First wall	15.3
Sub-modules	6
Velocity of helium (m/s)	
First wall	37
Sub-module	11
HTC (W/m^2 K) of with 0.2 mm roughness height	
First wall	6349
Sub-module	2132
Pressure drop of helium (MPa)	0.08
First wall	0.071
Sub-module	0.008

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.

3.2. Thermal–hydraulic analysis

Under the extreme condition, i.e., with 0.5 MW/m^2 of the FW surface heat flux, the relevant thermal–hydraulic parameters of the TBM were calculated and listed in Table 2.

According to these parameters, the thermal analysis was carried out by using FE code ANSYS [4]. The analysis results (Figs. 8 and 9) show that, the maximum temperature of FW is 499 °C, and for the SM, the maximum values in the structural material, Be and Li_4SiO_4 pebble beds are 516 °C, 660 °C and 687 °C, respectively, which are both within the permissible range.

3.3. Stress analysis

Based on the thermal analysis, the mechanical analysis was performed subsequently; the stresses in the TBM are mainly caused by the temperature distribution and the coolant pressure of 8 MPa.

Analysis results (Figs. 10 and 11) show that, the maximum total stress of FW and SM are 295 MPa and 373 MPa, respectively, compared with the maximum total stress limit 3.Sm at the corresponding temperatures, which are allowable [5]. The calculational results shown that the total stress are maily caused by the thermal effect; for the SM, among the thermal stress nearly occupied 92% share in the total stress.

3.4. EM analysis

In order to achieve a detailed and accurate electro-magnetic (EM) analysis on TBM, some results of plasma normal operation and disruption in 2D model (global analysis) around the TBM region of interest are calculated. According to ITER-FEAT magnet system design [6], poloidal field coils with fixed constant currents [7] are integrated in the model with ANSYS FEM code [8]. Considering the distance between CS, PF1,

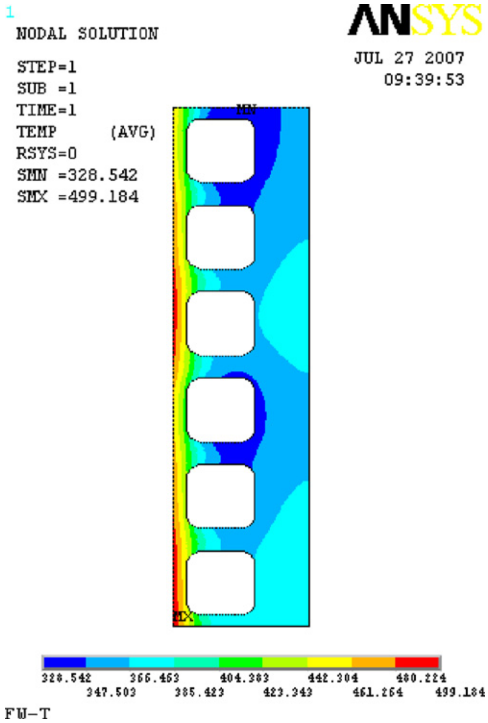

Fig. 8. Temperature distribution of FW.

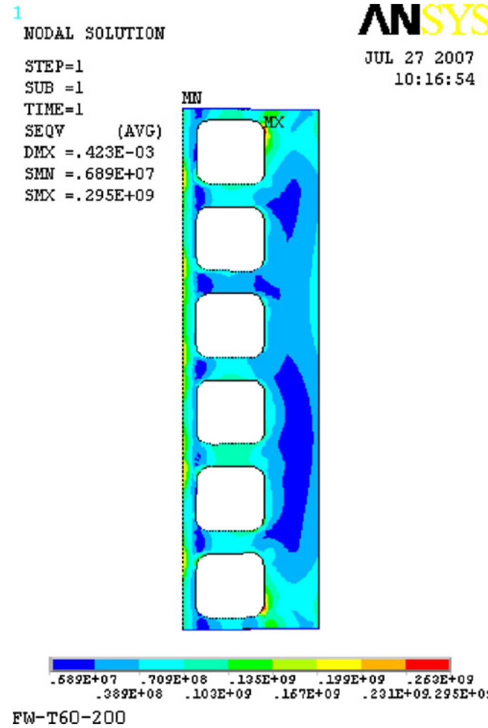

Fig. 10. Von Mises stress distribution of FW.

