ITER Shield Blanket Design Activities At SWIP

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Abstract. Design works on ITER shield blanket (BM) started at Southwestern Institute of Physics (SWIP) in the end of 2003 after China joined the ITER negotiation. Two shield blanket modules located in the Neutral Beam (NB) port region were studied at SWIP in cooperation with ITER International Team (IT). One module has a regular shape like the other outboard modules in the normal places. Another module is just beside the NB opening, which has a dog-leg shape and two Beryllium protected surfaces in both NB duct side and main vacuum vessel side. The regular module has the same structure as other regular modules designed by the IT. As the NB port is located at the equatorial level of the tokamak machine, the modules will suffer the highest nuclear heating. Detailed hydraulic and thermal/stress analysis were carried out to optimize the performance and some improvements were made. The special module is configured with 4 FWs, i.e., 3 L-shape FWs in NB duct side and one flat FW in the main vacuum vessel side. The overall structure is compatible with the regular module. The cooling configuration of first L-shape FW, which face both plasma and NB, is also given. Thermal/stress analysis shows that the straight tubes in the vertical part of the FW is feasible, which is in favor of fabrication.

1. Introduction

According to the ITER procurement allocation, China Participating Team (PT), as well as Europe, Korea, Japan, Russia and United State Participate Teams, will fabricate the shield blanket modules for ITER. The design and R&D works on ITER shield blanket modules started at SWIP in the end of 2003 after China joined the ITER negotiation. Since then, two shield blanket modules near Neutral Beam (NB) port were studied and the relevant design tasks were finished through the cooperation of the ITER International Team (IT) and CN PT. The works are conducted by the ITER task agreements. The interfaces of the modules are defined by the ITER IT and the detail cooling configuration, hydraulic, thermal and stress analysis are carried out at SWIP. Some results from the design activities are given in this paper.

Blanket system is one of the key components in the ITER machine, which provides thermal and nuclear shielding to the vacuum vessel and external machine components. For 500MW fusion power operation of ITER, the total thermal load on blanket system is estimated to be 690 MW. The structure of the shield blanket system was developed during ITER EDA period and improved by ITER international team in recent years [1,2]. To minimize the fabrication and maintenance cost, the ITER blanket system adopts modular configuration. It is segmented into 18 modules covering the vacuum vessel in poloidal direction. In toroidal direction, each inboard module takes 20° sector (18 modules) and outboard module takes 10° sector (36 modules). Normally, the blanket module has a thickness of ~450 mm and less than 4.5 t in weight. Each module is mechanically attached to the vacuum vessel though 4 flexible support. As the components will be activated in the D-T operation, the maintenance of the blanket modules will employ a remote controlled robot. The details of the blanket system are depicted in the reference [2].

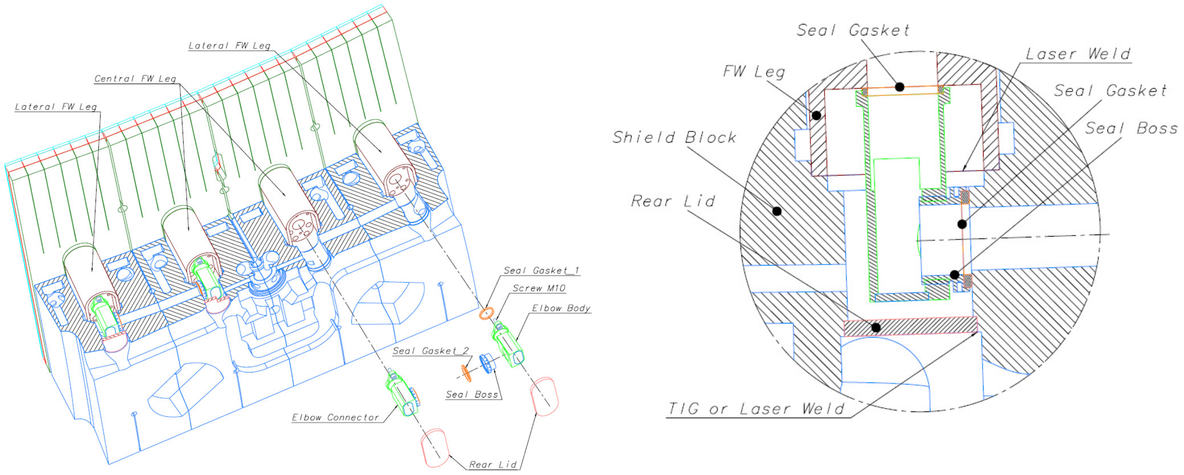
During ITER EDA, two design options of the shield blanket modules were developed and the relevant R&D works were carried out to demonstrate the fabrication feasibility in EU and Japan respectively[1,3,4]. In both options, the modules comprised Beryllium protected First Wall to face the plasma and thick shield block made by Stainless Steel to provide nuclear shielding. In the option A (by EU), the design of shield block was based on the powder HIP technology, and in the option B (by JP), the shield block was based on Forge + Weld. After EDA, the design of option B was improved by ITER IT and the improved structure was given in the ITER DDD 1.6 edition 2004 [2]. Some typical modules were designed in detail by the ITER IT and other PTs in recent years. As the ITER shield blanket system consists of various types of blanket modules with different geometry and thermal loads, detailed design and optimization are needed for each type of the blanket modules before fabrication starts.

