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ITER plasma-facing components

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ABSTRACT

The ITER plasma-facing components directly face the thermonuclear plasma and include the divertor, the blanket and the test blanket modules with their corresponding frames.

The divertor is located at the bottom of the plasma chamber and is aimed at exhausting the major part of the plasma thermal power (including alpha power) and at minimising the helium and impurity content in the plasma.

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 machine components. It consists of modular shielding elements known as blanket modules which are attached to the vacuum vessel. Each blanket module consists of two major components: a plasma-facing first wall panel and a shield block.

The test blanket modules are mock-ups of DEMO breeding blankets. There are three ITER equatorial ports devoted to test blanket modules, each of them providing for the allocation of two breeding modules inserted in a steel frame and in front of a shield block.

1. Introduction

The ITER plasma-facing components (PFCs) directly face the thermonuclear plasma and cover an area of about 850 m^2. They include the divertor, the blanket and the test blanket modules with their corresponding frames (Fig. 1).

The main functions of the PFCs are:

- Absorb the radiated and conducted heat from the plasma and contribute to the absorption of neutronic heating;
- Minimize the plasma impurity content;
- Provide limiting surfaces that define the plasma boundary during startup and shutdown;
- Contribute to the plasma passive stabilization.

The PFCs consist of a plasma facing material (armour) mounted onto a heat sink and supported by a structural/shielding material. The total thermal power, which is deposited onto the PFCs, can reach a value of 850 MW and is removed by pressurized water.

2. Blanket system

2.1. Overview

The basic function of the blanket module (BM) system is to provide the main thermal and nuclear shielding of the vacuum vessel (VV) and the ex-vessel components such as the poloidal field (PF) and toroidal field (TF) coils during plasma operations. The blanket system is also a plasma limiter and therefore its plasma-facing part is designed to withstand the plasma heat loads generated during the transition phases including plasma “start up” and “shut down” (Table 1).

This system comprises two different sub-systems that cover the ITER inner vessel wall. The wall-mounted and port-mounted BMs, covering ~600 m^2 and ~30 m^2, respectively. The former consist of two major components, a plasma-facing first wall (FW) panel and its supporting shield block (SB). Each BM is attached to the VV through a mechanical attachment system of flexible supports and a system of keys. Each BM has electrical straps providing electrical connection to the VV. Cooling water to the BM is supplied by manifolds supported off the VV behind or to the side of

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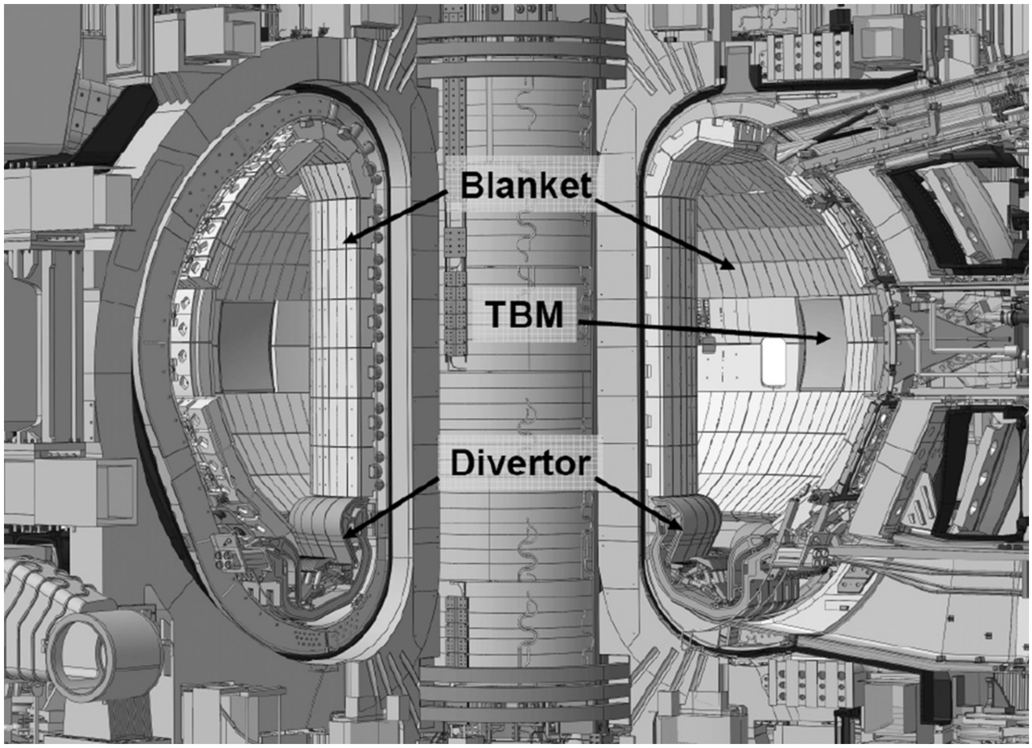

Fig. 1. ITER plasma facing components.

the SB. The cooling water is fed to and from the SB through a flow-separator-type connector and branch pipes. The branch pipes provide sufficient flexibility to cope with the differential thermal expansion of the VV and manifolds during operation and baking cycles. The BMs are segmented into 18 poloidal locations: rows 1–6 constitute the inboard region, rows 7–10 the upper region and rows 11–18 the outboard region (Fig. 2).

The inboard and upper modules (except BM10) are segmented toroidally into 18 equal modules, and the outboard modules (except BM14 and 15) into 36 modules. In the upper and equatorial port