PF6 and TBM, its contribution of poloidal magnetic fields is ignored.

According to the ANSYS calculation results, maximum toroidal magnetic field in TBM produced by toroidal coils is 3.96 T. In plasma normal operation, maximum poloidal magnetic fields in TBM of poloidal coils and plasma current are 0.52 T and 0.603 T, respectively. After thermal quench of 2.2 ms, maximum poloidal magnetic field in TBM of plasma current is 0.590 T.

3.5. Accident and safety analysis

Activation analysis has been performed assuming a continuous irradiation over 1 year at full fusion power (500 MW). Neutron fluxes are provided in 46 energy groups by 3D neutron transport code, MCNP for each specified material zone. Activation and dose calculations are performed by means of computation codes FDKR [9] and DOSE. The composition data of structure material in the module is from reference material, EUROFER97. Results shown that

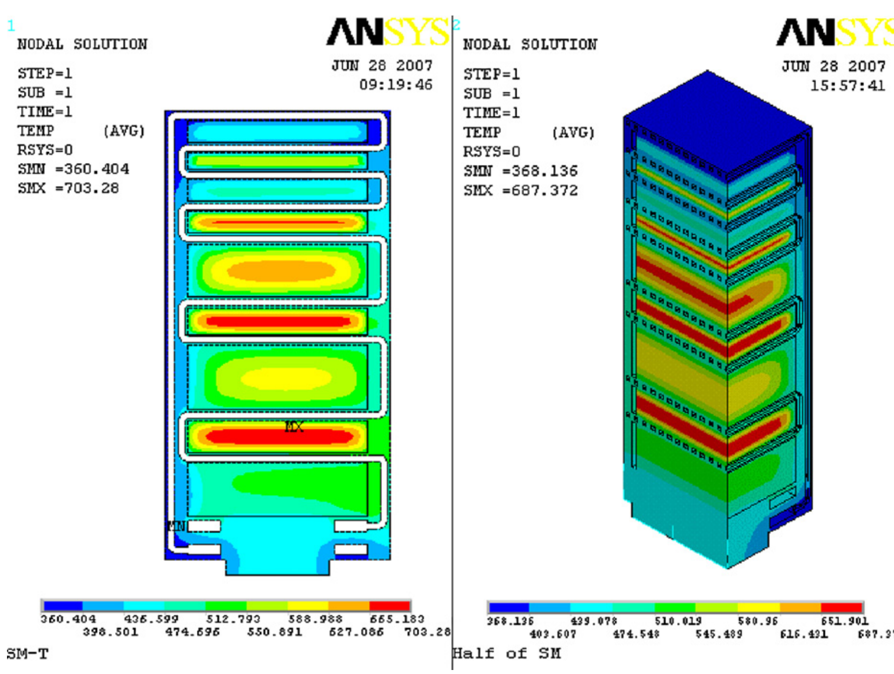

Fig. 9. Temperature distribution of SM.

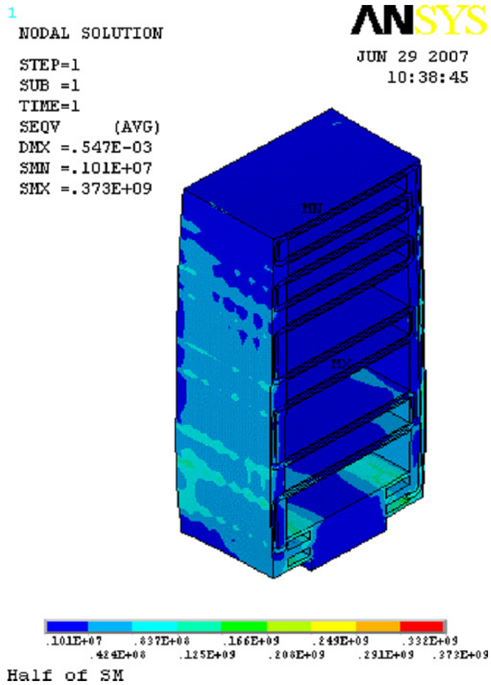

Fig. 11. Von Mises stress distribution of SM.