The NB ports provide the passage for neutral beam heating and diagnostics. There are 18 modules in the NB region with 12 different types. Two modules were studied in this work. One has a regular shape like the other outboard modules in normal places. The basic features depicted in the ITER DDD 1.6 were applied in the configuration of the module. As the module is located at the equatorial level of the ITER machine, it will suffer the highest neutron wall loading, i.e. 0.78 $MW.m^{-2}$. Detail hydraulic/thermal /stress analysis were carried out to optimize the performance of the module. Another is a special module just beside NB opening, which has a dog-leg shape and two Beryllium protected surfaces in both NB duct and main vacuum vessel. The layout of the FW segmentation and support is compatible with the regular modules to utilize the same maintenance tools. In the configuration of the cooling passages in the FWs, both plasma and NB loadings are considered.

2. The regular module in the NB Region

2.1. Overall Structure of the Regular Module

The overall structure of the regular module in the NB region is shown in Figure 1 (a) and it is similar to that of the other regular modules given in ITER DDD 1.6 2004 by the ITER IT. The total weight of the module is about 3850 kg. It consists of 4 replaceable first walls (FW). Each FW is mounted on the rear side of shield block through a central leg. The leg provides mechanical support for the FW, and also electrical and hydraulic connections between FW and SB. To withstand the electromagnetic force during the VDE and plasma disruption, the shape of the leg is racetrack section with 100 mm straight length and 60 mm radius of the two half circles.



(a) *(b)*

Fig. 1. (a) Illustration of the overall structure of the regular module.
(b) Illustration of the inner connector in the FW house

During operation, the module is cooled by water at 100°C and 3 MPa. The mass flow rate is estimated as ~8 kg/s. Cooling water flows into and out of the shield blanket through a coaxial connector at the back center of the shield block. The two cooling branches at the left and the right of the module are symmetric. The water passages between FWs and FW/SB are drilled inside the shield block instead of using additional tubes. The FW house also forms part of the cooling passage. A set of inner connector is bolted inside the FW house to feed the FW, as shown in Fig. 1(b). The overall water flow path is as follows: Coaxial Connector -> Central FW -> Lateral FW -> Lateral SB -> Central SB -> Coaxial Connector. The detail flow pattern and hydraulic analysis are given in the design report [5,6], and some results are also summarized in the reference [7].

2.2. The FW of the Regular Module

To reduce fabrication and maintenance costs, the surface of the FW is flat and the 4 FW panels are identical. With this configuration, one prepared spare FW panel can replace a damaged FW at any place. The FW panels are cut to fingers of 60 mm width to reduce the magnetic force caused by induced currents. Different access holes/slots (for assembly and maintenance of the BM) are located at the edges of lateral fingers, where cooling channels are properly shifted. The structure of the FW is illustrated in Fig. 2(a).

As the FW directly faces the plasma, the plasma-surface interaction is one of the key issues in the selection of materials. Beryllium was selected as plasma facing material by ITER, due to its low Z and good thermal properties. The Be tiles are 10 mm in thickness. Cu alloy is selected as heat sink material, in which stainless steel tubes of 10 mm in diameter are embedded for cooling. The 49 mm thick stainless steel back plate with ϕ24 mm cooling holes is to withstand the electromagnetic force.

