Fusion Engineering and Design 114 (2017) 141–156

Contents lists available at ScienceDirect

Fusion Engineering and Design

journal homepage: www.elsevier.com/locate/fusengdes

Full Length Article

Numerical research on the neutronic/thermal-hydraulic/mechanical coupling characteristics of the optimized helium cooled solid breeder blanket for CFETR

Shijie Cui, Dalin Zhang*, Jie Cheng, Wenxi Tian, G.H. Su

School of Nuclear Science and Technology, Xi'an Jiaotong University, Xi'an 710049, China

ARTICLE INFO

Article history:
Received 26 August 2016
Received in revised form 25 October 2016
Accepted 13 December 2016
Available online 24 December 2016

Keywords:
CFETR
Blanket
Coupling analysis
Neutronics
Thermal-hydraulics
Thermo-mechanics

ABSTRACT

As one of the candidate tritium breeding blankets for Chinese Fusion Engineering Test Reactor (CFETR), a conceptual structure of the helium cooled solid breeder blanket has recently been proposed. The neutronic, thermal-hydraulic and mechanical characteristics of the blanket directly affect its tritium breeding and safety performance. Therefore, neutronic/thermal-hydraulic/mechanical coupling analyses are of vital importance for a reliable blanket design. In this work, first, three-dimensional neutronics analysis and optimization of the typical outboard equatorial blanket module (No. 12) were performed for the comprehensive optimal scheme. Then, thermal and fluid dynamic analyses of the scheme under both normal and critical conditions were performed and coupled with the previous neutronic calculation results. With thermal-hydraulic boundaries, thermo-mechanical analyses of the structure materials under normal, critical and blanket over-pressurization conditions were carried out. In addition, several parametric sensitivity studies were also conducted to investigate the influences of the main parameters on the blanket temperature distributions. In this paper, the coupled analyses verify the reasonability of the optimized conceptual design preliminarily and can provide an important reference for the further analysis and optimization design of the CFETR helium cooled solid breeder blanket.

1. Introduction

The Chinese Fusion Engineering Test Reactor (CFETR) is a new test tokamak device based on the International Thermonuclear Experimental Reactor (ITER) and the Chinese Tokamak experiences, which is proposed by the China national integration design group for the magnetic confinement fusion reactor [1]. As one of the critical components of CFETR, blanket is required to achieve the function of tritium self-sufficiency, fusion energy transformation and radiation shielding. As a result, the safety of the blanket system is one of the decisive factors for the normal operation of CFETR. At present, three kinds of breeding blanket concepts including helium-cooled solid breeder blanket [2], water cooled solid breeder blanket [3], and liquid lead-lithium blanket [4], are under development and test in parallel for CFETR. As one of the candidate blankets, a kind of helium cooled solid breeder blanket was proposed for CFETR as shown in Fig. 1 [2], and a series of preliminary analyses have been carried out to access the performances of the conceptual design scheme [5–8]. However, the required tritium breeding ratio (TBR) for CFETR not lower than 1.2 [1] and its complex structure design and highly nonuniform heat deposition require farther thorough coupled neutronic/thermal-hydraulic/mechanical analyses and optimizations at the design phase.

In this paper, the neutronic/thermal-hydraulic/mechanical coupling analyses of the typical helium cooled solid breeder blanket module (No. 12) for CFETR were performed. First, three-dimensional coupled neutronic/photonic model of the blanket was performed using the Monte Carlo N-Particle transport code MCNP with IAEA general purpose neutron sublibrary FENDL-3.0 [9]. The neutronic calculation results and the subsequent thermal calculation results were manually iterated, which made it possible for the coupling of the neutronics and thermal-hydraulics. Then the tritium breeding capability of the scheme was assessed and the adopted three radially arranged U-shaped breeding zones were optimized for higher TBR. Based on the preliminarily optimized blanket structure, the influences of different factors on TBR and nuclear heating rate were investigated for the comprehensive optimal scheme. Then, thermal and fluid dynamic analyses of the scheme were performed by CFX. The temperature distribu-

* Corresponding author at: Xi'an Jiaotong University, No.28 Xianning west road, Beilin County, Xi'an, Shaanxi, 710049, China.
E-mail address: dlzhang@mail.xjtu.edu.cn (D. Zhang).

http://dx.doi.org/10.1016/j.fusengdes.2016.12.013
0920-3796/

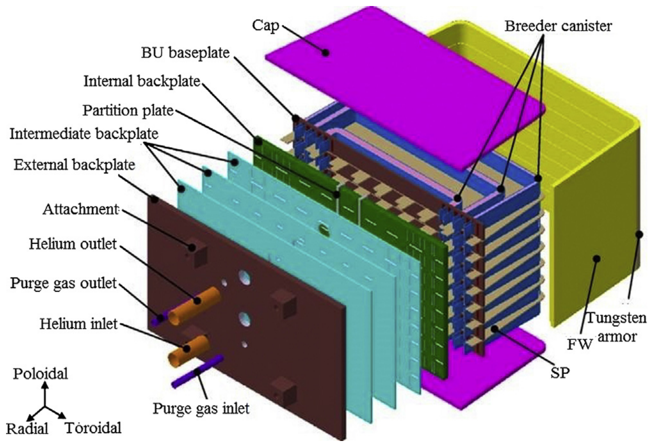

Fig. 1. Schematic view of the typical outboard equatorial helium cooled solid breeder blanket module.

tions and hydraulic behaviors of the whole typical blanket module under both normal and critical conditions were calculated. With thermal-hydraulic boundary conditions, thermo-mechanical analyses of the structure materials under normal, critical and blanket box over-pressurization conditions were carried out using ANSYS Workbench. In addition, several parametric sensitivity studies were conducted to investigate the influences of the main parameters on the temperature distributions of the blanket components. The neutronic/thermal-hydraulic/mechanical coupling analyses verify the reasonability of the optimized conceptual design preliminarily and are significant for the further analyses and optimizations for the CFETR helium cooled solid breeder blanket.