the total activation inventory stays at a level of ~3.04 × 10^{17} Bq for ~1 min and drops slowly thereafter and reaches an extremely low level value of 1.09 × 10^{13} Bq after 100 years. The decay curves of the radioactivity, afterheat and BHP after shutdown time are shown in Fig. 12. The dose rate is 3.34 × 10^{7} mSv at shutdown time, then drops slowly up to 1 year. Thereafter the dose rate declines rapidly and reach 0.097 mSv after 10 years. Considering ITER operation factor 0.22, after 10 years' cooling, the dose rate is enough to meet ALARA threshold.

The thermal–hydraulic safety analysis for 2 × 6 TBM have been carried out by using system code RELAP5/MOD3 [10]. The FW and 12 sub-modules, including all of the TBM components and the helium-cooling system have been modeled. The loop steady state results reveal that the TBM inlet/outlet temperatures are approximately obtained to be designed values 300 °C/500 °C.

Three postulated initiating events have been assessed for the LOCA analysis of TBM. The In-Vessel LOCA analysis shows that the pressurization of VV is up to about 10 kPa after 50 s of the happened LOCA, and it is within the limit value of 200 kPa for ITER design. The results for Ex-Vessel LOCA reveal that the FW beryllium armor will melt about 80 s after LOCA initiation. Taking some measures to shut down the plasma is very critical before the FW melting. The pressurization of TCWS vault is neglected taken into account its large volume. The In-Box LOCA analysis reveals that the purge gas pressure increases to about 8MPa within 2 s after LOCA initiation. So we have to isolate the tritium extraction system from TBM just after the beginning of LOCA in order to avoid the damage of tritium extraction system.

A list of 21, public safety relevant post-irradiation examination (PIE) has been set assessing elementary failures related to the different components of HCSB TBM systems [11]. Each PIE has been discussed in order to qualitatively identify accident sequences arising from each PIE itself. Deterministic analysis will demonstrate the plant capacity in mitigating and, in every case, in withstanding accident consequences, arising from the overall set of PIEs, below fixed safety limits.

4. R&D activities

4.1. Structure materials

The research and development of Chinese low-activation ferritic/martensitic steel (CLF-1) are being preformed since 2006. Several 10 kg ingots of CLF-1 steel were produced by means of vacuum induction melting and hot forged at 1200 °C to a final bar diameter of 13 mm. These steels contain about 9.0 wt.% Cr, 1.2 wt.% W, 0.3 wt.% V, 0.11 wt.% C, 0.6 wt.% Mn, 0.1 wt.% Ta, and Fe for the balance. The CLF-1 steel has a fully martensitic microstructure with about 7 μm grain size, which is much smaller than that 16.5 μm of Eurofer97.

Several kinds of mechanical properties, such as tensile properties, Charpy impact properties have been obtained. Tensile properties were conducted at temperatures ranging from room temperature to 700 °C. Charpy test were carried out at temperatures ranging from −100 °C to room temperature. Fig. 13 shows the tensile properties and Charpy compact properties of CLF-1 steel compared with Eurofer97 steel and F82H steel [12,13]. CLF-1 steel has the highest tensile strength and comparatively better ductility among these three kinds of steels, Fig. 12a. Also, CLF-1 steel has the highest impact energy and the lowest ductile-brittle transition temperature, Fig. 13b.

Some of the materials subjects, such as physical properties, aging effect on mechanical properties, thermal creep properties and joining technologies have been carried out and will be finished by the end of 2007. Also, 200 kg large ingots have been planned to be developed by the end of 2007, and neutron irradiation effect on DBTT and helium embrittlement needs to be investigated in the future.

