Abstract—This paper summarizes the status of the ITER blanket system design and describes some of the key R&D activities in support of the design with the goal of starting procurement in the first half of 2013.

Keywords - ITER, blanket system, first wall

I. INTRODUCTION

The Blanket System provides a physical boundary for the plasma transients and contributes to the thermal and nuclear shielding of the Vacuum Vessel and external ITER components [1,2]. It covers ~600 m² and consists of Blanket Modules (BM) comprising two major components: a plasma facing First Wall (FW) panel and a Shield Block (SB). Each BM is about (1 m x 1.4 m x 0.5 m) and is attached to the vacuum vessel through a mechanical attachment system of flexible supports and a system of keys. Each BM has electrical straps providing electrical connection to the vacuum vessel. Cooling water (3 MPa and 70°C) is supplied to the BM is by manifolds supported off the vacuum vessel behind or to the side of the SB and is designed to remove up to 736 MW of thermal power from the blanket. The BMs are segmented into 18 poloidal locations: rows 1 to 6 are the inboard region, rows 7 to 10 are the upper region and rows 11 to 18 are the outboard region. Figure 1 illustrates the arrangement of blanket modules in a sector.

The ITER blanket design has substantially evolved since the ITER design review of 2007 [2]. Two major incentives for the design changes have been the need to account for large plasma heat fluxes to the first wall and the need for efficient maintenance of first wall components. This design was presented at the Conceptual Design Review (CDR) in February 2010 and was accepted in the ITER baseline in May 2010.

Since the CDR and in preparation for the Preliminary Design Review scheduled for late 2011, the blanket effort has focused on resolving the key issues brought out as part of the CDR effort and findings, particularly on improving the design of the first wall and shield block attachments to better accommodate the anticipated electromagnetic (EM) loads. This led to designing a first wall mechanical structure consisting of a beam placed in the poloidal direction. The plasma-facing units are attached onto the beam and fed by water manifolds from it. The beam is attached to the shield block by a single preloaded bolt located deep into the shield block. The SB attachment design has also been revised to enhance its load carrying capability including the use of a thicker bolt for the axial flexible attachments, while the shield block design has been optimized to include slits to reduce the eddy loads.

This paper describes these key design improvements in each blanket system component and summarizes the R&D being conducted in support of the design.

Figure 1. Schematic of Blanket Sector showing Blanket modules in inboard and outboard regions

II. FIRST WALL

The ITER First Wall (FW) has undergone in-depth conceptual evolution, after the recommendation of the 2007 ITER design review [3]. Requirements for handling of parallel heat loads from the plasma and credible maintainability have
been included as strong design drivers of the FW panel design. To allow in-vessel maintenance, one single independent FW panel covers each shield block, as illustrated in Figure 2 for BM 1.

![Figure 2. Schematic of Blanket Module 1 showing the FW panel and the Shield Block](image)

The panel has its own supporting structure, a beam oriented in the poloidal direction, which is embedded into the shield block and held by a deep bolt. Plasma facing units are attached to the beam and are toroidally-oriented to equalize as much as possible the distribution of thermal load. The design lifetime was reduced to 15,000 plasma discharges, and the FW is now classified as a component which will be replaced at least once during the lifetime of ITER and for which in-vessel maintenance is explicitly planned. The FW panels are shaped to avoid high heat loads in case of panel to panel misalignment (5 mm step). The plasma facing units consist of fingers to reduce eddy current-related loads. They can be manufactured and inspected separately and then assembled into the support beam. The cooling circuits of the fingers are then joined to the cooling circuit of the central beam using a bore welding process.

Accommodating the high heat fluxes resulting in some areas (in particular in the inboard and outboard for start-up and shut-down and in the upper region near the secondary X-point) has necessitated the use of “enhanced heat flux” panels capable of accommodating an incident heat flux of up to 5 MW/m² in steady state. “Normal heat flux” panels, which had been developed and well tested for a heat flux of the order of 1-2 MW/m², are kept in the other locations [4]. Two types of fingers are considered for these different heat flux levels: (i) SS 316L(N)-IG tube for normal heat flux FW panels, in which the SS tubes are embedded into a copper alloy (CuCrZr); and (ii) CuCrZr alloy hypervapotron channels for enhanced heat flux FW panels. The joining of beryllium armor tiles to the CuCrZr heat sink is a critical process that has been validated through a number of processing techniques, including Hot Isostatic Pressing (HIP) and fast Brazing [5,6,7].