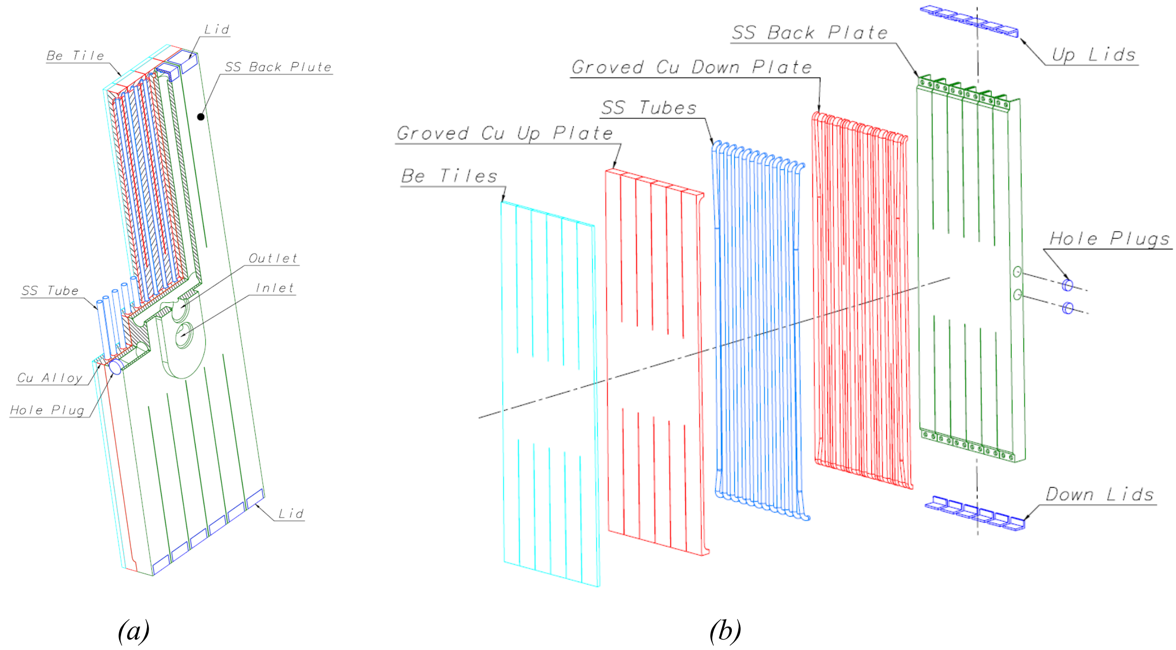


Fig. 2 (a) Illustration of FW structure
(b) Illustration of FW components

The components of the FW are illustrated in Fig. 2(b). The joining method of Be/Cu/SS is proposed as hot isostatic press (HIP). During ITER EDA, the relevant technology was developed in some PTs [8,9]. As Be easily forms brittle intermetallic phases with Cu alloys, the joining technology is believed as a key factor in the fabrication of FW. The R&D works for the fabrication of FW is still undergoing in China[11,12], as well as in the other PTs, which is conducted by the ITER "FW fabrication qualification program".

2.3. The SB of the Regular Module

The main function of the shield block is to provide nuclear shielding. The body is segmented into 4 similar parts made from forged stainless steel and each part is about 750 kg in weight. Cooling passages are drilled in the forged blocks. The 4 parts are joined together by electron beam (EB) welding. Figure 3(a) illustrates the shield block.

The cooling passages in the left and right side of the module is symmetric. In each side, there are 8 series of radial holes. In each series, the cooling water flows through the radial holes in parallel from poloidal holes to front manifold, or in the reverse direction. Fig. 3 (b) shows the cooling passages in the right central part of the SB and it is similar in the other parts. Special flow drives, as shown in Fig. 3 (c) and (d), are inserted into the radial holes to increase the heat transfer coefficient (HTC). The front side of the shield block is cooled by the front manifold manifolds.

