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ITER vacuum vessel, in-vessel components and plasma facing materials

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ABSTRACT

The vacuum vessel (VV) design is being developed in more detail considering manufacturing and assembly methods, and cost. Incorporating manufacturing studies being performed in cooperation with ITER Parties, the regular VV sector design has been nearly finalized. Design of the neutral beam (NB) ports including duct liners has been developed.

Design of the in-wall shielding has been developed in more detail considering the supporting structure and the assembly method. Additional ferromagnetic inserts to be installed in the outboard midplane region will minimize the maximum ripple and the toroidal field flux line fluctuation.

Detailed studies were carried out on the ITER vacuum vessel to define appropriate codes and standards in the context of ITER licensing in France.

The blanket module design has progressed in cooperation with participant teams. Fabrication of mock-ups for qualification testing is under way and the tests will be performed in 2007–2008.

The divertor activities have progressed with the aim of launching the procurement according to the ITER project schedule.

1. Introduction

Design improvements and additional R&D of ITER vessels and in-vessel components have progressed to fix interfaces and define technical specifications. The procurement specifications for long lead-time items are under preparation. Review of the design requirements and the current design is now going on before start of the procurement of ITER components.

2. Vacuum vessel

The main components that make up the vacuum vessel (VV) are the main vessel, the port structures and the VV supporting system. The VV is a torus-shaped double wall structure with shielding and cooling water between the shells [1,2]. The basic vessel design is an all-welded structure. Only the inner shell serves as the first confinement barrier. The VV components need to be designed and manufactured consistent with an accepted code or standard. The RCC-MR code is expected to be used for the ITER VV. The VV is divided into nine toroidal sectors joined by field welding using splice plates at the central vertical plane of alternate ports. The final welding is performed in parallel at three locations between 120° sectors. The VV has upper, equatorial, and lower port structures (including local penetrations located mainly at the lower level of the machine) (see Fig. 1). At the upper level, there are 18 ports of a similar design. At the equatorial level, there are 14 regular equatorial ports and three ports for the neutral beam injection (NB ports). There is one “blind” port. At the lower level, there are five ports for divertor cassette replacement and/or diagnostics (the divertor RH/diagnostic ports), and four ports for vacuum pumping (the cryopump ports). Between these ports, there are local penetrations for the divertor piping, the in-vessel viewing and glow discharge systems.

Detailed design is progressing on the main vessel and ports for the procurement specification document to start the call for tender in 2008. The regular VV sector design has been nearly finalized.

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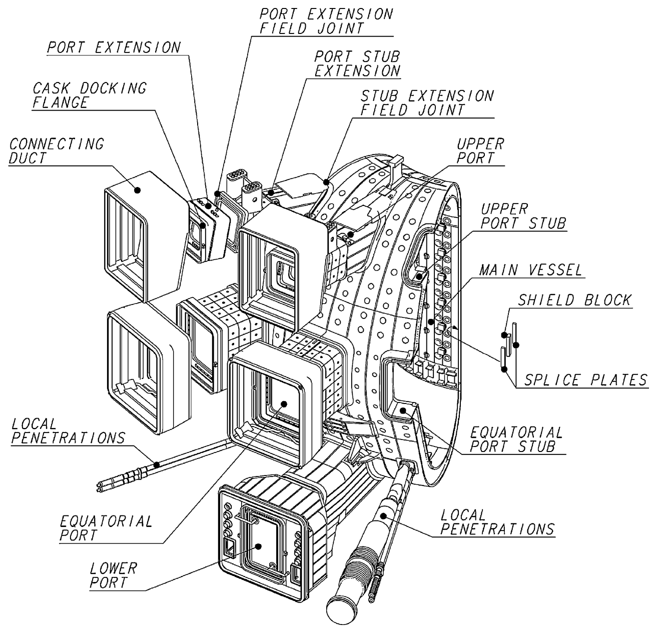

Fig. 1. ITER vacuum vessel and ports.

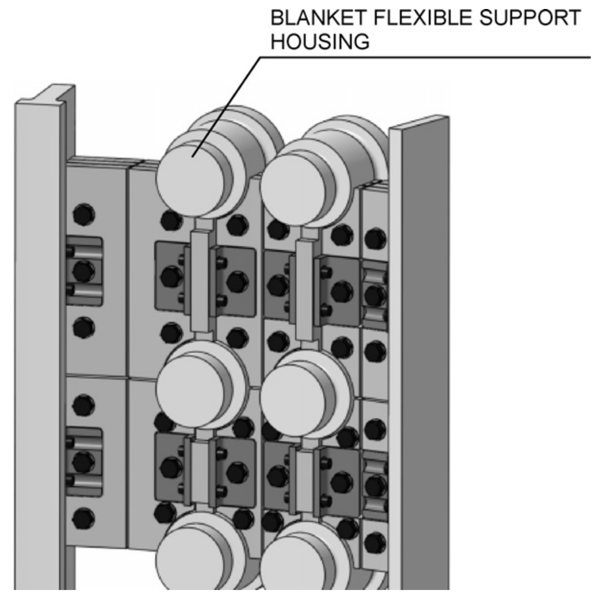

Fig. 3. Supporting structure of the in-wall shielding.