The FW shaping design requires a compromise between the conflicting requirements for accommodation of steady state and transient loads. A shaped surface increases the heat loads which are due to plasma particles following field lines compared to a perfectly toroidal surface. Limited damage is acceptable during rare events, while steady state heat loads must be maintained for long periods within what is conventionally allowable for beryllium clad components (1-5 MW/m²).

Detailed blanket design activities are on-going in parallel with supporting analyses to consolidate the approach adopted. They address electromechanical, thermal, thermo-hydraulic and structural aspects. Loads, allowances and criteria come from the general ITER requirements, the blanket system requirement documents, the load specifications and/or the ITER Structural Design Criteria for in-vessel components [8]. Design cases are categorized according to their probability of occurrence (Categories I to IV), and allowable stress or temperature levels depend on the event category and the capability of the design to withstand the design number of cycles for each of the events must be demonstrated.

![Figure 3. Schematic of EHF FW finger.](image)

![Figure 4. Example thermal stress results for FW finger.](image)
For example, the flow through the hypervapotron fingers of the EHF FW panel (shown in Figure 3) is about 2 m/s (with 2 fingers in series). With a coolant inlet temperature of 70°C and pressure of 2.5 MPa, the CHF margin is 1.4 for a maximum heat flux of 5 MW/m². Results from the thermo-mechanical analysis of this EHF hypervapotron finger with Be tiles are illustrated in Figure 4 in terms of the thermal stresses. All general stresses are within limits and the corresponding maximum temperatures of the Be, CuCrZr and SS 316L(N)-IG are about 550°C, 285°C and 274°C, respectively.

III. SHIELD BLOCK

The main function of the SB is to provide nuclear shielding and supply the FW panel with cooling water. It is required to accommodate all the components located on the vacuum vessel (in particular the in-vessel coils and the diagnostics). The steel/water ratio has been optimized with respect to neutron shielding to about 85/15. This ratio is achieved by optimizing the number of poloidal cooling channels and their size within the SB. To further improve the nuclear shielding of the TF coils, the thickness of some of the inboard blanket modules was increased. A number of deep slits are machined into the SB to reduce the impact of the EM loads on the structural loads of the support system and vacuum vessel. As an example, a schematic of SB 1 is shown in Figure 5.

![Figure 5. Schematic of Shield Block 1.](image)

The front face of the SB has a much higher nuclear heating than the rear side. Holes of 12-mm diameter are drilled to cool the front part, which are fed in parallel with an average water velocity of about 0.8 m/s. In the rear part, larger holes are drilled to both distribute water and cool the back part. Water headers are machined on the side of the module with 10 mm welded cover plates. The basic fabrication method for a SB starts from either a single or multiple-forged steel blocks and includes drilling of holes, welding of the cover plates of the water headers, and final machining of the interfaces.

Thermo-mechanical analysis indicates that the stress levels are acceptable and that the temperature level is about 350°C or less, as illustrated by the example results for SB 1 shown in Figures 6 and 7.

![Figure 6. Example temperature distribution in Shield Block 1.](image)

![Figure 7. Example thermal stress distribution in Shield Block 1](image)

IV. ATTACHMENT

As illustrated in Figure 5, the SB is mechanically attached to the vacuum vessel via four axial supports and a system of keys. Details of the axial support are shown in Figures 8 and 9. It consists of a flexible cartridge attached on one end to the vacuum vessel and bolted to the shield block on the other end with a mechanical pre-load of up to 800 kN. Both the cartridge and bolt are made of Inconel-718 for high strength. The supports are designed to take radial loads up to 600 kN (Category III with very few occurrences) including dynamic amplification factor (DAF). Electrical insulation coatings are applied to the mechanical attachments to prevent electrical...
current flowing through the supports and to monitor the EM
loads on the blanket. The flexible supports are located at the
rear of SB, where the nuclear irradiation is lower. The flexible
supports are also used to compensate the radial positioning of
the SB on the vacuum vessel wall by means of custom
machining. An adjustment of up to ±10 mm in the axial
direction and ±5 mm transversely (on key pads) are built into
the design of the supports for the custom machining process.

Figure 8. Schematic of a flexible support.