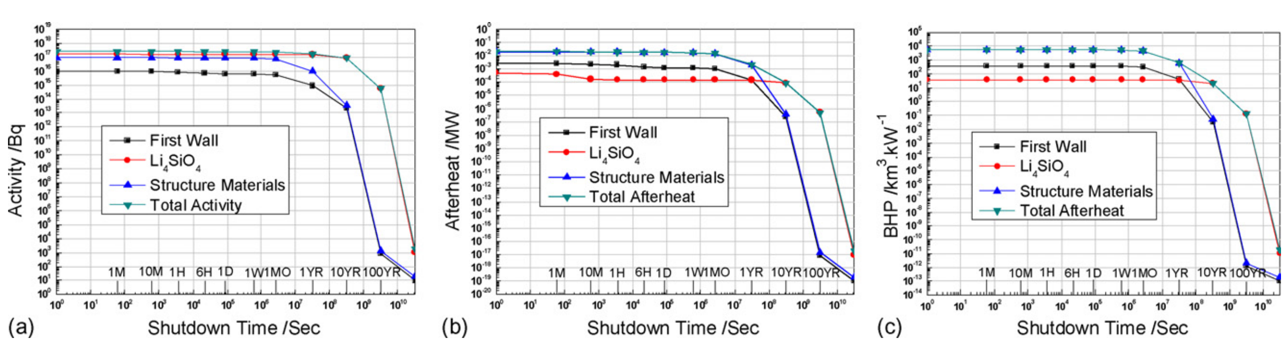

Fig. 12. Radioactivity, afterheat and BHP decay with shutdown time: (a) radioactivity, (b) afterheat and (c) BHP.

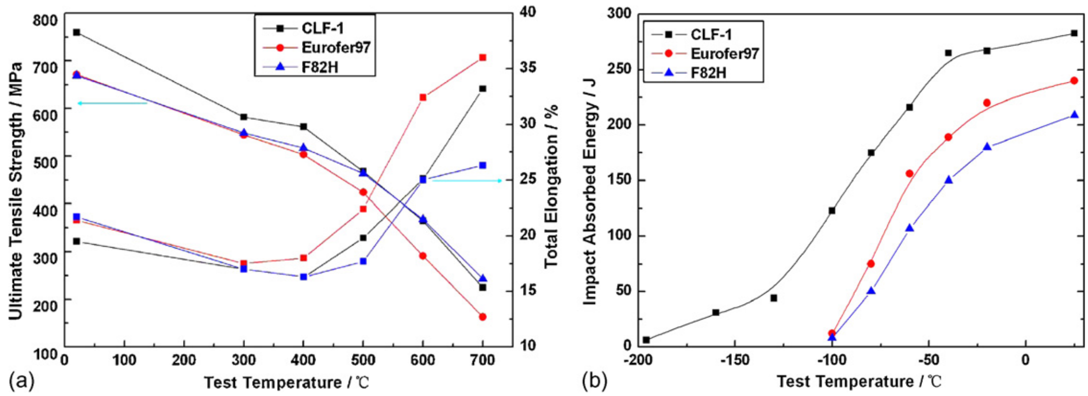

Fig. 13. Mechanical properties of CLF-1 steel compared with that of Eurofer97 and F82H steels: (a) tensile properties and (b) Charpy impact absorbed energy.

Table 3
Main performances of the breeder pebbles

Relative density	70–95% (TD)
Main phase center, *t* (%)	≥98
Grain size (μm)	0.5 to <10
Strength (bedding) (MPa)	100
Strength (compression) (MPa)	150–300
Li conductivity ($\times 10^{-3}$ S/cm)	1
Pore size (μm)	0.08–4
Impurity (ppm)	<100

4.2. Function materials

Two kinds of the solid tritium breeder, Li_2TiO_3 and Li_4SiO_4, have been investigated in China. Preliminary test results shows that two pebbles have good surface feature by using sol–gel method. Main performances of the breeder pebbles were shown in Table 3. The Li_4SiO_4 pebble sample with different diameters is shown in Fig. 14 (Table 4).