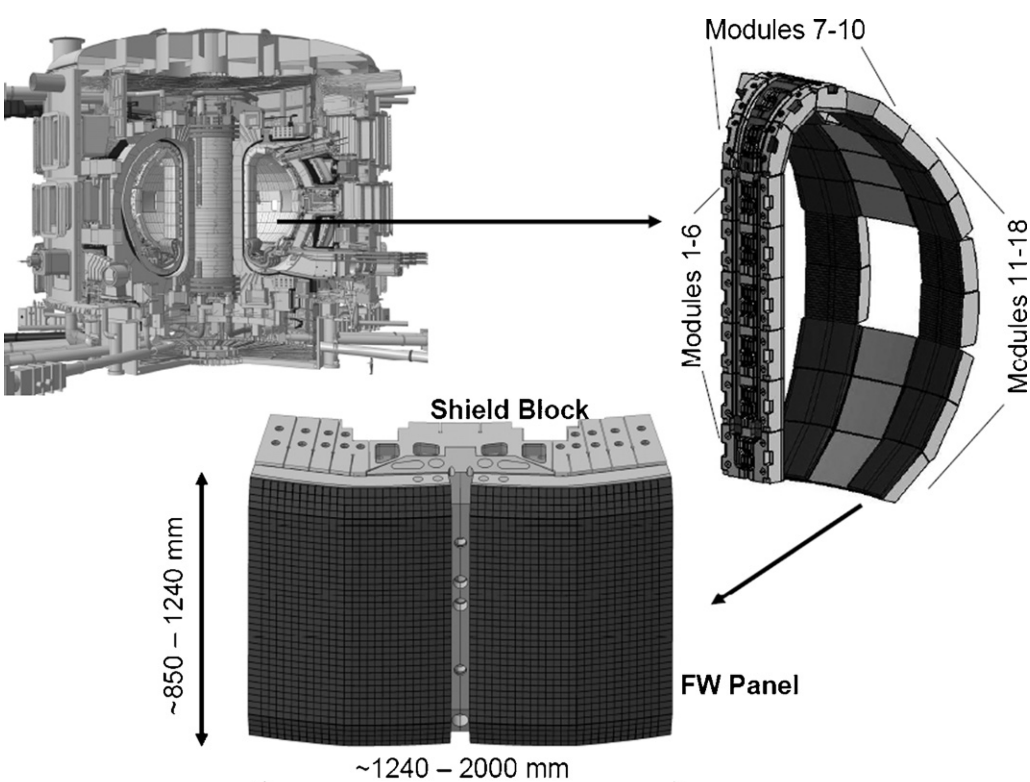

Fig. 2. Blanket module: overall configuration.

Table 1
Summary of main blanket design parameters.

Parameter	Value
Number of BMs	440
Typical module dimension	1.4 m × 1.0 m × 0.45 m
Max allowable mass of one module	4.5 t
Total mass on BMs	1530 t
Coolant type	Water
Inlet temperature (°C)	100
Inlet pressure at chimney bulk head (MPa)	3.0
Minimum FW panel flow velocity (m/s)	1.5
Minimum SB flow velocity (m/s)	0.8
Materials	
Heat sink of FW panels	CuCrZr-IG alloy
Armour	Beryllium (S-65C or equivalent)
Pipes and hydraulic connections	316L
Shield block	316L(N)-IG
Design heat flux on FW panels	1–2 and 5 MW/m^2 depending on location
Max neutron damage in Be/heat sink/steel pipe/SB	1.6/5.3/3.4/2.3 dpa

region (BM10, 14 and 15), the modules are located between ports and therefore segmented into 18 modules. In the neutral beam area, vessel sectors 2–4 have a custom segmentation for BM 14 and 15. In the outboard region, the nominal gaps between blanket modules are 20 mm both horizontally and vertically for installation and disassembly. To increase the nuclear shielding in the inboard region, the vertical gaps are reduced to 14 mm while the horizontal gaps are reduced to 10 mm. In the current configuration, the nominal thickness of the blanket modules is 450 mm for rows 1–17. The thickness of SB18, which is supported via a triangular support, is 400 mm.

The port-mounted blankets have identical requirements to those of the wall-mounted blankets. On the upper port, the port mounted blanket is customised to fit the 10 diagnostic port plugs (at ports #1, #2, #3, #8, #9, #10, #11, #14, #17, #18).

Six equatorial port-mounted blanket modules are customised to fit six diagnostic plugs (at ports #1, #3, #9, #10, #11, #12). All the diagnostic port mounted BMs are supported by diagnostic shield modules, which in turn are supported by port plugs. Each upper diagnostic port-mounted BM is split into two pieces and equatorial port BMs are split into four pieces. Splitting of the blanket shield simplifies the customisation process and reduces EM loads.

The BM components are designed for first installation and removal before D-T operations using either hands-on and/or using remote handling (RH) tools. All subsequent installation activities must be accomplished using only RH tools [1,2]. The BMs are designed such that in situ replacement of all the FW panels can be performed once over the 20-year operational lifetime of the ITER machine, with occasional replacement of a few individual FW panels being possible if required in the course of machine operation. There are no scheduled maintenance operations for the SB, though capability is built into the design to allow replacement of SBs in small numbers if required.

2.2. First wall panel

The FW panel is composed of an array of plasma-facing fingers (to reduce loads due to eddy currents) assembled into a stainless steel central support beam. The fingers can be manufactured and inspected separately and then assembled into the support beam. The finger cooling circuits are then joined to the central beam circuit using a bore welding process.

Plasma-facing fingers are designed for either “enhanced” (5 MW/m^2) or “normal” (1–2 MW/m^2) heat flux levels and with both variants employing the same interfaces and attachment features (Fig. 3). There are 234 normal FW panels (at poloidal rows

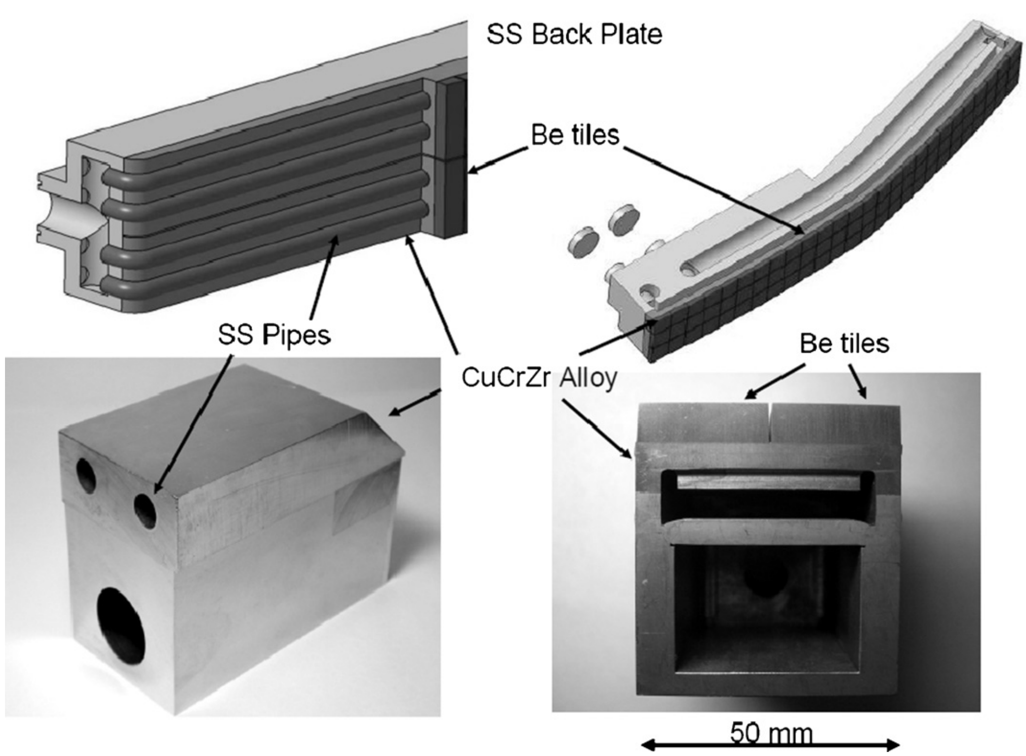

Fig. 3. Illustration of the basic FW panel structure and fingers. Left: normal heat flux fingers (concept with steel cooling pipes). Right: enhanced heat flux fingers (concept with rectangular channels).