In the outboard region, the keys are built as stubs
centric with the flexible supports, as illustrated in Figure
12. Limit analysis indicates a 30-45% margin for the keys and
pads. Electrical insulation and low friction coatings are also
utilized in a similar arrangement to that of the inboard region.
Rectangular key pads are also located on the shield block side
and can be custom-machined to recover manufacturing
tolerances of the vacuum vessel and SB.

Each inboard module has two inter-modular keys designed
to react the radial torque and the poloidal forces, whilst a
centering key reacts the toroidal forces. The inter-modular
keys are fitted with low friction coated bronze pads to allow
sliding of the module interfaces during relative thermal
expansion. The pads are attached to the shield block side and
are electrical isolated of by the application of an insulating
ceramic coating on the internal surfaces of the pad.

As illustrated in Figures 10 and 11, analysis of the inter-
modular keys indicate stresses above yield (~172 MPa at
100°C) in the case of a Category III load with an equivalent
poloidal force of 1.725 MN (including DAF). However, a
limit analysis with 5% plastic strain indicates a reasonable
load factor of 1.5 for the pads and 1.9 for the neck of the key.

Figure 10. Model of an inter-modular key with two isolated circular pads on
each side contacting adjacent modules.

Figure 9. Cross-section of a flexible support.

Figure 11. Von Mises stress distribution on an inter-modular key under
Category III load conditions of 1.725 MN (from eddy currents)
Each SB is electrically joined to the vacuum vessel by electrical straps (see Figs. 5, 12 and 13). The straps are formed and louvered from two sheets of CuCrZr alloy to achieve flexibility in all three directions. One electrical connection can handle up to 180 kA of electrical current.

A coaxial hydraulic water connector is utilized to provide the interface between the cooling circuit manifolds and the SB (see Fig. 5). The coaxial connector integrates the inlet and outlet of the cooling circuit thus minimizing the number of seal welds required. This coaxial connector is located in the center of the SB where the thermal displacements are at their lowest and between the two electrical straps where the protection against the halo currents is the highest. A single point connection also helps to minimize the eddy current. The coaxial connector is designed to be initially installed and removed for maintenance requirements using remote handling tools. Connection of the cooling circuit to the hydraulic connector is by Laser or TIG welding and a bore cutting tool is utilized for removal.

V. SUPPORTING R&D

A detailed R&D program has been planned in support of the design, covering a range of key topics such as critical heat flux (CHF) tests on FW mock-ups, experimental determination of the behavior of the attachment and insulating layer under prototypical conditions, material testing under irradiation, and demonstration of the different remote handling procedures. A major goal of the R&D effort is to converge on a qualification program for the SB and FW panels (with full-scale SB prototypes and FW semi-prototypes) by the procuring Domestic Agencies (Korea and China for the SB, the EU for the NHF FW Panels, and the RF and China for the EHF First Wall panels). The primary objective of the qualification program is to demonstrate that the supplying DA can provide FW and SB components of acceptable quality. The components must also be capable of successfully passing the formal test program including heat flux tests in the case of the FW panel. Acceptance criteria for manufacturing and test of the semi-prototype shall be demonstrated through this formal qualification program and prototypical thermal flux levels shall be sustained over the life cycle. The R&D program in support of the EHF hypervapotron CHF testing is illustrated in Figures 14 and 15 [9]. This has resulted in confirming the CHF margin of 1.4 for the EHF FW under an incident heat flux of 5 MW/m².
CONCLUSIONS

The Blanket system is one of the most technically challenging components of the ITER machine, having to accommodate high heat fluxes from the plasma, large electromagnetic loads during off-normal events and demanding interfaces with many key components (in particular the vacuum vessel and in-vessel coils) and the plasma.

The Blanket system has been subject to a substantial re-design following the ITER Design Review of 2007. The Blanket CDR in February 2009 has confirmed the correctness and the effort is now focused on finalizing the design work and the supporting R&D program. This paper has described the present status of the design showing that a viable solution exists, which is compatible with the ITER requirements and existing manufacturing technologies.

The Blanket design will be subjected to a Preliminary Design Review in late 2011 and a Final Design Review in late 2012. The procurement is planned to start in early 2013 and should last till 2019. In parallel to that, the construction technologies are undergoing a formal qualification process by the manufacturing and testing of full-scale semi-prototypes.

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