Exploration study of Be pebbles fabrication technology has been done. Related performance test is on going. Be alloy pearls are prepared by powder metallurgical (PM) methods.

In additional, some of test facilities prior to TBMs installation in ITER have been proposed. Relevant R&D on the key issues of the tritium system, the fabrication technical of the breeders as well as the tritium permeation barriers, etc., will be preformed in China.

Fig. 14. Li_4SiO_4 pebble sample with different diameters.

4.3. Helium test loop

Fig. 15 shows a flow diagram of the helium test loop (HTL) and interfaces to ancillary equipment. Because of the significant difference between the inlet/outlet temperature conditions at test section, the loop is designed as “8”-shaped to reduce the temperature difference between the primary and the secondary water-cooling loop. This arrangement will largely reduce thermal stress on the heat transfer structures inside the cooler. The bypass of circulator is used to control the flow rate of the circuit. The bypass of recuperator and cooler is used to control the inlet temperature of test section and circulator, respectively. The bypass of test section makes testing of the whole circuit without TBM possible. Fig. 16 shows the main components and the arrangement of the helium experimental loop.

5. TBM testing in ITER

Chinese TBM program is important one step towards the development of breeding blankets for fusion DEMO power plant.

The blanket test program of ITER aims at demonstrating the blanket functionality and performance, and validating the computational tools for the design of DEMO. Main testing objectives are

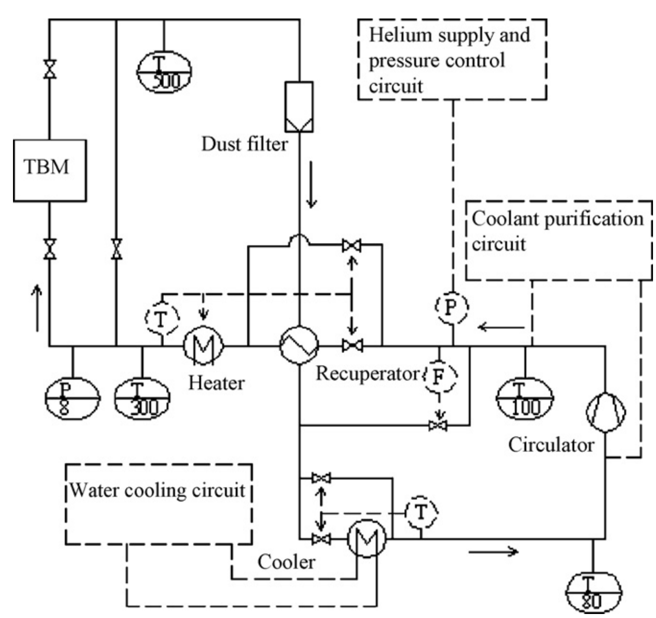

Fig. 15. Flow diagram of the helium test loop (HTL).

Table 4
Requirements for test sections [1,14]

Test section	HE mass flow rate (kg/s)	Pressure (MPa)	Pressure difference at test section (MPa)	He inlet/outlet temperature (°C)	Power supply (MW)
TBM	0.1–1	8	<0.3	100/300, 300/500	1
DEMO blanket	4	8	<0.4	300/500	5

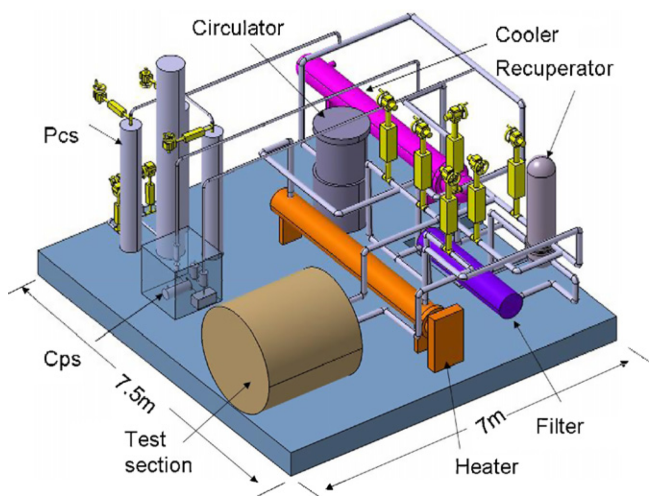

Fig. 16. Layout of the helium test loop.

to demonstrate the tritium breeding performance of the breeder blanket concepts and to check and validate the capability of the neutronics codes and data library.