1#, 2#, 6#, 10–13# and, partially, 18#) and 216 enhanced FW panels for a total of 440 wall-mounted BMs. Each finger comprises a series of beryllium (Be) armour tiles, a copper alloy (CuCrZr) water-cooled heat sink and stainless steel (SS) support structure. They are assembled on the central support beam with a toroidal orientation to equalise as much as possible the thermal load distribution.

Two finger types are considered necessary for the heat flux levels [3]:

(i) SS tube for FW panels with normal heat flux level, in which the SS tubes are embedded into a copper alloy (CuCrZr);
(ii) CuCrZr alloy rectangular or circular channels for FW panels with enhanced heat flux level.

The joining of Be armour 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. The FW panel central support beam is made of SS and provides the mechanical support for the fingers. A matrix of water cooling channels are drilled and plugged within the beam and connected to the fingers through ϕ 24 mm (ID) pipes. The FW surface of each panel is defined by the toroidal shaping of the individual fingers assembled into the two lateral sides of the central support beam. The sides are separated by a 60 mm wide slot running poloidally in the center of the panel. The recess is protected from the open plasma flux tubes by the shadowing effect of the FW shaping.

The panels are shaped so that the heat loads on the lateral sides of the panels are reduced down to acceptable levels.

2.3. Shield block

The main function of the SB is to provide nuclear shielding and supply the FW panel with cooling water. The steel/water/void ratio (86:10:4 and 85:12:3 for the inboard and outboard BMs, respectively) has been optimised with respect to neutron shielding. This ratio is achieved by optimising the number of poloidal cooling channels and their size within the SB. To further improve the nuclear shielding of the TF coils, an increase of the thickness of the inboard blanket modules is presently being considered. A number of deep slits are machined into the SB to reduce the impact of the electromagnetic (EM) loads on the structural components of the support system and VV.

The front face of the SB has a much higher nuclear heating than the rear side. As a consequence, cooling water hole diameters (ϕ12 mm) are minimised for cooling of the front part. They are fed in parallel with an average water velocity of 0.8 m/s. In the rear part, larger holes are drilled both for water distribution and cooling. Water headers are machined on the side of the module with 10 mm welded cover plates. Taking into account the very low nuclear heating at the rear side and the pulse mode of ITER, all the rear cooling channels are arranged above the large cut-out. This arrangement simplifies both the water circuit and the fabrication with a maximum local temperature of 250 °C. The basic fabrication method for a SB starts from either single or multiple-forged steel blocks and includes drilling of holes, welding the cover plates of the water headers, and final machining of the interfaces.

The SB is mechanically attached to the VV via four flexible supports and a system of keys. Electrical insulation coatings are applied to the mechanical attachments to prevent 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. They are also used to compensate the positioning of the SB on the VV wall by means of custom machining. An adjustment of up to ±10 mm in the axial direction and ±5 mm transversely are built into the design of the supports. The keys are fitted with bronze pads covered with low friction coating to allow

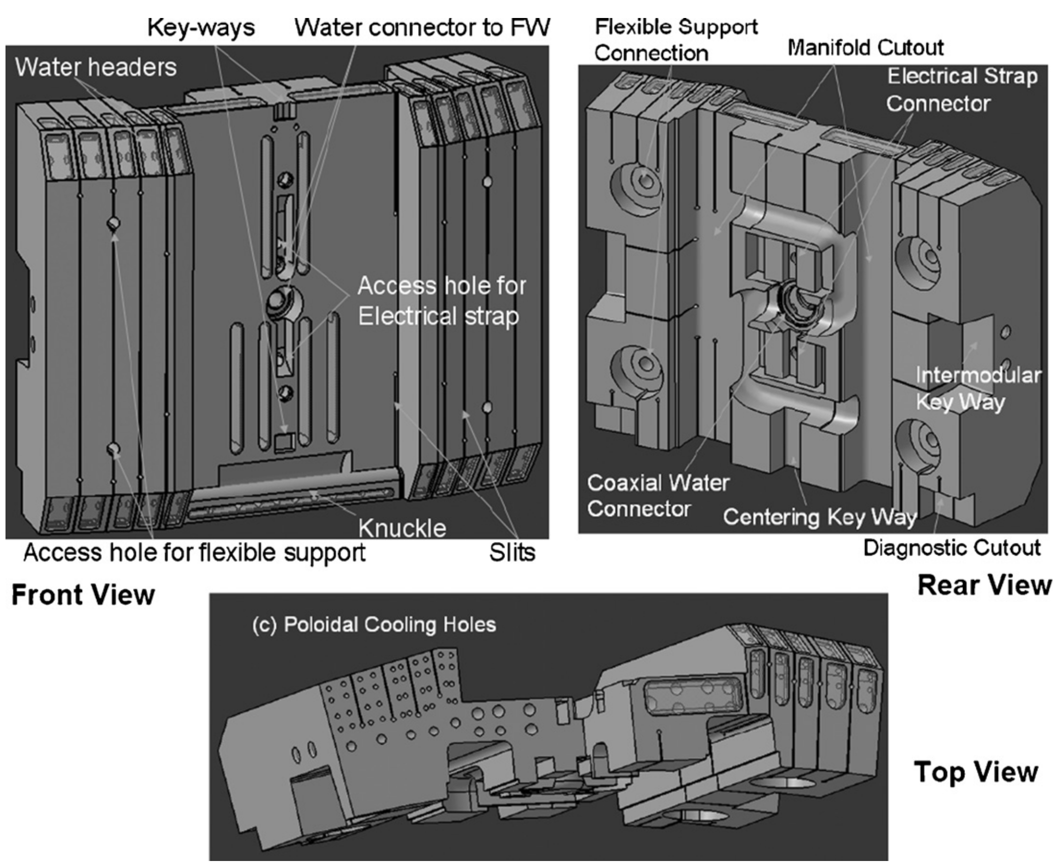

Fig. 4. Illustration of the SB structure.

