Preliminary Design of China ITER Test Blanket Module with Helium-Cooled and Solid Breeder Concept

K.M. Feng 1), C.H. Pan 1), G.S. Zhang 1), D. L. Luo 2), Z.W. Zhou 3), H.Y. Wang 2), X.Y. Wang 1), Z. Chen 1), T. Yuan 1), Z. Zhao 1), Z.Q. Li 1), C.M.Gao 4), G. Hu 1), G.Y. Zheng 1) , Z.X. Li 1), Z. X.F. Ye 1)

1) Southwestern Institute of Physics, P.O. Box 432, Chengdu 610041, P.R. China.
2) Institute of Engineering Physics, Chinese Academy of Science, P.R.China
3) Institute of Nuclear Engineering Technology, Tstinghua University, Beijing 100084, P.R.China
4) University of Electronic Science and Technology of China, Chengdu 610054, P.R.China

email contact of main author: fengkm@swip.ac.cn

Abstract. A modified design based on a half space of ITER test port for China Helium-cooled Solid Breeder (CH HC-SB) Test Blanket Module (TBM) for testing in ITER device have been carried out recently. In this paper, the design description, the performance analysis and the related ancillary systems design for the HCSB-TBM were given. The key features of the design are based upon the breeder outside tube (BOT) –concept, the use of ceramic breeder material; helium as coolant and tritium purge -gas, Low Activation Ferrite/Martensite steel (LAFMs) as structural material, and Beryllium as neutron multiplier. Results show that proposed TBM concept can meet the design requirement.

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 has planned to develop own the ITER TBM modules for testing during ITER operation period. Although different concepts for ITER TBMs have been proposed by other parties [1]. However, the he-cooled solid breeder (HCSB) blanket with Ferritic /Martensitic steel (FMs) is still main stream for the fusion DEMO blanket and has foundation of the world R&D database. Therefore, a helium-cooled solid pebble bed concept has been adopted, as one of options, in China ITER TBM modules design. 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. CH TBM has the following overall sizes: 1660mm (H)×484mm(W)×630mm (D). The low-activation ferritic/martensitic steel (LAFMs) is chosen as design structural material for the first wall and the main components. The HCSB TBM consists of first wall, breeding sub-modules, back plate, caps, grid and support plate. A beryllium armor of 2 mm-thickness on the FW is used as the plasma facing component (PFC).
A U-shaped structure with the thickness of 30mm is used for the first wall with 87 cooling channels with dimensions of 14mm×7mm. As shown in Fig.2, Arrangement of the sub-modules by 3×6 is adopted in the structure design. The integral HCSB

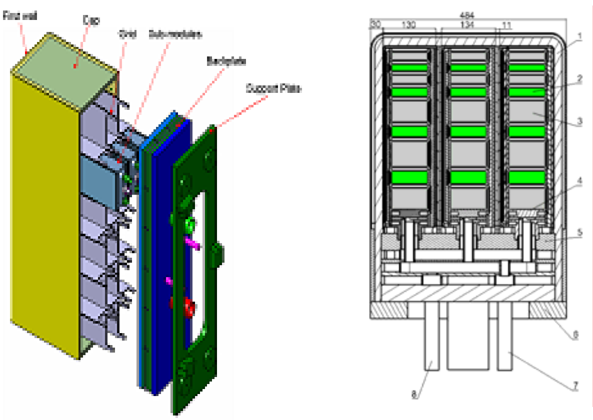

FIG.1 Schematic view of HCSB TBM

TBM consists of 18 breeding sub-modules. Each has independent cooling circuit and purge gas circuit. The back plate forms the back support, where the return manifolds used to collect the helium flow and the purge gas manifolds are connected.

The lithium orthosilicate, Li_4SiO_4 of with 1mm diameter and with enriched lithium of 80% ^{6}Li are selected as the tritium breeder. To assure an adequate Tritium Breeding Ratio (TBR), beryllium pebble is adopted as neutron multiplier in the neutron breeding zone. In order to increase the filling ratio in neutron multiplier zone, Be pebble of diameters 0.5-1.0mm are used. The helium gas is used as coolant of the Helium Cooling System (HCS) and the purge gas of Tritium Extraction System (TES). The pressure of the helium cooing 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. Fig.3 shows the general layout of CH HC-SB TBMs in ITER test port.