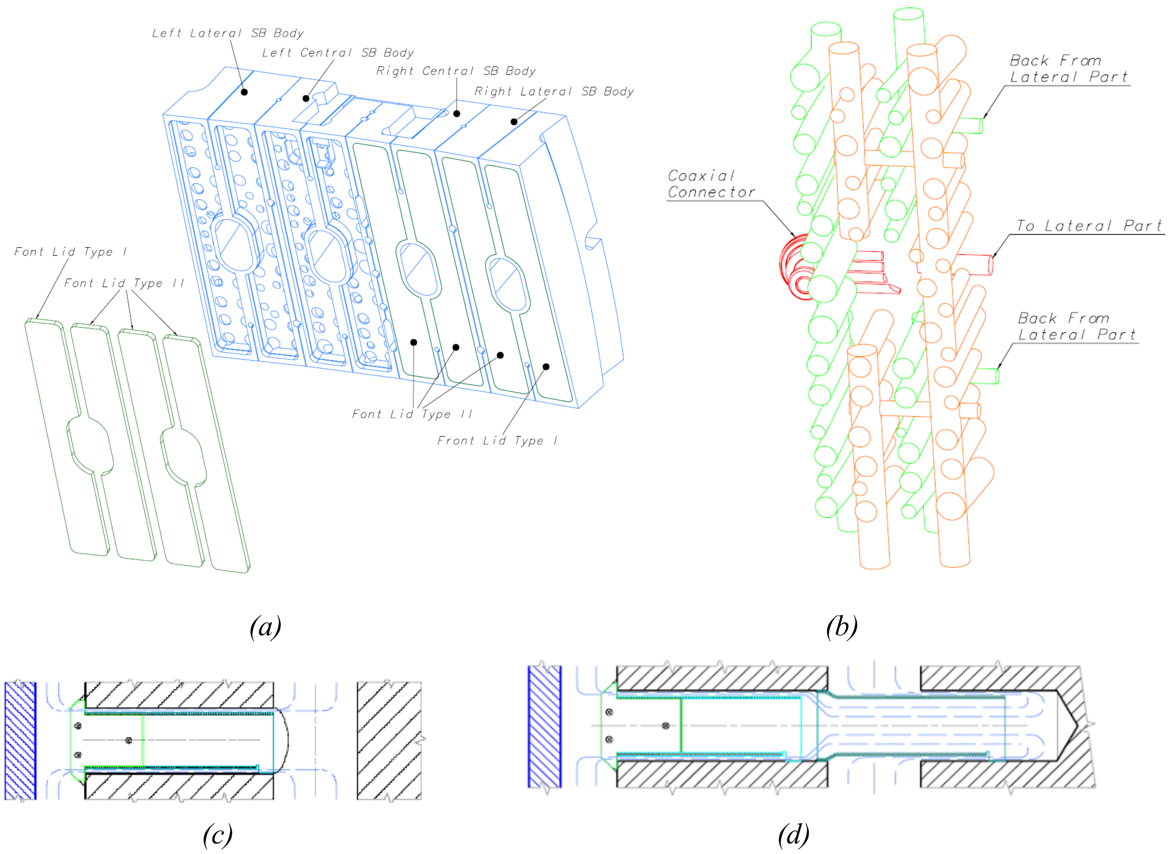


Fig. 3 (a) Illustration of the shield block (Flow drives not shown)
(b) Illustration of the cooling holes in the Right Central SB Body
(c) Illustration of the flow drive of type I
(d) Illustration of the flow drive of type II

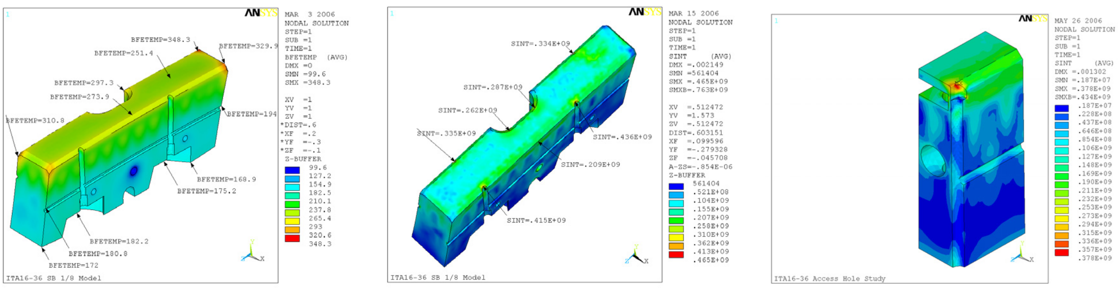
In the design of FDR 2001, the diameter of poloidal hole is 45 mm and the inserting of flow drives caused large pressure drop in the poloidal hole [1]. In current design, the same as in the FDR 2004, the diameter of poloidal hole increased to 60 mm. Hydraulic analysis shows that the pressure drop is about 1000 Pa (no longer an issue) [6,7].