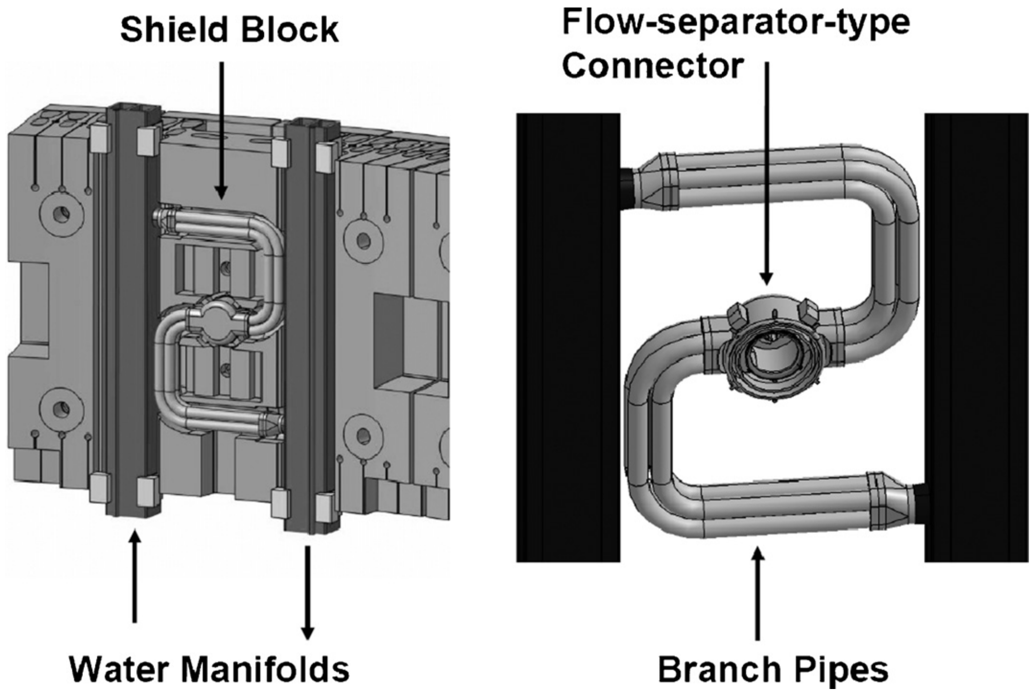

Fig. 5. Illustration of the flow-separator-type connector and branch pipe interfaces.

sliding of the module interfaces during relative thermal expansion and to allow custom machining to recover manufacturing tolerances of the VV and SB.

Each SB is electrically joined to the VV by electrical straps. They are formed and louvered from two sheets of CuCrZr alloy to achieve flexibility in all three directions. This electrical connection can handle up to 280 kA of electrical current (Fig. 4).

A flow-separator-type hydraulic water connector is utilised to provide the interface between the cooling circuit manifolds and the SB. It integrates the inlet and outlet of the cooling circuit with separator unit instead of coaxial tubes thus minimising the number of seal welds required (Fig. 5). This flow-separator-type connector is located in the centre of the SB where the thermal displacements are at their lowest and between the electrical straps where the protection against the halo currents is the highest. A single point connection also helps to minimise the eddy current. The flow-separator-type connector is designed to be initially installed and removed for maintenance requirements using RH tools. Connection of the cooling circuit to the hydraulic connector is by laser or TIG welding and a bore cutting tool is used for removal.

3. Divertor

3.1. General description

The ITER divertor is installed in the bottom part of the vacuum chamber. Its main function is to exhaust the scrape-off layer (SOL) power, which arrives at the divertor target plates by plasma conduction and convection or by radiation (photons and neutral particles) from the divertor plasma volume. It must perform this function whilst maintaining acceptable core plasma impurity (both due to helium ash produced by fusion reactions and impurities released as a consequence of the plasma-surface interaction). As the main interface component between the plasma and material surfaces, the divertor must tolerate high heat loads while at the same time providing neutron shielding for the VV and magnet coils in its vicinity. Its design must provide an engineering solution compatible with today's plasma physics expectations but, given the uncertainties inherent in such predictions, should also allow for flexibility to ensure a means for rapid replacement and refurbishment [4,5] (Table 2).

The ITER divertor consists of 54 cassette assemblies which are inserted radially through three lower level ports and moved toroidally before being locked into position. The cassette concept is fundamental for the maintenance strategy, as it allows installation to be limited to a few integrated components inside the vessel, thus minimising the maintenance operations in-vessel. The diver-

Table 2
Summary of main divertor design parameters.

Parameter	Value
Size	
Toroidal extent of a cassette	6.67°
Number of cassettes	54
Coolant type	Water
Inlet temperature (°C)	100
Inlet pressure at pipe stubs (MPa)	4.2
Max permissible pressure drop (MPa)	1.6
Max permissible total flow rate (kg/s)	1000
Materials	
Cassette body and supporting structures of PFCs	SS 316L(N)-IG and XM-19
Heat sink of PFUs	CuCrZr-IG alloy
Armour	CFC, pure sintered and deformed W
Pipes	316L
Pins of multilink attachments	C63200 Ni-Al bronze
Inner and outer locking system: nose and knuckle	C63200 Ni-Al bronze
Hard cover plates of divertor rails	Steel 660
Shear pins of hard cover plates	C63200 Ni–Al bronze
Design load for target in strike point region	$\sim$10 MW/m^2
Design load of target in baffle region	$\sim$5 MW/m^2
Min critical heat flux margin	1.4
Thickness of carbon fibre composite (CFC) at end of life	>4 mm
Plasma-facing surface area	$\sim$190 m^2

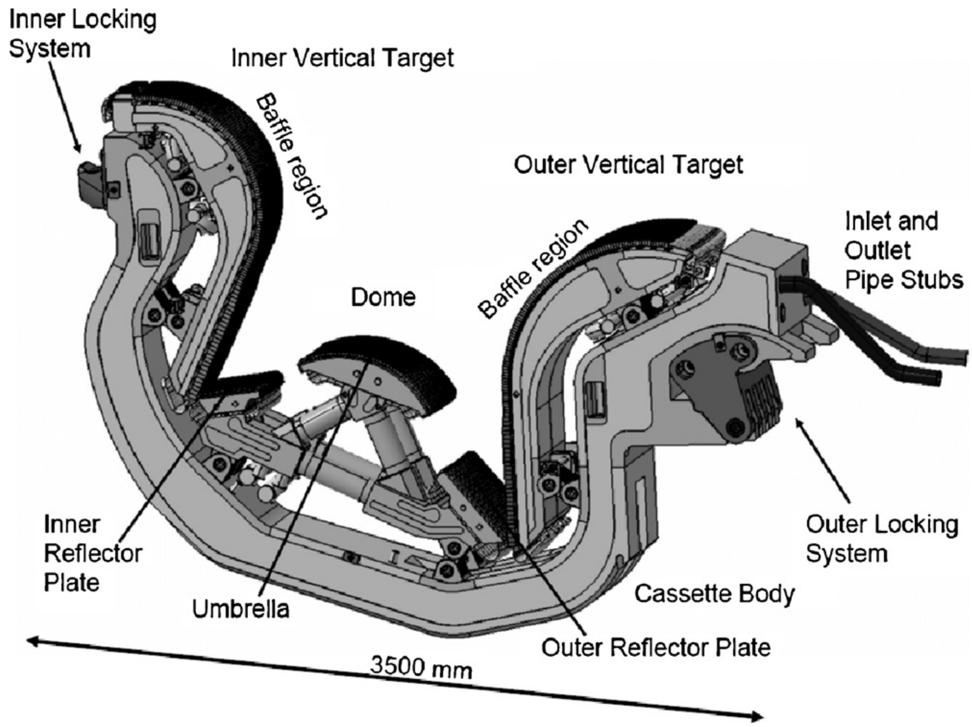

Fig. 6. Divertor cassette assembly.

tor is designed for relatively frequent (say three times during the nominal life) fully remote assembly and disassembly [6].