Design of the NB ports including duct liners under heat loads of the neutral beams has been developed.

2.1. In-wall shielding

The space between the double walls will be filled with shield structures (see Fig. 2) mainly made of an austenitic stainless steel containing 1–2% weight boron to improve the neutron shielding efficiency. The shielding structures occupy 55–60% of the in-wall space. A ferritic stainless steel (SS 430) is used as the shielding material in the shadow of the TF coils in the outboard area to reduce the toroidal field ripple. These plates fill up to 60% of the volume between the shells. This steel has a high saturated magnetization at about 1.5 T. The shield blocks are fixed by bolts to the ribs to withstand the mechanical forces (see Fig. 3) [2,3]. The gaps between the shield blocks and between the blocks and the ribs are minimized to avoid excessive neutron streaming.

The ferromagnetic insert was previously not included in the outboard midplane region between equatorial ports due to irregularity caused by the tangential ports for neutral beam injection. The absence of the ferromagnetic inserts in the midplane causes a relatively large ripple (~1%) in a limited region of the plasma and the toroidal field flux lines fluctuate ~10 mm due to the large ripple in the FW region. It is difficult to achieve the same configuration when including the additional ferromagnetic insert in each toroidal location due to the constraint of the supporting structure of the shield blocks. Therefore, the same volume of ferromagnetic insert is added in every sector but with different shapes between regular sectors and NBI sectors (see Fig. 4) [4]. The volume of the additional ferromagnetic insert is adjusted to make the magnetic configuration toroidally cyclic as much as possible to minimize the effect of lower mode error fields.

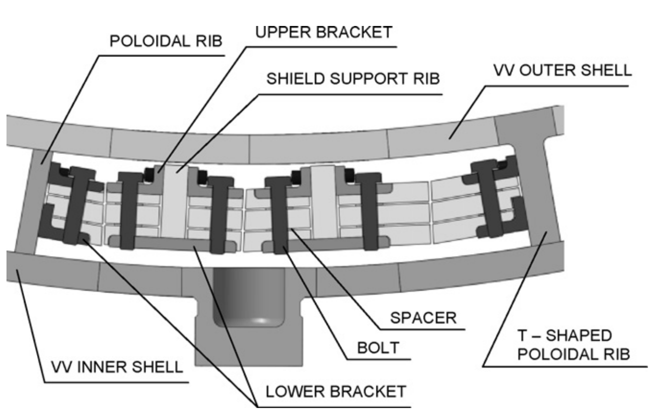

Fig. 2. Layout of the in-wall shielding.

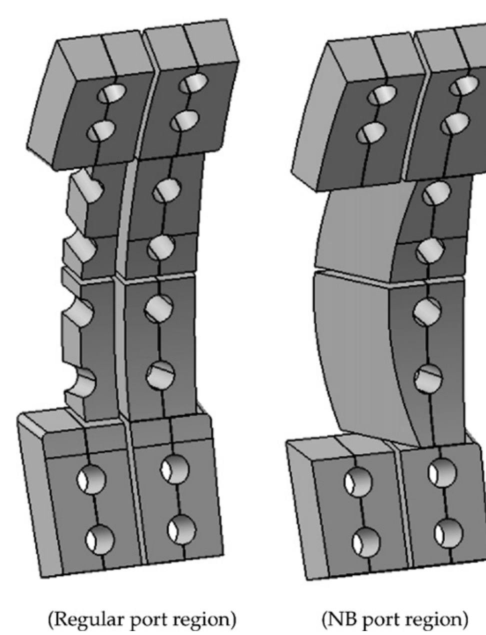

Fig. 4. Additional ferromagnetic inserts between equatorial ports.