2. Characteristics of the blanket

2.1. Basic structure of the blanket

In this conceptual design of the typical blanket module, the toroidal width of the FW and the outmost backplate is 1448 and 1606 mm, respectively. The radial thickness is 800 mm and the poloidal height is 960 mm. The blanket is mainly composed of the FW, caps, stiffening plates (SPs), breeding zone, multiplier zone, backplates, attachment system, helium coolant and purge gas inlet/outlet pipes. The FW is a U-shaped plate and the front wall is directly facing plasma. There are 45 radial-toroidal-radial cooling channels uniformly arranged in the FW structure material in parallel, in which the high-speed helium coolant flows to take away the power deposition and heat flux derived from the plasma. The helium inlets and outlets are staggered in the two sides of internal backplates to simplify the fabrication, and the coolants in the neighboring channels flow in the opposite directions to achieve a uniform temperature distribution on the FW.

The FW top and bottom are welded to the top and bottom trapezoid caps respectively to form a blanket box, in which seven trapezoid SPs with same internals are welded to the FW internal wall to enhance the blanket structure mechanically. Similar to the FW, helium also flows in the W-shaped internal channels to take away the heat deposited on the caps and SPs. The spaces divided by SPs and FW like drawers are used to accommodate the breeder units (BUs). There are totally 8 BUs in the blanket, the poloidal height of which is 106 mm. According to Ref. [10], breeding zones parallel to FW, in which the Beryllium pebble beds are designed surrounding the lithium ceramic pebble beds, can increase the amount of neutrons participating in the breeding reaction with the lithium and improve the breeding performance. In this design, each BU is subdivided into three parallel layers of lithium ceramic breeding zones and four parallel layers of Beryllium neutron multiplier

Table 1
Main parameters of the original typical outboard blanket module.

	Parameters
Blanket size	1448–1606 mm (toroidal) × 800 mm (radial) × 960 mm (poloidal)
FW	Thickness: 28 mm (3/15/10), radial-toroidal-radial channel: U-shaped, cross section: 15 × 15 mm, pitch: 20 mm, number: 45
BU	Pebble bed: radial thickness: 20/15/180/30/200/45/40 mm, toroidal width: 1448/1330/1290/1180/1110/1000/900 mm Curvature radius of pebble bend: 40/17.5/22.5/22.5/30/30/10 mm Cooling plate: U-shaped, thickness 5 mm Radial-toroidal-radial channel: U-shaped, cross section: 6.1 mm × 2.6 mm, pitch: 10.1 mm, number: 80
Cap	Thickness: 28 mm (12/4/12), channel: W-shaped, cross section: 6.5 × 4 mm, pitch: 14.5 mm
SP	Thickness:8 mm (2/4/2), channel: W-shaped, cross section: 6.5 × 4 mm, pitch: 14.5 mm
Backplate	Radial thickness: 35/10/10/10/40 mm
Pipe	Diameter of helium inlet/outlet: 80 mm Diameter of purge gas inlet/outlet: 35 mm

zones, which are separated by six parallel U-shaped cooling plates (CPs) with internal cooling channels. Similar to the FW, helium coolants in the neighboring CPs also flow alternatively to achieve a uniform temperature distribution on the whole blanket. Considering the practical complexities of the manufacturing of manifolds, the coolants in the neighboring channels of one CP are designed to flow in the same directions [2]. The main parameters of the original typical outboard blanket module are listed in Table 1.

2.2. Material selection

The original conceptual design scheme of the helium cooled solid breeder blanket adopts Reduced Activation Ferritic/Martensitic (RAFM) steel as structure material. Lithium ceramic of Li_4SiO_4 with 90% ^{6}Li enrichment is used as a tritium breeder in the form of pebbles with packing factor about 62%. Beryllium pebbles are adopted as neutron multiplier with packing factor about 80%. Helium gas of 8 MPa pressure is employed as the only coolant to extract the deposited heat in the blanket, and the tritium produced in the BUs is taken out by 0.12 MPa purge gas (He + 0.1% vol. H_2). The thermal physical properties of RAFM steel, ceramic breeder (CB) pebble and Beryllium pebble beds were taken from the physical properties experiment report of Southwestern Institute of Physics [11]. And the temperature dependent thermal physical properties of helium were taken from Ref. [12].

2.3. Helium coolant and purge gas flow scheme

Fig. 2 shows the flow scheme of helium coolant and purge gas in the blanket. It can be seen that the helium coolant and purge gas flow separately in the blanket. Firstly, the cold helium at 300 °C under the pressure of 8 MPa is fed into the manifold 1 through the helium inlet and then distributed into the FW cooling channels at the two sides of the internal backplate. Then the helium out of the FW is gathered in the manifold 2 and flows in parallel into the channels inside the caps and SPs at the ratio of 22.2%:77.8%. After that, the helium is collected in the manifold 3 and flows into the CPs cooling channels simultaneously. Finally, the hot helium out of the CPs is gathered in the manifold 4 and flows out of the blanket through the helium outlet.

For the purge gas, as shown in Fig. 2, it firstly enters the purge gas inlet manifold through the purge gas inlet penetrating the manifold 1–4. Then it passes through the polyporous plates which are installed on the BU baseplate and sweeps all of the ceramic breeder and Beryllium pebble beds parallelly along the radial-toroidal-radial direction defined by the BU canisters. Finally, it is collected in