Each cassette assembly includes a cassette body (CB) that supports three separate components: the inner and outer vertical targets (VTs), and the dome (DO) (Fig. 6).

- The inner and outer VTs a constitute the highest heat flux components—their lower regions interact directly with the SOL plasma and the upper (curved) regions, or baffles, mostly with charge exchange (CX) neutral particles;
- The DO is located in the private flux region (the space below the separatrix which has no field line connection to the main plasma) and consists of:
  - (i) The umbrella, located below the X-point protects the CB and is exposed mainly to radiation and CX neutrals. It plays the important role of baffling neutral particles in the sub-divertor region, preventing excessive backflow to the main plasma;
  - (ii) The inner and outer neutral particle reflector plates that stop the plasma strike points from falling down below the divertor in case of lose of control, or if confinement transitions make the strike points displace.
- The CB that is reusable to minimise activated waste. It routes the coolant and provides both neutron shielding and a mechanical support for different possible PFC arrangements;
- Inner and outer locking systems, which are integrated into the CB to provide locking and alignment of the divertor cassettes against the VV through the hard cover plates on the inner and outer divertor rails;
- Inlet and outlet cooling pipe stubs, located in the outboard region of the CB and which interface to the radial cooling pipes; the latter are connected to the three cassettes in each 20° VV sector. These are cut and orbitally re-welded from outside the pipe.

The total design coolant water flow rate through the divertor is 934 kg/s. The inlet temperature and pressure is 100 °C, and 4.2 MPa, respectively. The maximum in-vessel pressure drop is estimated to be less than 1.6 MPa. The main scheme of the driver cooling system is to maintain an adequate margin to the critical heat flux (>1.4) for each component at the design heat flux [7,8].

During normal operation a steady state design heat flux of 5–10 MW/m^2 is assumed onto the bottom segment of the VT. However, the capability to remove up to 20 MW/m^2 during transient events (“slow transient” phase) of 10 s must also be provided. The upper segment of the VT is designed to handle steady state heat fluxes of up to 5 MW/m^2 to allow for configuration flexibility (for example during operation with raised strike points but at lower input power) and the high plasma radiation densities in the X-point vicinity which can occur when full performance plasmas operate in partially detached conditions. The DO is designed for to 5 MW/m^2 in steady state condition, plus the capability to remove 10 MW/m^2 for duration of up to 2 s.

3.2. Selection of plasma-facing materials

The PFCs consist of a plasma-facing material, the armour, which is made of either carbon fibre reinforced carbon composite (CFC) or tungsten (W). The armour is joined onto an actively cooled substrate, the heat sink, made of precipitation hardened copper based alloy CuCrZr.

The decision to begin with CFC armour on the lower part of the VTs in the reference design is motivated by a long history of experience in present and past tokamaks, the proven range of compatibility of carbon with a number of plasma conditions and the superior tolerance of CFC to transient load conditions (carbon does not melt). All will be important in the early phases of ITER operation, in which low density start-up with significant additional heating is expected to be required and during which the frequency of disruptions is likely to be high. The use of CFC also promises to facilitate the development of techniques for ELM control which must be performed before the active phase begins. In contrast, macroscopic cracking under repeated transient loads, melt layer loss and edge melting, possibly leading to surface deformation, make potential damage much more serious for W. Even if the onset for material damage under transient heat loads of CFC is similar to that of W,

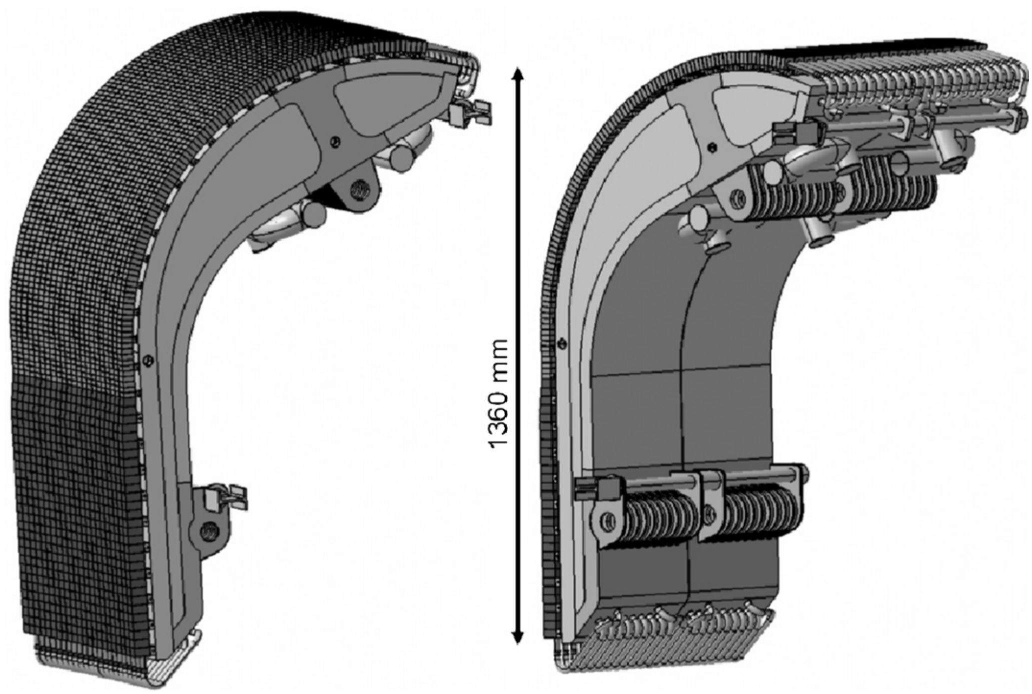

Fig. 7. 3D views of outer vertical targets consisting of two similar components.

vapour shielding effects at high power fluxes tend to enhance W erosion compared with C [9,10].