As nuclear irradiation in SB decays exponentially with the distance from FW, the front side has much higher thermal load than the rear side. The thermal loads will cause the bowing of the module and high thermal stress in the front region. In the original configuration of the module, there existed 423°C thermal spots at the font corners and 726 MPa peak stress intensity at the access holes [5].

3D FEM analysis was employed to study the radial cooling of SB and to improve the performance. The detail of the analysis is given in the reference [5,6] and some results are summarized here: (i) The cooling in radial hole is enough efficient when the span (i.e. the distance between the axes) of radial holes ranges from 70 mm to 90 mm. (ii) 15mm chamfers on the front edge are effective to diminish thermal spots and thermal stress in the front region. An additional slit across the FW house was also suggested. (iii) Slightly changing the depth of the radial holes had little effect on the stress distribution, but significantly on the bowing of the module. (iv) The heat transfer coefficient varing from 6000 ~12000 $W.m^{-1}.K^{-1}$ has little effects on the performance. But the heat transfer coefficient in the front manifold changing from 3800 to 8000 $W.m^{-1}.K^{-1}$ can significantly reduce the thermal stress in the front region.

Due to the back cuttings and FW house, the radial holes in the module cannot be configured with only ϕ45 mm hole as periodic distribution. Some holes have to be configured at fixed places to introduce water into the rear side of the shield block. The position and diameter of other holes were optimized according to 3D FEM analysis.

The overall performance of the module was confirmed by a FEM model which covers 1/8 size of the full module. Temperature distribution, stress distribution and thermal deformation were checked. The HTC in each radial hole was obtained from the hydraulic analysis[6,7], and the nuclear loads are as defined in the reference [13]. Fig. 4 (a) and (b) show the distributions of temperature and stress intensity respectively. The maxium temperature decreased to 348°C and the peak stress at the ϕ30 mm access hole decreased to 436 MPa. Thermal bowing of the models were controlled well within the requirement of flexible support [2]. Further studies shows that the stress concentration can be largely relieved by a 22 mm × 30 mm elongated pocket in the front of the access hole, as shown in Fig. 4 (c). The peak stress intensity further decreased to 378 MPa, which is within the limit of the 3Sm limit [13].



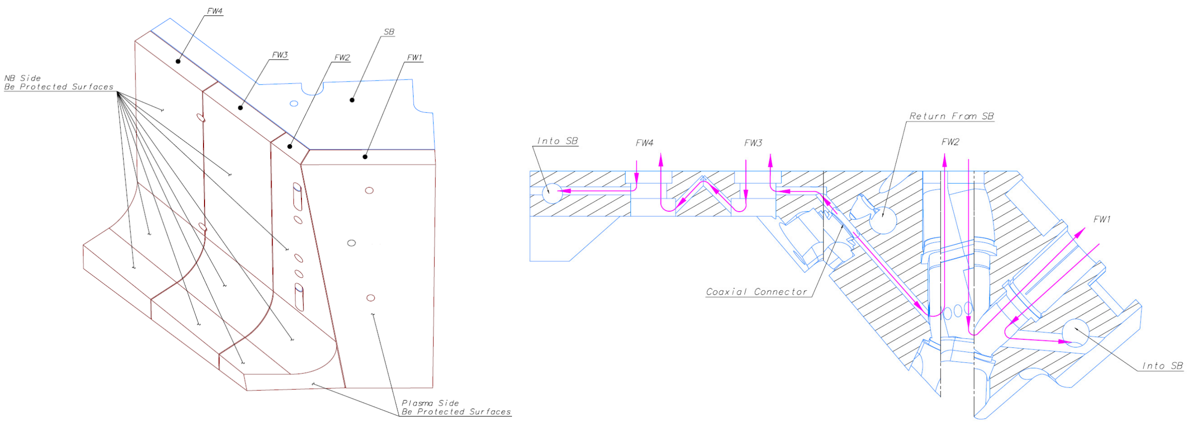
(a) *(b)* *(c)*

Fig. 4 (a) The temperature distribution of the 1/8 model
(b) The stress intensity distribution of the 1/8 model
(c) The stress intensity distribution with an elongated pocket