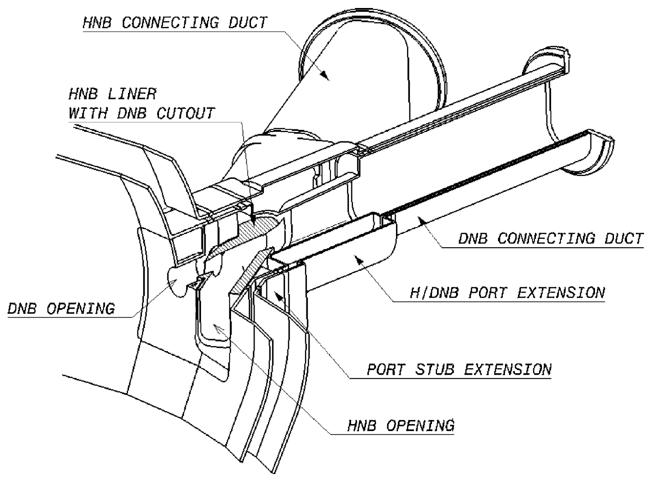

Fig. 5. Isometric view of the H/DNB port.

2.2. NB ports

There are three tangential ports (HNB ports) for the heating neutral beam injection at the equatorial level. The diagnostic neutral beam port (DNB port) is at the same location with one of the three HNB ports (see Fig. 5). There is an internal duct liner inside each NB port. The liner is equipped with the beam-facing panels to withstand the NB power affecting its surface (see Fig. 6). Additional shielding components (like shield inserts) are also used in these ports. Further design of the NB ports and duct liners is progressing according to requirements from the NB injection and the remote maintenance.

2.3. VV supports

The VV is vertically supported at its nine lower ports by sliding supports resting directly on the ring pedestal (see Fig. 7). These sliding supports are radially restrained against fast displacements taking place during seismic events or plasma disruptions/VDEs. They are however radially free to move during thermal expansion. The VV is also restrained in the upward direction through a set of vertical links, located between the pedestal and the lower port. Additionally, the VV is restrained toroidally for the position centring. PTFE is the reference sliding material and polyamide is also a candidate material, the friction factor being lower than 0.05 in both cases.

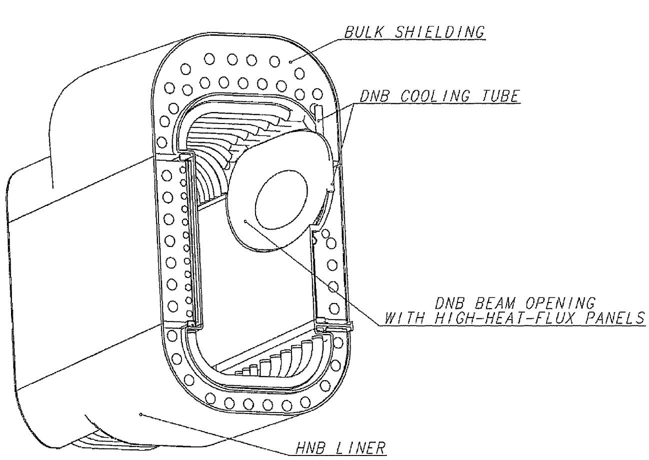

Fig. 6. HNB liner with DNB opening.

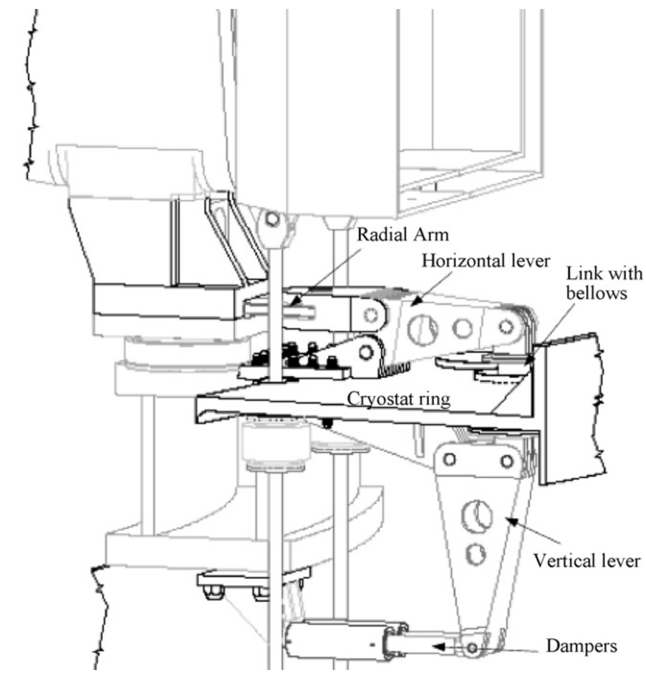

Fig. 7. The VV supporting system.

The steady state vertical load is 10 MN per support due to the gravity force, and the maximum transient vertical load is ~25 MN per support which gives an average compressive pressure ~50 MPa. R&D using a mock-up will start this year to demonstrate the feasibility of this supporting system under the required load conditions.