The power handling parts of the upper region of the VTs and the DO will be manufactured in W because of its high threshold for physical sputtering, low tritium retention, high melting temperature and good thermal conductivity.

3.3. Vertical targets

Each cassette carries one inner and one outer VT. In order to reduce the EM loads, each VT is split into two similar and independent components. Each consists of a steel supporting structure onto which the plasma-facing units (PFUs) are mounted. A PFU is a single poloidal element, which directly faces the plasma.

Each outer VT (Fig. 7) has 22 PFUs (11 per component) and two steel supporting structures (one per component). At the inner target, each VT has 16 PFUs (8 per component) and two steel supporting structures (one per component). The PFU geometry is based on the so-called “monoblock” concept, consisting of armour tiles with a drilled hole. A cooling tube made of CuCrZr copper-based alloy (12/15 mm ID/OD) is inserted into these holes and is intimately joined to the tiles. Each PFU has a CFC monoblock segment in the lower part and a curved W monoblock segment in the upper part. The reference armour thicknesses at the start-of-life are 15 mm for CFC and 8 mm for W above the heat sink tube.

To reduce the joint interface stress, a pure copper interlayer (thickness range: 0.5–1.5 mm) is envisaged between the CFC or W and the CuCrZr tube. The cooling pipe is made of CuCrZr and is joined to a 316L steel pipe outside the plasma-facing region. A twisted tape is inserted into the straight part of the cooling tube to increase the critical heat flux limit of the water coolant. It has a twist ratio of 2. The “twist ratio” is the axial length, measured in inner diameters, required for a 180° turn of the tape so that a twist ratio of 2 means that the tape rotates by 180° in $2 \times 12 = 24$ mm.

The steel supporting structure of the VT is a box fabricated structure made of austenitic stainless steel 316L(N)-IG plates and austenitic steel XM-19 forged pieces.

3.4. Cassette body

The CB supports the PFCs and is designed to withstand the EM forces, provide neutron shielding of the VV and coils and incorporates internal coolant channels which cool the CB and act as manifolds for the PFC coolant. The design of all CBs is identical and independent of location in the tokamak, with the sole exception of the cassette accommodating the lower vertical neutron camera. The CB is a welded component made of forged austenitic stainless steel XM-19 and 316L(N)-IG plates. Each has dimensions of 3.3 m length, ~2.0 m high, 0.4–0.8 m wide and a mass of 5.15 tons (the mass of whole divertor cassette assembly, namely CB + PFCs, is 8.6 tons). The minimum radial thickness of the CB is 240 mm and includes a 40 mm steel front plate, a 60 mm steel back plate and 140 mm of water. The front and back plates are welded to side and internal plates (both with 40 mm thickness).

3.5. Manufacturing technologies

The development of suitable technologies for the manufacturing of the divertor PFCs has been one of the most technically challenging R&D efforts to be performed within the ITER project [11]. More recently, the performances of the selected manufacturing technologies have been experimentally demonstrated via the successful manufacturing and testing of medium-size “Qualification Prototypes” (QPs). Indicatively, these QPs are around 400 mm long and consist of three high heat flux units mounted onto an actively cooled fabricated supporting structure. They include all the main features of the corresponding ITER design. Each QP had to meet all the prescribed acceptance criteria and withstand the high heat flux qualification tests (Fig. 8).

3.6. Divertor cassette integration into the vacuum vessel

The divertor cassette faces to the blanket module #01 (inboard) and #18 (outboard) with nominal clearances of 28 mm and 85 mm, respectively.

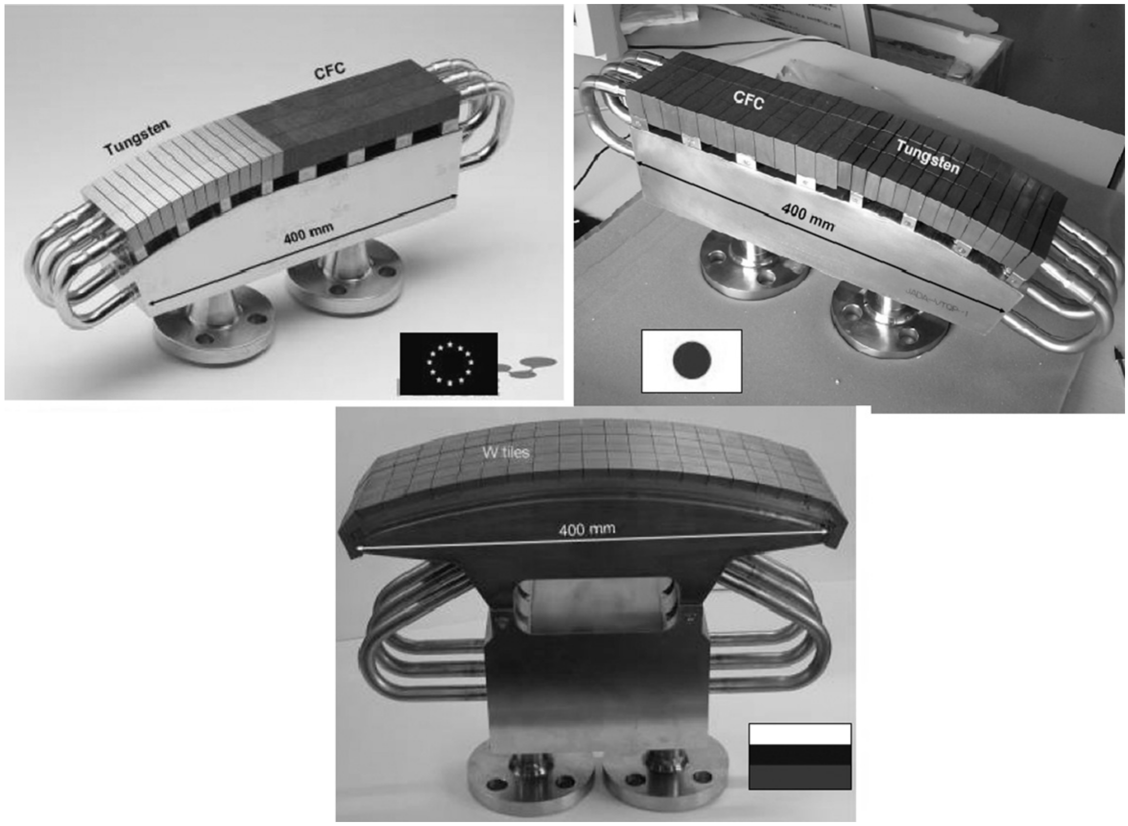

Fig. 8. Vertical target (top) and dome (bottom) medium-scale prototypes.