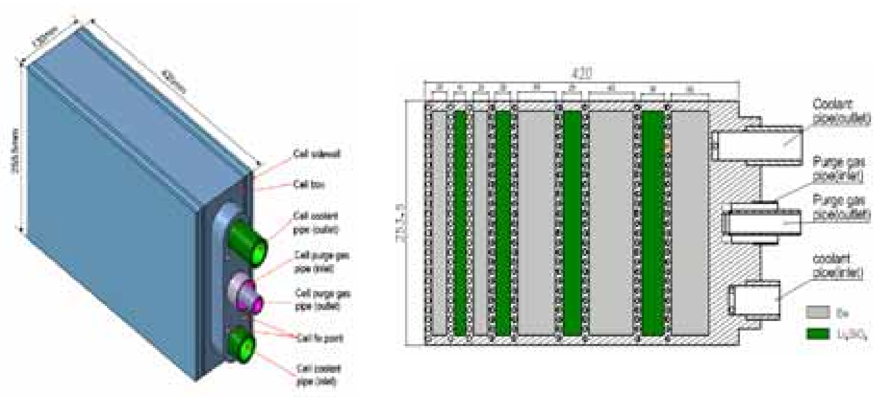

FIG.2 overall isometric view of sub-module

Table 1 Design parameters for the China HCSB TBM

Configuration	BOT (Breeder Out of Tube)	Modules: 3×6 Sub-modules
FW area Neutron wall loading Surface heat flux	0.484m (W)×1.660m(H)	$0.803m^2$. $0.78MW/m^2$ $0.30\text{-}0.50MW/m^2$
Total heat deposition	NT-TBM, PI-TBM	0.76MW
Globe TBR	Lithium orthosilicate, Li_4SiO_4	0.63 (3-D), 80% Li-6
Tritium production rate	ITER operation condition	1.7×10^{-2}g/d
Sub-module dimension	(P)×(T) ×(R)	253.5mm×130mm×420mm
Ceramic breeder (Li_4SiO_4)	Diameter Thickness Max. Temperature	1mm, pebble bed 90mm (four zones) 742℃
Neutron multiplier (Beryllium)	Diameter Thickness Max. temperature	0.5~1mm, Pebble bed 200mm (five zones)+2mm (armor) 550℃ (Armor) 613℃ (Be Pebble bed)
Structure material	RAFM Max. temperature	RAFM 537℃
Coolant helium (He) Pipes size	Pressure Pressure drop Temp. range (inlet/outlet) Mass flow Diameter (OD/ID)	8MPa 0.22MPa 300/500℃ 0.99kg/s 101.6/85.5mm
He purge flow (He)	Pressure Pressure drop	0.12MPa 0.02MPa

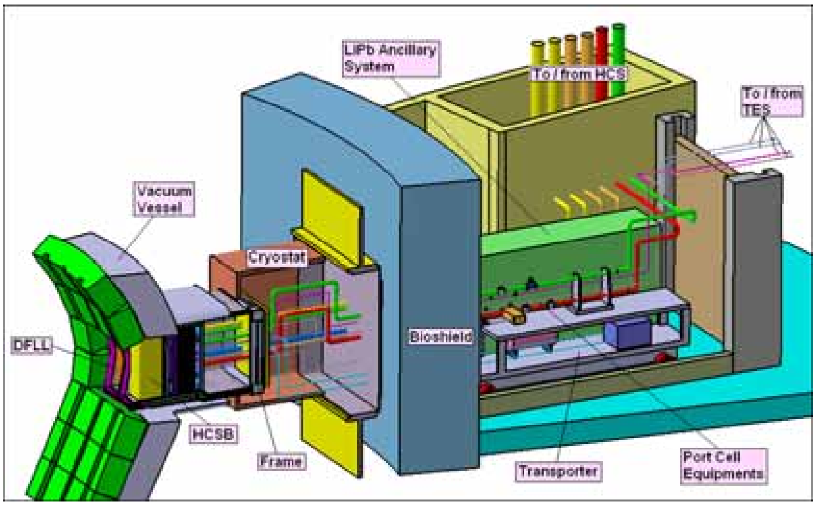

FIG 3 General layout of CH TBMs in ITER port cell C

3. Performance analysis

3.1. Neutronics ananysis

The neutronics calculations are 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 FENDL2.0 [6]. The results of 1-D and 2-D neutronics transport calculation yield local tritium breeding ratio (TBR) of 1.29 and 1.23, respectively. A peak power density of 9.71W/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. The 3-D MCNP calculation models are shown in Fig.4. The results shown that peak power density, energy deposition and tritium breeding ratio are 5.85MW/m^3, 0.71MW, and 0.6, respectively.