2.4. Code and standards

Detailed studies were carried out on the ITER vacuum vessel to define appropriate codes and standards in the context of ITER licensing in France. A set of draft documents regarding the ITER vacuum vessel structural code were prepared including an RCC-MR addendum for the ITER VV with justified exceptions or modifications. The main deviation from the base code is the extensive use of UT in lieu of radiography for the volumetric examination of all one-side access welds of the outer shell and field joint. In 2007 the content of the addendum prepared for the ITER vacuum vessel is being incorporated into an updated version of RCC-MR. The VV components need to be designed and manufactured consistent with an accepted code or standard.

2.5. VV R&D

A full-scale partial poloidal-segment mock-up of the ITER vacuum vessel was fabricated to gain knowledge, experience and confidence on manufacturing of the VV sector which is consistent with the current design with keys and flexible support housings for blanket modules. One of the most important objectives of this R&D is to simulate welding deformations and to confirm the achievable tolerances required for the ITER vacuum sector fabrication. The mock-up includes a part of the inboard straight section (PS1) (see Fig. 8) and a part of the inboard upper most-curved section (PS2) with a frame simulating the stiffness of the remaining part. The blanket support housings are welded to the vessel inner shell with electron beam to minimize welding deformations and residual stresses. Welding shrinkages and distortions were measured in fabrication of PS1 and PS2 and weld joint connection between PS1 and PS2 (see Fig. 9). Finally, fixtures were dismantled from the

Fig. 8. Partial VV sector mock-up: PS2 (curved section) fabrication (EU).

mock-up PS2 (see Fig. 10) and the dimensions and positions were measured. It is confirmed that the mock-up was fabricated within the required tolerances [5].

Liquid dye penetrant testing is widely used as a surface NDT method according to the code requirements. One concern is whether it is compatible with ultra-high vacuum requirements when it is applied on the inner surface of the inner shell. Test results showed that the outgassing rate is acceptable under the conditions of (i) selection of "nuclear grade" dye penetrant with small amounts of impurities, (ii) adequate cleaning after testing and (iii) 200 °C baking. The test equipment is shown in Fig. 11 [6].

Fig. 9. Partial VV sector mock-up: welding joint connection between PS1 (straight section) and PS2 (inboard upper curved section) (EU).

Fig. 10. Partial VV sector mock-up after dismantling fixtures (EU).

3. Cryostat

The ITER cryostat is also a safety class component and works as part of the second radioactivity confinement barrier. The cryostat is connected with the vacuum vessel bellows through bellows. The original cryostat design comprised a flat lid and a heavy reinforced welded beam structure (see Fig. 12). The flat top lid supported the concrete bioshield roof. In a new proposed design, the bioshield

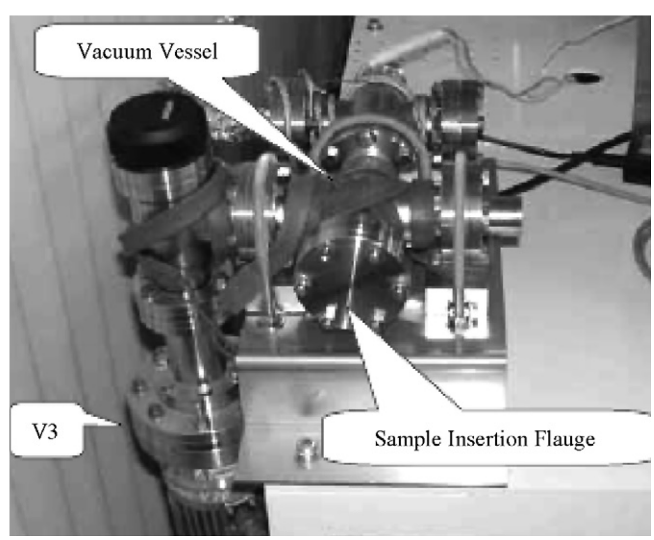

Fig. 11. Test equipment for outgassing measurement after the dye penetrant testing (JA).