In order to test and compare their technical feasibility, Chinese two TBM concepts (HCSB TBM and DFLL-TBM) will be tested in one port, with common auxiliary systems, including a common helium-cooling system, a liquid lithium–lead-cooling system (LLCS), a helium coolant purification system and a tritium extraction system. Of course, we could collaborate with others party through negotiation.

6. Summary

Preliminary design and performance analysis for the HCSB 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 modularization structure, and the high TBR achieved can meet the design requirement. The updated design description document of Chinese HCSB TBM has been carried out. The further optimization design and related R&D activities will be performed under the ITER IO unified TBM framework.

References

[1] K.M. Feng, C.H. Pan, G.S. Zhang, D.L. Luo, Z.W. Zhou, et al., Preliminary design for a China ITER test blanket module, J. Fus. Eng. Des. 81 (2006) 1219–1224.
[2] J.F. Briesmeister (ed.): MCNP—A General Monte Carlo N-Particle Transport Code, Version 4C, LA-13709-M, Los Alamos National Lab., NM, April 2000.
[3] H. Wienke, M. Herman, FENDL/MG-2.0 and FENDL/MC-2.0, The processed cross-section libraries for neutron–photon transport calculations, version 1, March 1997, IAEA Vienna, Report IAEA-NDS-176, Rev. 1, October 1998.
[4] ANSYS Technical Report, Rev. 5.1, IMAG Industries Co., February, 1995.
[5] American Society of Mechanical Engineers (ASME), Boiler and Pressure Vessel Code, Section III-Materials, ASME, 2004.
[6] Load Specification (LS)—Annex of Project Integration Document (PID) ITER_D_222QGL v 3.0, July 2005.
[7] M. Huguet, Key engineering features of ITER-FEAT magnet system and implications for the R&D programme, Nucl. Fus. 41 (2001) 1503–1513.
[8] ANSYS Documentation, ANSYS Co., Ltd., 2006.
[9] Y.X. Yang, K.M. Feng, J.H. Huang, CCC-541/FDKR, Radiation Shielding Information Center, Oak Ridge National Laboratory, 1989.
[10] The RELAP5 Code Development Team: RELAP5/MOD3 Code Manual vol. I: Code Structure, System Models, and Solution Methods, NUREG/CR-5535, INEL-95/0174, August 1995.
[11] C. Zhi, F. Kai-ming, Z. Zhi-wei, G. Chun-ming, L. De-li, et al., Preliminary failure mode and effect analysis on Chinese ITER helium cooled solid breeder test blanket design concept, in: Presented at the Second IAEA TM on First Generation of Fusion Power Plants—Design and Technology, Vienna, Austria, June 20–22, 2007.
[12] A. Moslang, E. Diegele, M. Klimiankou1, R. Lasser, R. Lindau1, E. Lucon, et al., Towards reduced activation structural materials data for fusion DEMO reactors, Nucl. Fus. 45 (2005) 649–655.
[13] S. Jitsukawa, M. Tamura, B. van der Schaaf, R.L. Klueh, A. Alamo, et al., Development of an extensive database of mechanical and physical properties for reduced-activation martensitic steel F82H, J. Nucl. Mater. 307–311 (2002) 179–186.
[14] B.E. Ghidesa, M. Ionescu-Bujor, G. Janeschitz, Helium Loop Karlsruhe (HELOKA) – A valuable tool for testing and qualifying ITER components and their He cooling circuits[J], Fusion Eng. Des. 81 (2006) 1471–1476.

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