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

Fig.5 shows the distribution of power density along the radius direction. The production rate of different tritium breeding zones in the sub-module is shows in Fig.6. The tritium generation amount is about 0.017g/d under the ITER standard operation condition [7]. The tritium generation amount will also be a basis of the tritium extraction system and coolant purification system design.

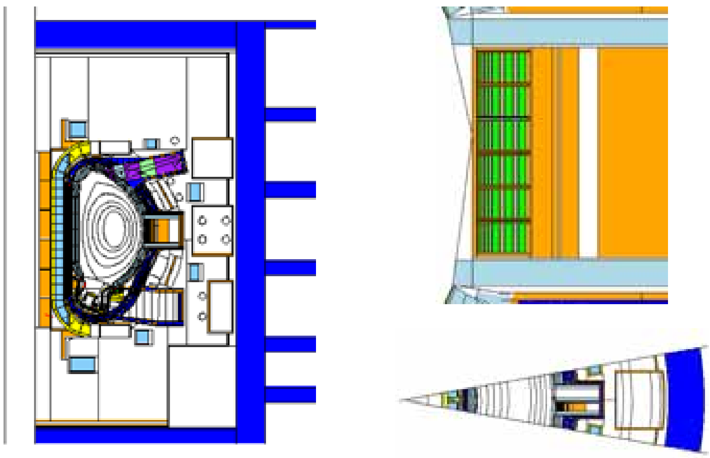

FIG.4 MCNP Model for HCSB TBM

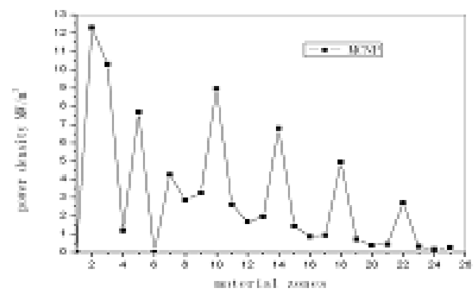

FIG.5 The distribution of power density

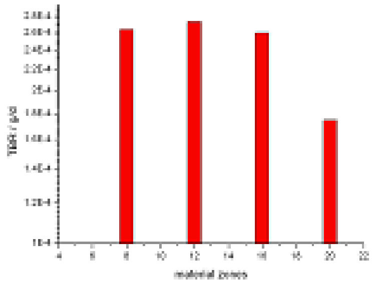

FIG.6 Tritium production rate

3.2 Activation analysis

The neutron multiplier Be, the structure material and the tritium breeder Li_4SiO_4 can be seriously activated in high energy D-T neutron field and yield many radioactive materials. Activation analysis was performed assuming a continuous irradiation over one year at full fusion power (500 MW). Neutron fluxes and spectra are provided in 46 energy groups by 1-D BISON1.5 calculation for each specified material zone. Total radioactivity and afterheat are calculated using the activation code FDKR and its decay chain library DCDLIB.

At shutdown, the total radioactivity inventory is as low as about 0.895MCi. It is mainly due to the structure materials in the shield, back plate, the back breeder channel, and the FW. The level is 0.147MCi after 1 year and 0.013MCi after 10 years. Figs.7-8 shows the activity and the afterheat as a function of the shutdown time, respectively.

The total decay heat at shutdown is about 7.78×10^{-3}MW. The total decay heat is dominated by the contribution from the structure at all time. The total BHP at shutdown is 89.6km^3/kW with a contribution of 89.4km^3/kW, 0.13km^3/kW from structure material and Li_4SiO_4, respectively. From a fraction of an hour up to 1,000 years after shutdown, the total BHP is attributed to the contribution from the structure.