Each cassette occupies 6.67° in the toroidal direction, defined by the maximum size of cassette that can be handled via the RH ports whilst retaining an integer number of cassettes per sector. Of the nine ITER lower VV ports, 3 equally spaced ports are allocated to divertor remote maintenance. These ports are 2175 mm high and 1385 mm wide (723 mm width at the bottom of the trapezoidal port cross section), which is large enough to allow the cassettes to pass through during the installation. The divertor cassette is fastened

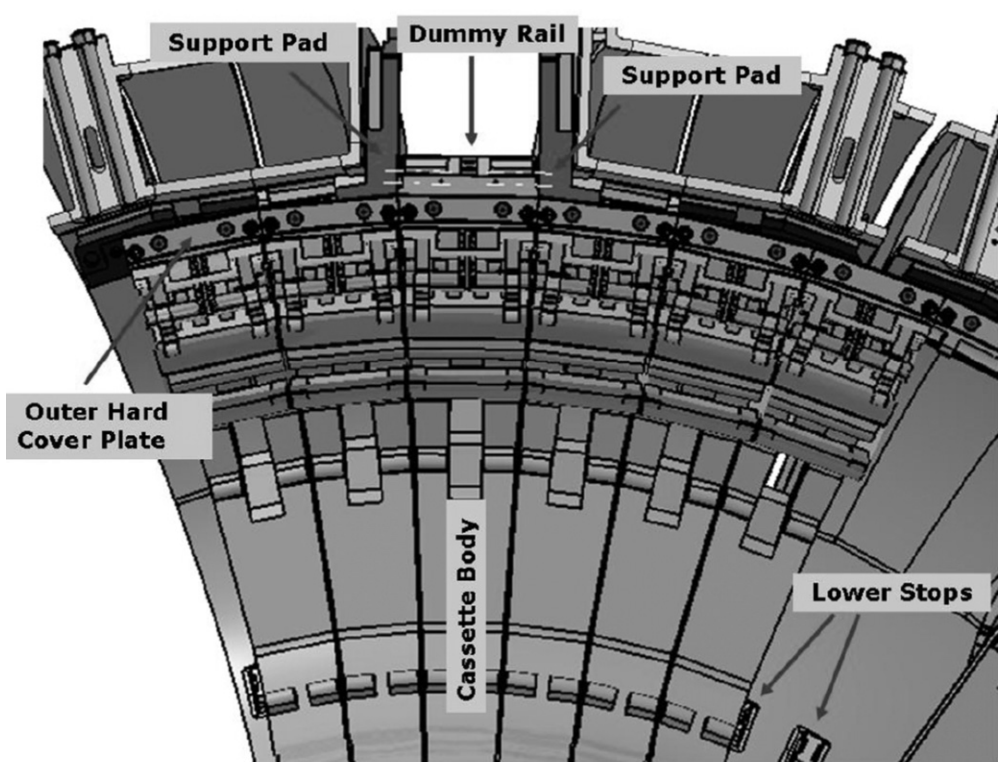

Fig. 9. View from top of the divertor cassette installed onto the VV rails.

to two concentric inner and outer toroidal rails welded to the VV via the so-called "hard cover plates". The lower ports include support pads for fixation of the "dummy rail" which ensures toroidal continuity of the outer rail in front of the lower ports (Fig. 9).

The divertor cassette is attached to the toroidal support rails integrated into the VV. The inboard locking system allows rotation in all directions whilst the outboard system permits rotation around the toroidal axis only. Combined toroidal and poloidal translation is blocked by the supports. The cassettes are installed into the VV with nominal gap of 20 mm between adjacent cassettes to accommodate optical diagnostics and to account for manufacture and installation tolerances.

After a comprehensive 3D survey of the lower part of the VV, custom machining of the hard cover plates on the rails will be performed. This will ensure an accurate positioning of the hard cover plates whilst recovering the expected manufacturing and assembly tolerances of the VV. The vertical step between adjacent hard cover plates of the toroidal rails will be limited to ± 0.5 mm.

The design relies on the CB acting as a spring to hold the cassette in place, sprung between supports adjacent to the inner and outer toroidal rails [12]. Both inner and outer locking parts, so-called, nose and knuckle, are machined out of Ni–Al-bronze. The nose shape ensures the mechanical lock to avoid disengagement under dynamic loads. Fig. 10 illustrates the locking process of a divertor cassette. The divertor cassette is first carried to the approximate locking position (the inboard attachment being initially aligned with the inboard socket centre within ± 5 mm toroidally, when the nose is supported at the inner hard cover plate by a physical contact). At this stage the nose is located loosely on the inner hard cover plate. The cylindrical end of the knuckle is inserted into the outer hard cover plate. The jack is then inserted and load applied to the link. As the load is applied the inboard attachment engages into the inner hard cover plate and the divertor cassette is guided to the correct final position at the same time as being compressed radially through the application of a radial preload ($\sim$10 tons, 5 mm displacement) by a jacking system. This ensures that the divertor cassette remains in contact with the supports under all loading conditions.

Lower stops, an integral part of the VV, are located between the underside of the divertor cassettes and the inner surface of VV to reduce the toroidal displacement of a divertor cassette during dynamic loads. This being the point of maximum displacement, it is also the point where the minimum force is required to cancel the torque. During installation, once the divertor cassette has reached its toroidal position, it is pushed inward to engage the inner attachment to the VV. In this way the lower stop also engages into a slot, which is obtained in the divertor cassette.

4. Test blanket modules

The test blanket modules (TBMs) are mock-ups of DEMO breeding blankets. They are installed in three ITER equatorial ports (Nos. 2, 16 and 18) devoted to TBM testing, each of them allocating two TBMs, inserted inside a steel frame. The TBMs and the related TBM programme is described elsewhere [13–15]. Here, only a description of the TBM frame and dummy TBM will be given.

The TBM frame is a water-cooled 316L(N)-IG steel port structure, which is based on the same materials and manufacturing process as those used for the blanket SB. It has a radial length of 3250 mm, a poloidal height of 2160 mm and a toroidal width of 1710 mm. It can accommodate up to two TBMs, having a plasma-facing surface of 484 mm (toroidal) $\times$ 1660 mm (poloidal). Since such surface is relatively small compared to the total FW area, it is expected that the TBMs do not need a plasma-facing beryllium-armour as required for the BM FW panels. Behind each TBM, there is a shield block that has a function to protect the cryostat/magnets region. Both components form the so-called TBM-set.

The water-coolant circuit of the frame is connected in parallel with the water-coolant circuit of the shield blankets and with the shield part of each TBM-set.