3. The Special module in the NB region

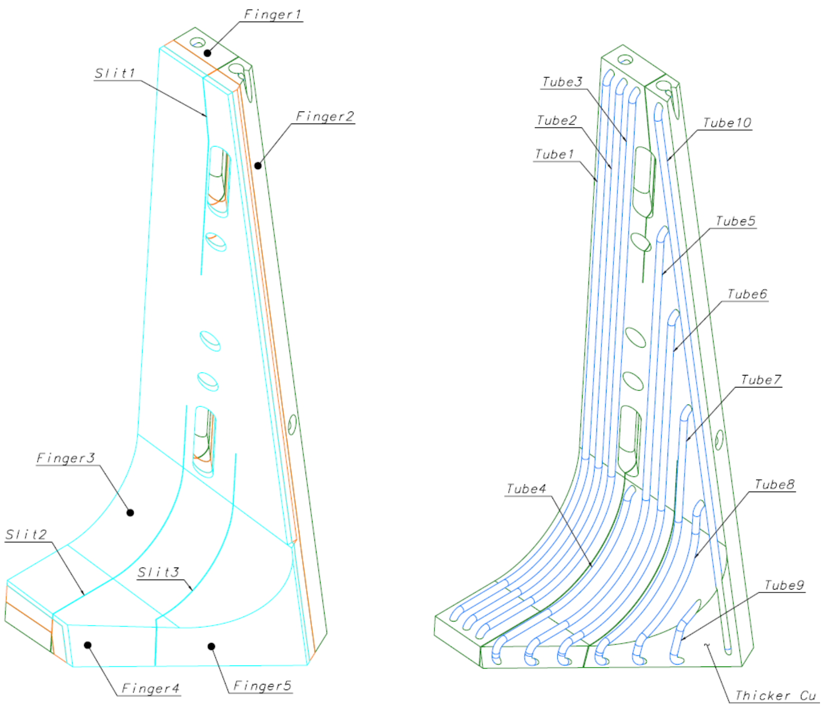
The special module in the corner of the NB opening has a dog-leg-like shape. The surface on the NB duct side will suffer the thermal load caused by the divergence of NB, as well as plasma thermal load. The thermal load on the plasma side is similar to that of a regular module. The surfaces on both sides are protected by Be armors.

According to the IT requirement, the FW segmentation and mechanical support should be compatible with the standard modules, thus the same assembly and maintenance tools will be also available for the special module. The special module is configured as 4 replaceable FWs, i.e., 3 L-shape FWs in NB duct side and one flat FW in the main vacuum chamber side, as shown in Fig. 5 (a). The size of FW panels is suitable to fabrication with middle sized HIP facilities. After the IT allow to reduce the size of Electrical Connector of the module, the central support FW legs were configured for all panels. The overall cooling water flow pattern, as shown in Fig. 5 (b) is also similar to the regular module.



(a) *(b)*

Fig. 5. (a) Illustration of the overall structure of the special module
(b) Illustration of the overall flow pattern in the special module



(a) *(b)*

Fig. 6 (a) Illustration of the first L-shape FW.
(b) Illustration of the cooling tubes in the first L-shape FW

As the first L-shaped FW (i.e. FW2 in Fig. 6 (a)) has a variable width and inclined access cuttings, the cooling configuration is difficult. Several configurations were studied and FEM were used to analyze the performance under NB and plasma heating, together with nuclear heating. Results shows that it is possible to configure straight tubes instead of curved tubes to surround the access cuttings in the vertical region, as shown in Fig. 6 (b). The straight tube configuration is in favor of HIP process. Some vertical tubes are extended with special fingertips to cool the plasma side. The FW is also cut to fingers to diminish the electromagnetic force, as shown in Fig. 5(a). The cooling circuit in the L-shape FW is also simple and standard-FW-like. Hydraulic analysis showed that the pressure drop in FW2 was little higher than that in Standard FWs due to the longer length of the cooling tubes, but it is acceptable. The detail of the configuration of the special module is given in the reference [14].

4. Conclusion

The design activities are carried out at SWIP in recent years in cooperation with the ITER IT. From the design works, the layout of the two shield blanket modules in the NB region is given. The regular module is configured based on the structure given in the ITER DDD1.6 2004. The performance of the module is optimized according to the detail hydraulic/thermal/stress analysis. The cooling of the module is robust and the performance can meet the ITER design requirements. The configuration of the special module is compatible with the regular modules and the same tools can be utilized in the maintenance process. The cooling passages in the L-shape special FW is also similar to the standard FW. Straight tubes are configured in the vertical side of the FW, which is in favor of fabrication.

Reference

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