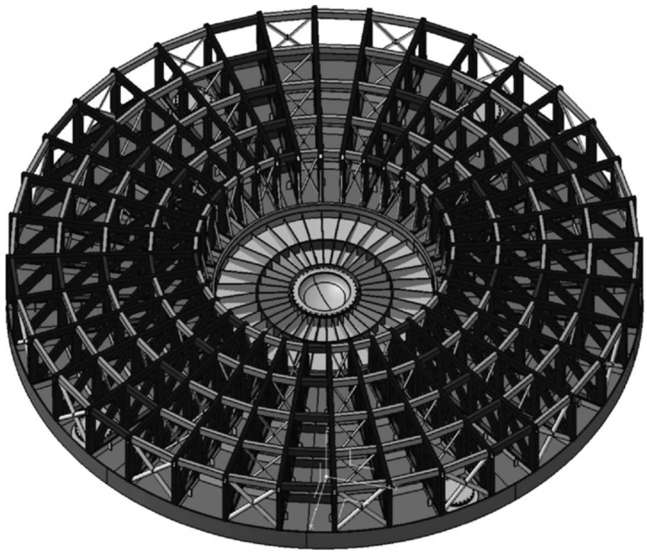

Fig. 12. Upper cryostat structure of original design.

roof is self-supported and keeps leak tightness to maintain the negative pressure boundary. The cryostat is also self-supported and required to withstand the pressure loads (vacuum and 0.2 MPa over-pressure due to off normal events) and gravity/seismic loads.

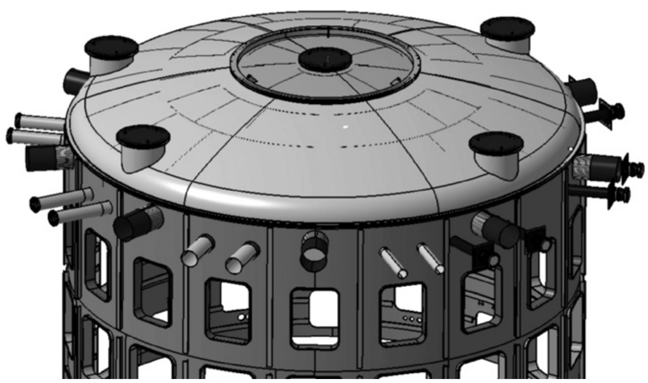

Fig. 13. Upper cryostat structure of proposed new design.

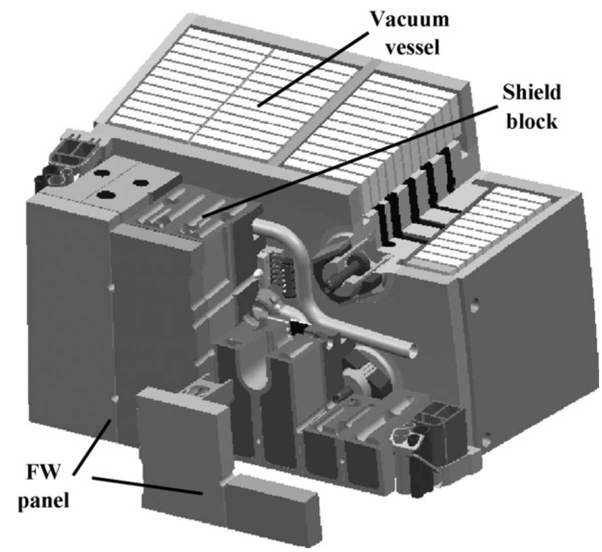

Fig. 15. Blanket module structure (outboard).

The top lid has a semi-hemispherical shape (see Figs. 13 and 14). In this new design, the cryostat cylinder shell becomes thinner and the assembly/welding work in the pit will become easier. Details of the assembly, NDT and leak testing are under development,

4. FW/blanket

As shown in Fig. 15, each blanket module consists of four FW panels and a shield block [1]. Each FW panel is fixed to a shield block with a central support leg (CSL). The FW panel is 81 mm thick, and completely slitted except the central part to reduce eddy currents during disruptions. Based on detailed thermal and structural analysis, the layout of slits and FW cooling pipes has been further optimized to avoid thermal "bowing" of each "finger" (a section of the FW panel between adjacent slits) to the toroidal direction. The toroidal cross section of each first wall panel is multi-facetted in the inboard region (see Fig. 16) and flat in the outboard region [7]. This configuration has been selected to avoid exposing leading edges and at the same time to avoid excessive concentration of halo current in one of the "fingers". A small modification has been made