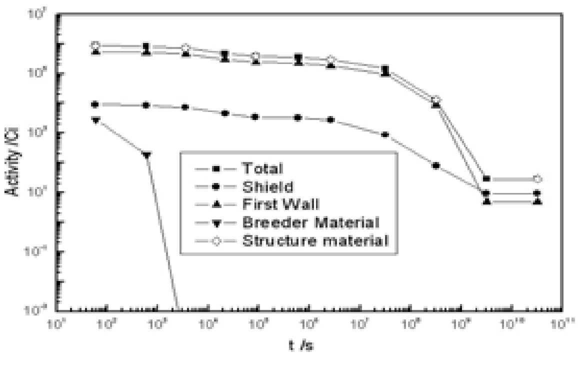

FIG.7 Activity as a function of the cooling time

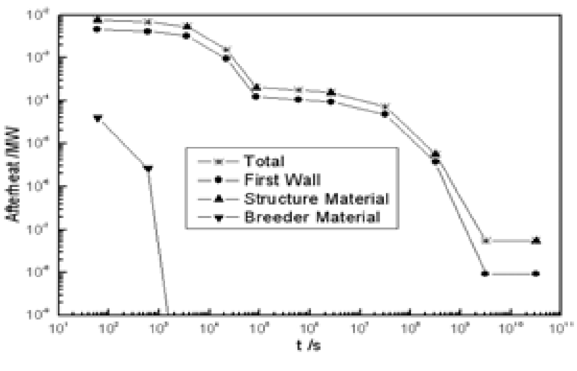

FIG.8 Afterheat as a function of the cooling time

3.3. Thermal hydraulic analysis

The thermal-hydraulic and stress calculations for the first wall, cooling tube and cooling plate and back-plane, were preformed 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 °C with a fusion power 500MW and a surface heat flux of 0.5 MW/m^2. It can be found that the peak Max. temperature of the structure materials is 523 °C, which is located at the first wall. The peak temperature of the Beryllium armor on the first wall is 550 °C. Total heat power of 0.76 MW will deposit in the blanket module. The temperature distributions profile for test blanket module and cooling plate obtained are shown in Figs.9-10. The results show that the temperature of different zones is in the permissible range of different materials (790 ^{O}C for Beryllium pebble beds, 550^{O}C for Ferritic steel, and 900^{O}C for Ceramic Li_4SiO_4). Especially, the temperature of the lithium silicate pebble bed is in the range of 420 °C –742 °C, which are the best temperature windows for extracting tritium in the lithium silicate pebble bed.

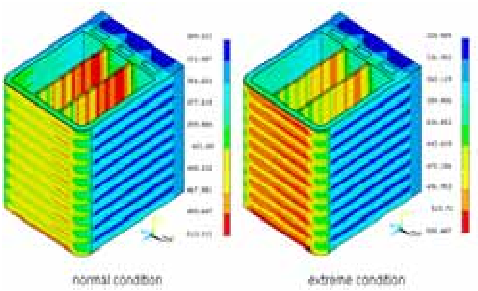

FIG.9 Temperature distribution of FW

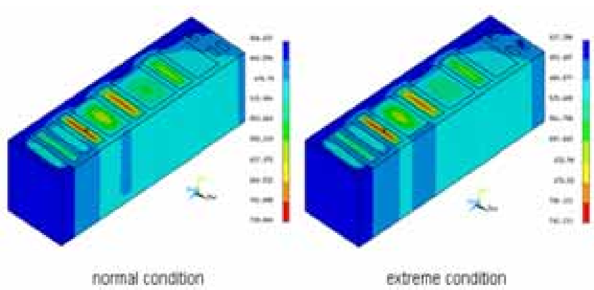

FIG.10 Temperature distribution of sub-module

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 were obtained by means of elastic approach. The thermo-mechanical properties of structure material were used in calculations are Young's modulus of 181.5 (GPa), Poisson ratio of 0.3, and thermal expansion coefficient of 11.9 (10^{-6}/k), and thermal conductivity of 29 (W/mk). As shown in Figs.11-12, Max. equivalent stress of the first wall is 329 MPa. Max. equivalent stress of the cooling plate in the breeding zone amounts to 208 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.