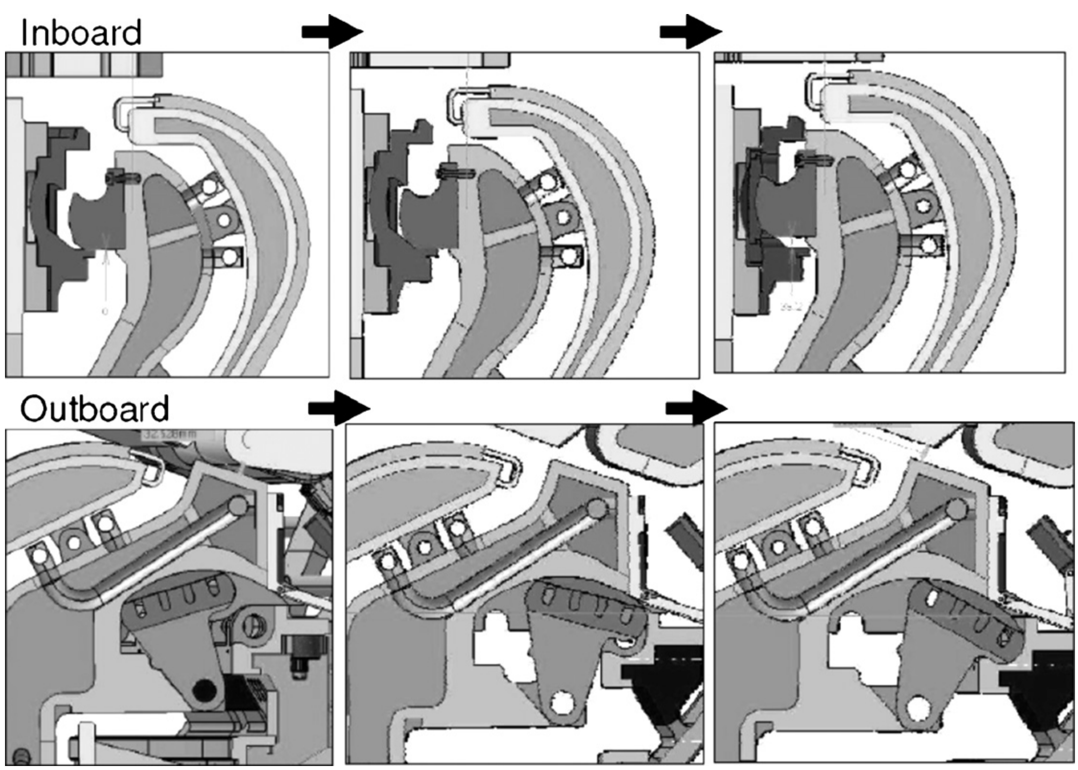

Fig. 10. Locking of the divertor cassette onto the divertor rails.

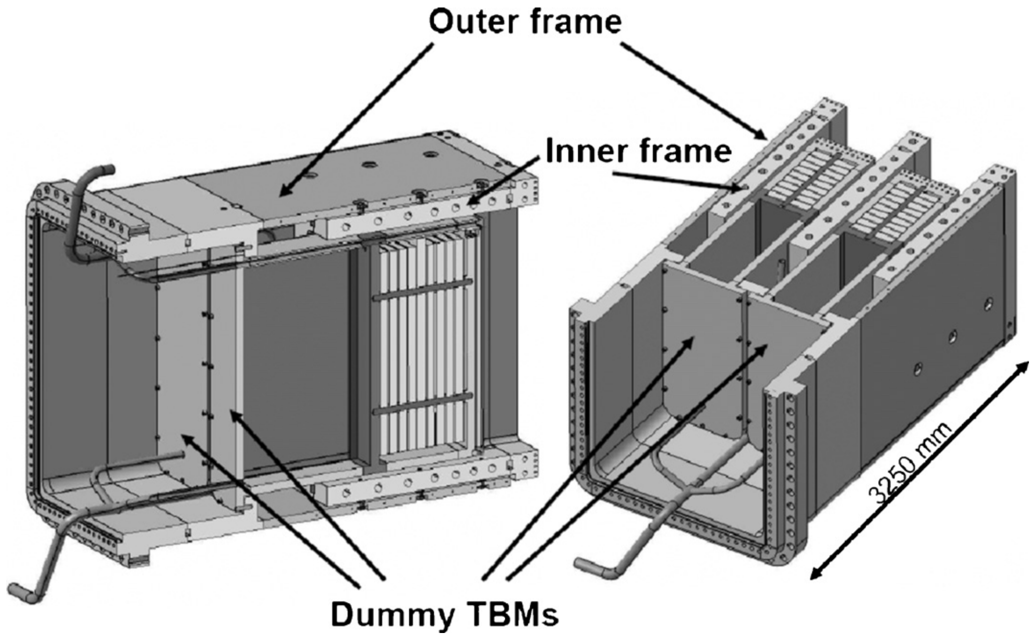

Fig. 11. Illustration of the TBM frame: poloidal (left) and toroidal (right) cut-out.

The TBM frame comprises two parts: a structural part, called the "outer frame", and a shield part, the "inner frame". These are steel fabricated structures, where the cooling channels are obtained by drilling. The outer frame has a design similar to that of the ITER generic equatorial port plug frame and is obtained by welding 60 mm thick plates. It is welded to the equatorial port closure plate. The inner frame is bolted inside the outer frame and has a more massive structure with 155 mm wall thickness. It has the function to neutronically and thermally insulate the two TBMs between themselves and from the surrounding environment.

A TBM-set can be replaced by a dummy TBM in the form of a steel box structure which, like the outer frame, is also obtained by welding 60 mm thick plates. The box contains an array of 60 mm thick plates, which, together with the water coolant, ensures the shielding function. Each dummy TBM can be mounted and dismounted from the frame via a set of M20 bolts located on the rear side. This operation is performed in the hot cell, once the whole TBM port plug is removed through the port. The dummy TBM and the TBM frame are cooled in parallel with the coolant connections located inside the equatorial ports (Fig. 11).

5. Conclusions

This paper has described the present status of the design of the ITER blanket, divertor and TBM frames.

For each system a viable design solution exists, which is compatible with the ITER requirements and existing manufacturing technologies.

The blanket design was subjected to a conceptual design review in February 2010, prior to entering into its present detailed design phase, which will culminate in a preliminary design review in mid 2011 and a final design review in mid 2012. The procurement is planned to start late-2012 and should last till 2019. In parallel to that, the construction technologies will undergo a formal qualification process by the manufacturing and testing of full-scale semi-prototypes.

The final design review of the divertor was successfully completed in December 2008 and the procurement phase has already started in June 2009 with the fabrication of the outer VT and the DO.

The design of the TBM frame is at a pre-conceptual design stage and design supporting analysis is planned late in 2010 and includes electromagnetic, neutronic, thermal and mechanical calculations, thus allowing to finalize the frame and dummy TBM conceptual design by early-2011. The procurement of the TBM frame and dummy TBM is planned to start in 2014.

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