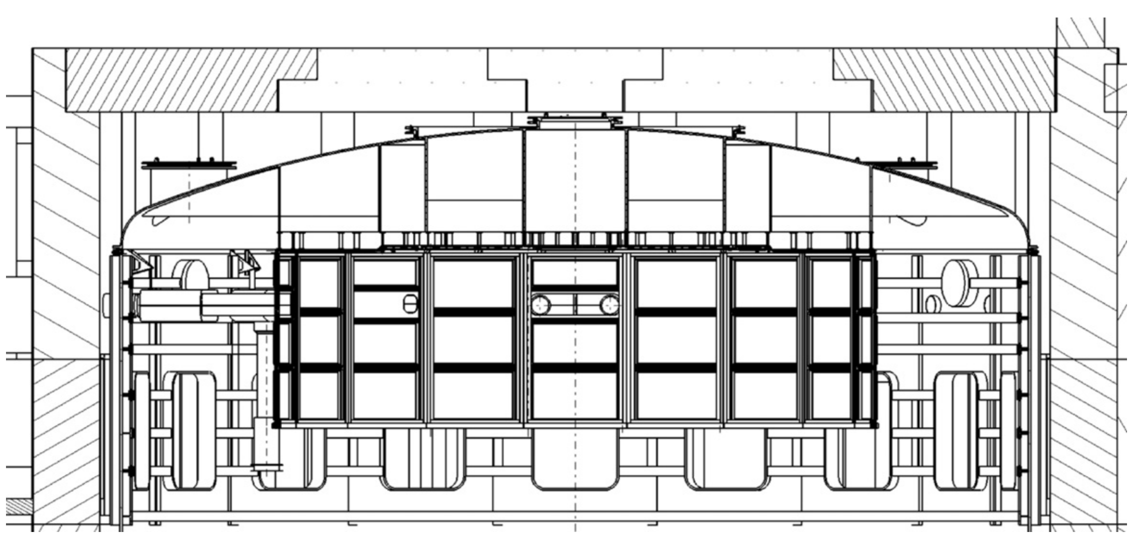

Fig. 14. Cross-section of the upper cryostat and self-supported top nuclear shielding.

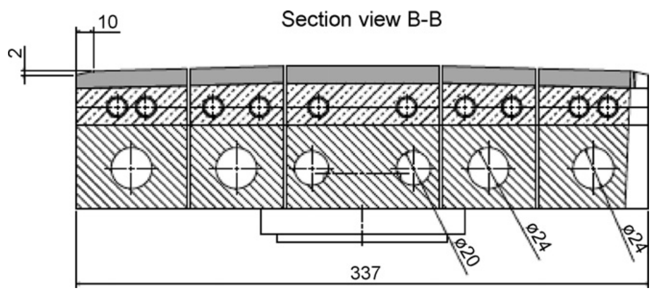

Fig. 16. Toroidal cross-section of the inboard FW panel [8,9].

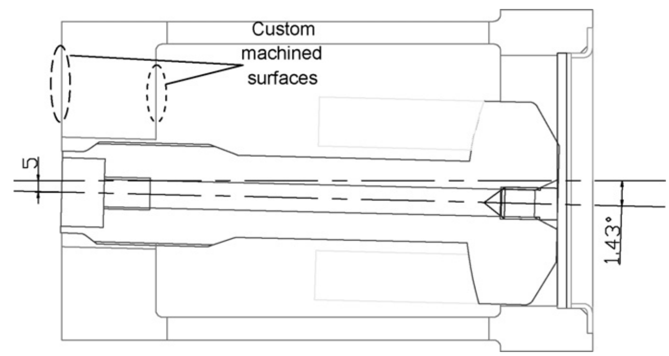

Fig. 18. Blanket flexible support and support housing (new updated design).

in the FW panel edge structure to simplify the fabrication method (see Fig. 17).

The procurement allocation of blanket modules among six ITER parties has been fixed and the blanket module design has progressed in cooperation with them. Fabrication of mock-ups for prequalification testing is under way [8] and the tests will be performed in 2007–2008. The test conditions will be basic heat cycles 0.625 MW/m^2 × 12,000 cycles + MARFE heat loads + NB shine-through heat loads. Development of beryllium materials for the ITER FW is progressing in China and Russia.

The blanket module is attached to the vacuum vessel with flexible supports made from titanium alloy Ti6Al4V [9]. The position and angle of the blanket module is adjusted by custom machining of the flexible supports. In the original design, threads contacting the vacuum vessel were to be custom machined. However, custom machining of the titanium alloy threads is time consuming and there is a concern on the fabrication and assembly tolerances in this method. In the new reference design, the surface of the flexible support contacting the blanket module is custom machined (see Fig. 18). Before final machining, the flexible support is pre-installed onto the vacuum vessel and dimensional measurement is performed to get accurate data for the final machining step. By using this method, the final fabrication and assembly tolerance can be minimized.