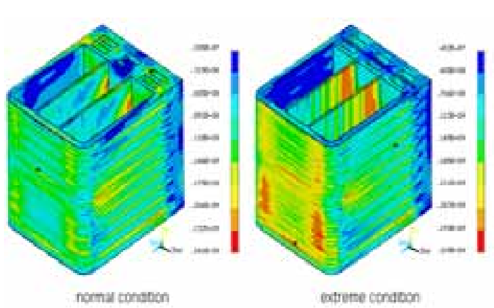

FIG.11 Stress intensity distribution of FW

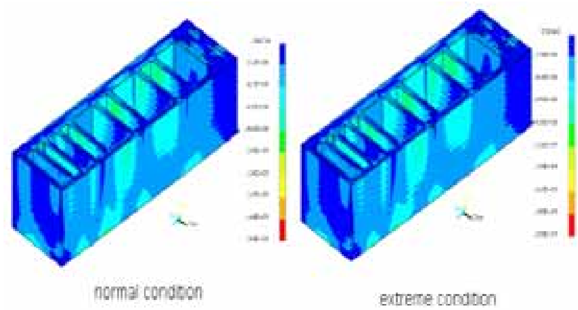

FIG.12 Stress intensity distribution of cooling plate

4. TBM Ancillary system

4.1. Helium cooling subsystem

Preliminary design of the helium cooling subsystem (HCS) have been performed by using ANSYS code. Fig.13 shows the helium cooling system flow diagram. This system includes the primary helium heat transport loop with all components and the secondary heat removal loop. The inlet and outlet temperatures of the helium coolant are

Table 2 Main design parameters of the HCS

Parameters	Values
Operation pressure, MPa	8
Coolant temperature (in/out), ^{O}C	300/500
Power deposition in module, MW	0.84
Diameter of helium tube (OD/ID), mm	85/80
Helium mass flow rate, kg/s	0.842
Helium velocity, m/s	35

300°C and 500°C, respectively. 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 an assumed condition of 35 ^{0}C and 75 ^{0}C. The pressure of the secondary water loop is lower than 1 MPa. Main design parameters of the HCS are listed in Table 2.

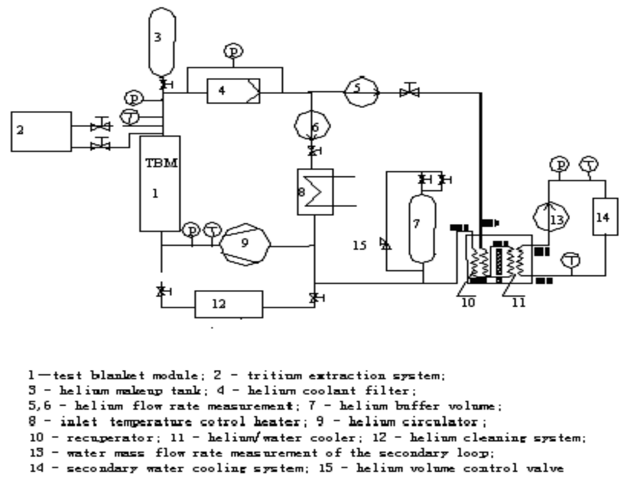

FIG.13 Flow diagram of HCS subsystem

4.2. Tritium extraction subsystem

Main design parameters for the tritium extraction system (TES) are that the composition of purge gas of 0.1%H+He, pressure at the inlet of TBM blanket of 0.12 MPa, extracted amount of tritium of 0.1 g/d, helium mass flow of 0.65g/s, tritium extraction efficiency of ≥95%. Tritium extraction system will be located in ITER's tritium building. The TES design has following features: firstly, a hot Mg bed is used to decomposed HTO released during regeneration of the MSB beds for tritium recovery in HT; secondly, a small size ISS subsystem is designed to separate product HT of the TES for H_2 recycle, thus greatly reduce the amount of the discharged waste H_2.

The main design parameters of the TES are given in table 3. For reasons of radiological safety, the system must be installed in a glove box. A flow diagram of the TES is shown in Fig.14.

Preliminary design of the coolant purification subsystem (CPS) has been preformed. Main parameters of the CPS subsystem are the following, Max. Flux of CPS-450mg/s, coolant pressure of 8 MPa; by-pass line pressure of 1 MPa, tritium extraction efficiency is ≥95%, respectively.