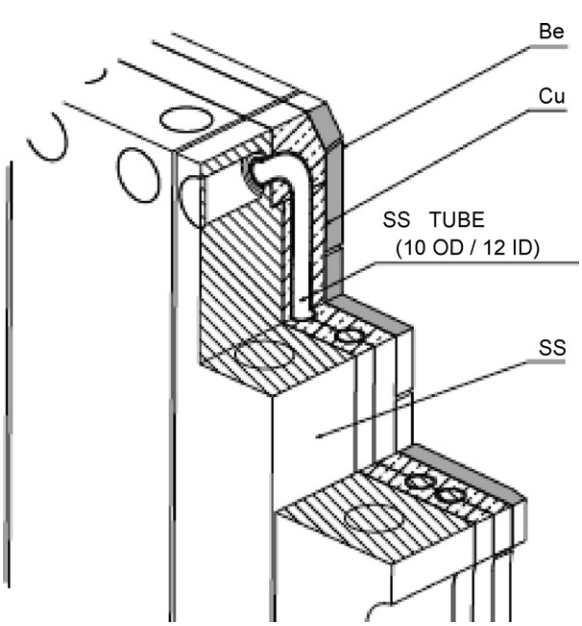

Fig. 17. Edge internal structure of FW panel [9].

The moveable limiters are to be installed in equatorial ports at two toroidal locations. A more detailed design effort is in progress considering the driving mechanism for positioning and maintenance. There are bellows, helical coolant pipes and gaps near the central support shaft of the moveable limiter system. They do not provide enough neutron shielding efficiency in these areas and additional shields are taken into account in the limiter system design update.

Design of the port plug BSM (blanket shield module) has progressed. To reduce the electromagnetic loads, intensive slitting is used in the poloidal direction. Upper port BSMs in the NB region are fixed with bolts due to the required maintainability inside the tokamak as shown in Fig. 19. The hydraulic connection which has been developed for the blanket modules can be used for the hydraulic connection of the upper port BSM in the NB port region.

5. Divertor

Divertor activities have progressed with the aim of launching the procurement during the second half of 2009 according to the ITER project schedule. An important milestone was the formal start of the divertor qualification phase with the EU, JA and RF participating participant (PTs). For the qualification, each PT must first demonstrate its technical capability to carry out the procurement with the required quality, and in an efficient and timely manner.

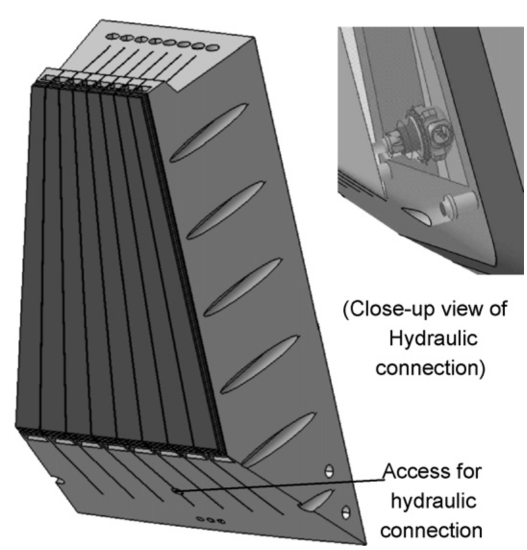

Fig. 19. Port plug BSM (blanket shield module).

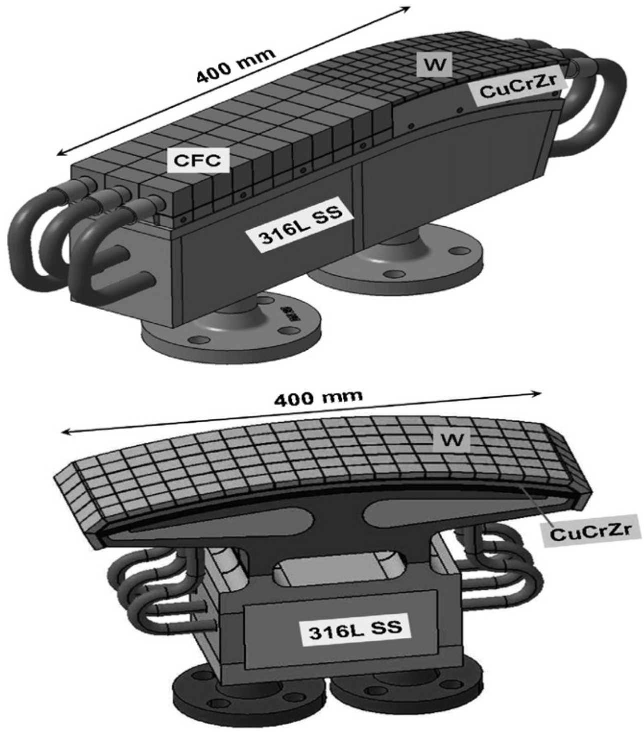

Fig. 20. Qualification prototypes for the vertical target (procured by EU and JA PTs – top) and dome (procured by RF PT – bottom).