Table 3 Main design parameters for the TES

He Mass Flow	0.85g/s
Swamping Ratio	He: H2=1000:1
Pressure of purge gas at TBM inlet at TBM outlet Pressure drop in TBM	 0.12Mpa 0.10Mpa 0.02Mpa
Temperature of purge gas at TBM outlet	673K
Tritium Generation Rate	0.017g/d
Tritium Extraction Efficiency	≥90%
Leakage rate of the TES	$\leq 1.5\times10^{-8}$ Pa.m^3.s^{-1}

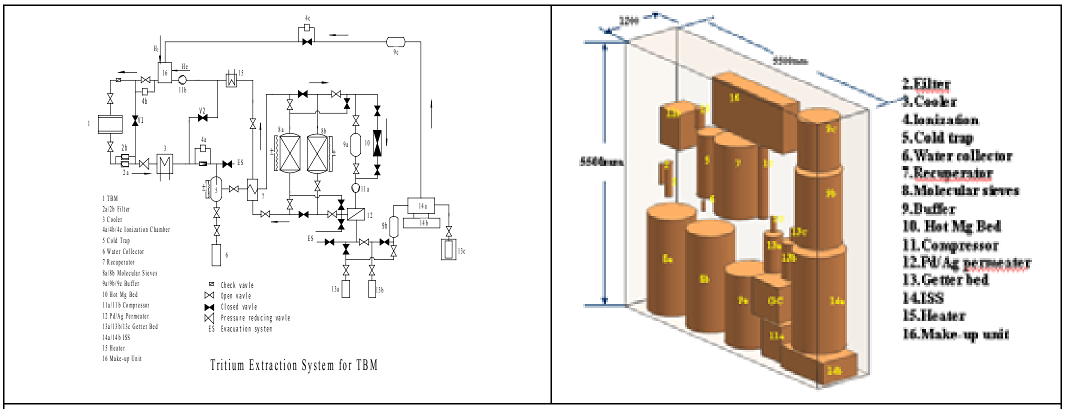

FIG.14. The Flow and layout of Tritium Extraction System (TES)

5. Summary

A design concept for the China 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 technologies. Ii is characterised by simple structure, mature technical. The design description document (DDD) have been carried out in last year. The further HC-SB TBM design works will update and optimize the structure design as well as ancillary subsystem parameters.

Acknowledgment

This work was supported by the National Science Foundation in China (NSFC), under contract No. 10275017.

References

[1] L.V. Boccaccini, Editor, European Helium-cooled Pebble Bed (HC-PB) Test Blanket, ITER Design Description Document, Status 1.12.1998, FZKA 6127, Mach (1999).
[2] K.M. Feng, et al., Design Description Document for the Chinese Helium-cooled Solid Breeder (CH HC-SB) TBMs, (draft design report), Dec. 30(2004).
[3] ODELL, R., et al., CCC-547/ONEDANT, Radiation Shielding Information Center, Oak Ridge National Laboratory, (1989).
[4] ODELL, R., et al., TWODANT: A Code Package for Two-Dimensional, Diffusion-Accelerated, Neutral-Particle Transport, Rep. LA-10049M, Rev. 1, Los Alamos National Lab., NM (1984).
[5] BRIESMEISTER, J., et al., MCNP/4B, A Monte Carlo Neutron and Photon Transport Code, Rep. LA-7396-M, Los Alamos National Lab., NM (1997).
[6] H. Wienke, M. Herman, FENDL/E2.0, The processed cross-section libraries for neutron-photon transport calculations, version 1 of February, 1998, IAEA-NDS-176 Rev. 0.
[7] R. Aymar, ITER reduced cost design, Fusion Engineering and Design, 49-50 (2000) 13-25.
[8] Y.X. Yang, K.M. Feng, and J.H. Huang, CCC-541/FDKR, Radiation Shielding Information Center, Oak Ridge National Laboratory, (1989).
[9] K.Furuta, Y.Oka and S. Kondo, CCC-464/BISON1.5, RSIC, Oak Ridge National Laboratory, (1987).
[10] ANSYS Technical Report, Rev. 5.1, IMAG Industries Co., Feb (1995).
[11] American Society of Mechanical Engineers, ASME, Boiler and Pressure Vessel Code, Section III – Materials, (2004).