This is achieved via the successful manufacturing and testing of medium-size "qualification prototypes" (QPs, Fig. 20).

A PT is considered qualified if:

- It delivers at least two QPs, which meet all the prescribed acceptance criteria. The delivered QPs can be manufactured via different technologies and/or different suppliers.
- At least one of the delivered prototypes withstands all the specified performance tests, which will be carried out in the high heat flux test facility located at the Efremov Institute in St. Petersburg, Russia.

In preparation to this qualification activity, the JA PT has manufactured a full-poloidal length outer vertical target prototype, which will be high heat flux tested late this year, the EU PT has manufactured a medium-scale vertical target prototype and tested up to 2000 cycles at 20 MW/m^2 and the RF PT has launched a large R&D work to qualify and optimize the various joining technologies, which are required for the dome.

Substantial design work has been carried out by the ITER organization to finalize the detailed design of the divertor and to investigate possible modifications to the standard divertor cassettes to accommodate the diagnostics.

The development of suitable acceptance criteria for the divertor plasma-facing components is progressing in EU with the manufacturing and high heat flux testing of more than 100 mock-ups with artificial and calibrated defects. This wide experimental database will form the basis for the determination of the maximum acceptable defect size for the vertical targets. With regard to the dome, the RF PT has just started a similar experimental activity, where a number of small-scale dome mock-ups with artificial defects will be also high heat flux tested.

The neutronic, thermal, mechanical and electromagnetic analyses of the standard divertor cassette, carried out by the EU and RF PTs are nearing completion and form a solid basis for the present reference design. In 2008, a similar activity is planned for the diagnostic cassettes.

In view of their responsibility on the final divertor assembly and integration, the EU PT has completed the manufacturing of a complete set of full-scale divertor components with dummy armour. They will be used to select and finalize the assembly procedures, to validate the hydraulic design of the divertor, and to identify the draining and drying methods.

6. Plasma-facing materials [10]

The plasma-facing materials (PFMs) for the initial operation of ITER consist of the following; beryllium for the first wall for its small impact on the plasma performance and high oxygen gettering; divertor targets covered by carbon-fibre-composite (CFC); tungsten for the dome and baffle (upper target) regions for its low yield of physical sputtering by neutral particles. CFC has high thermal shock resistance without any melting and CFC is widely used as a PFM in plasma experimental devices due to its compatibility with a wide range of plasma parameters. However, tritium retention control remains a key issue of CFC PFMs and more efforts are required to investigate retention mechanism and develop efficient removal techniques. Early experiments with a horizontal target and with a limiter configuration suggested that approximately 30% of tritium injected is retained in the vessel with carbon walls and divertor targets. However, recent experiments with vertical target show that approximately 3% of injected tritium is retained in the vacuum vessel. Furthermore, it was shown that the build-up of tritium retention could be reduced significantly by the coverage of the carbon surface by beryllium. The level of tritium retention predicted for ITER will be thus reduced but the uncertainties are large and the methods for its removal need more development. The primary challenge is to remove tritium from shadow areas, and also from surfaces with mixed material deposition (e.g. Be/C/W). High Z materials such as tungsten are attractive for their long lifetime and are expected to be most applicable to future fusion reactors. However, experiments with tungsten show that tungsten can accumulate in the centre in discharges with an ITB (internal thermal barrier). It has also been demonstrated that high central heating could suppress the accumulation. Further experimental studies with all high Z plasma-facing components in large and medium-sized tokamaks are required to investigate the specific restrictions that tungsten may impose on the ITER operation. Surface melting after disruption and subsequent formation of irregular surfaces are of serious concern, since these irregular surfaces can easily melt or evaporate even under normal operation. Since plasma operational flexibility is essential for the initial operation of ITER, the tungsten target is to be avoided. After establishing reliable operation modes, CFC targets would be replaced with tungsten to facilitate tritium retention control and to demonstrate a target suitable for a future fusion reactor and it may be considered to replace the beryllium first wall with tungsten at least partially. The current combination of CFC/tungsten targets and beryllium walls is expected to minimize T retention and W erosion with a wide range of plasma parameters.

7. Conclusions

The design of the vacuum vessel and in-vessel components has progressed considering the fabrication methods, component reliability, costs and interfaces. The plasma physics and assembly/remote handling are taken into account as important interfaces.

Technical specifications are being prepared towards the start of the procurement in 2008. Design of the typical FW/blanket modules and in-vessel components are now well advanced. A qualification program of the FW fabrication methods has started and testing will be performed by the participant teams in 2007–2008.

Review of the design requirements and the current design is now going on and further design changes or optimizations may be proposed in the near future.

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