Progress in Design and R&D on ITER Plasma Facing Components

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Abstract. This paper outlines the changes in the design requirements for the ITER blanket system resulting from the analysis initiated by the design review, and how the blanket design is evolving in order to meet these requirements.

1. Introduction. It has been widely recognised, for decades, that plasma-wall interaction issues will present some of the greatest challenges in achieving a workable fusion reactor. This recognition is based mainly on the high predicted power and particle fluxes to plasma facing surfaces, especially in the divertor region. The proposed resolution of this issue is to operate the ITER divertor in a partially detached regime [1], as has been demonstrated on existing tokamaks (e.g. JET, DIIID, ASDEX-Upgrade [2,3,4]). However, there are still outstanding issues with respect to edge localised modes (ELMs) and their extrapolation to a larger machine, which will only be resolved when ITER is operated.

Less widely recognised are the engineering issues related to all plasma-facing components (PFCs) in a next step device, such as ITER. The mechanical loads are very high, with body forces resulting from induced currents during plasma disruptions, and thermal stresses resulting from a wide range of temperature profiles due to plasma and neutronic heating. These loads need to be reacted in a neutron environment, taking into account fatigue associated with the pulsed nature of the device, whilst allowing for reliable and expedient repair or replacement using remote handling tools.

The ITER construction phase started last year with a design review, including the in-vessel components.It was recognised that the ITER programme should make an important contribution to the suitability of different plasma-facing materials, and that a change of material during the lifetime of the machine would be likely. On the technical side it is clear that no design of PFCs would be able to survive the worst case heat loads without some damage. These combined observations put a great emphasis on the need to have a reliable and expedient method of replacing some or all of the PFCs. For the divertor, which was designed to be able to be replaced several times in the lifetime of ITER, reparability and replacability was considered acceptable. However, for the first wall where front access is restricted, the reliability of the welding and cutting operations of the hydraulic connector could not be guaranteed in the originally proposed design. Plasma loading specifications of the in-vessel components has also been revisited, and in particular the need to use the detailed geometry in estimating the steady state and off-normal power loads to the surface was highlighted. Similarly the flow of halo currents during vertical displacement events (VDEs). All have pointed to a need to readdress the details of the first wall (FW) shaping.

In response to these needs the ITER team, working together with the Parties, has embarked on a design programme to address both of these issues, whilst limiting the impact on the ITER construction schedule. The underlying principle of the re-design is to ensure a reliable procedure for cutting and re-welding of the hydraulic connector, by providing generous frontal access. This is achieved by shaping of the FW elements to provide a central region shadowed from power fluxes flowing along the magnetic field lines, which will also be used to allow for an in-vessel accessible mechanical attachment between the FW and blanket shield module, further simplifying the exchange procedure. On a global level the FW shaping must ensure that no edges are exposed to the parallel heat flux, also accounting for the variable toroidal field ripple, and the penetrations in the port plugs. On the low field side this is achieved by slightly advancing those FW panels between the ports. In this way it has been possible to produce a FW design with sufficient power handling capability to render the start-up limiters redundant.

2. The Requirements.

The specifications of the expected power and particle fluxes to the ITER in-vessel components have been revisited over the past year. In addition to incorporating the latest predictions resulting from ongoing experimental scalings and developing understanding of the processes, the loads have been expressed as much as possible in design independent terms, namely as parallel fluxes. In this way the implications of changes in the wall geometry, for instance to shadow edges, can be directly assessed [5,6].

Important developments for the wall loads have been the observations in several tokamaks of significant power fluxes deep into the scrape-off layer (SOL) both between and during the ELMs. The power fluxes are predicted to be 3.5 MWm^{-2} along the field between the ELMs and 5 MWm^{-2} time averaged power flux during the ELMs at the outside midplane. The fluxes can reach 8.5 MWm^{-2} and 25 MWm^{-2} respectively in the vicinity of the upper X-point. Even these 'normal operation' loads would lead to melting on any exposed edges of first wall panels, which due to their steady state nature could be more damaging than 'off normal' loads.

The energy pulse from each ELM has been the subject of many investigations over the last decade [7]. Apart from the role of the initial surface temperature the effect of the ELM's is little influenced by the cooling capacity of the high heat flux components. It therefore follows that the divertor will present the operational limit for ELM's, the first wall being deeper into the SOL. Even so, based on present scalings, it is unlikely that the first wall would be able to withstand the largest ELM's without some damage.

The most severe thermal loading will be during 'off normal' events such as VDEs, major disruptions, and positional excursions. The power density along the field at the separatrix of a full power H-mode plasma is about 800 MWm^{-2}. The power to the FW will depend upon the proximity, but simulations show that under some conditions the separatrix will touch the FW for a limited duration, which will certainly lead to some melting.

Plasma start-up and ramp-down, also have a large effect on the design of the FW. The heat loads at the higher currents of the limiter phase necessitate the use of special port limiters. In response to this, and for advanced scenario operation, an early X-point formation start-up and ramp-down has been developed, which reduces the maximum power density along the field to 170 MWm^{-2}. It may also be possible to exploit the FW for plasma start-up. For a large number of poloidal limiters >9 the maximum power density along the last closed flux surface is reduced to 56 MWm^{-2}.

In addition to the thermal loads the disruption halo current requirements have been re-expressed to give the parallel current density in the plasma, and the field angle at the wall, figure 1. In a similar way to the power, this allows the halo current loads to be manipulated by changes in the first wall surface geometry [8].

A further significant change to the power requirements has resulted from the increase in energy of the heating neutral beam. The resulting increased shine-through can lead to power densities reaching 4 MWm^{-2} on some outer wall first wall panels.

Another area where the requirements were re-examined was the remote handling (RH). Two issues are central to the RH strategy. First, as mentioned above, some of the load conditions expected will lead to damage of the FW components, and at some level this will require that some FW elements be replaced. Second, as there still exist many uncertainties regarding the best choice of the combination of materials for the first wall and divertor the RH strategy should allow for a complete change out of the plasma facing materials on a ‘reasonable’ timescale. Considering the number and size of the components, and the requirement for hydraulic connections the existing requirement of 2 years for a complete change of the first wall material is challenging, whilst being a non-negligible intrusion on the operational plan. Although the RH requirements have remained unchanged a careful examination of the operation for the blanket modules resulted in the conclusion that there remained significant uncertainty on whether R&D would be able to resolve all of the remaining issues prior to installation of the blanket.

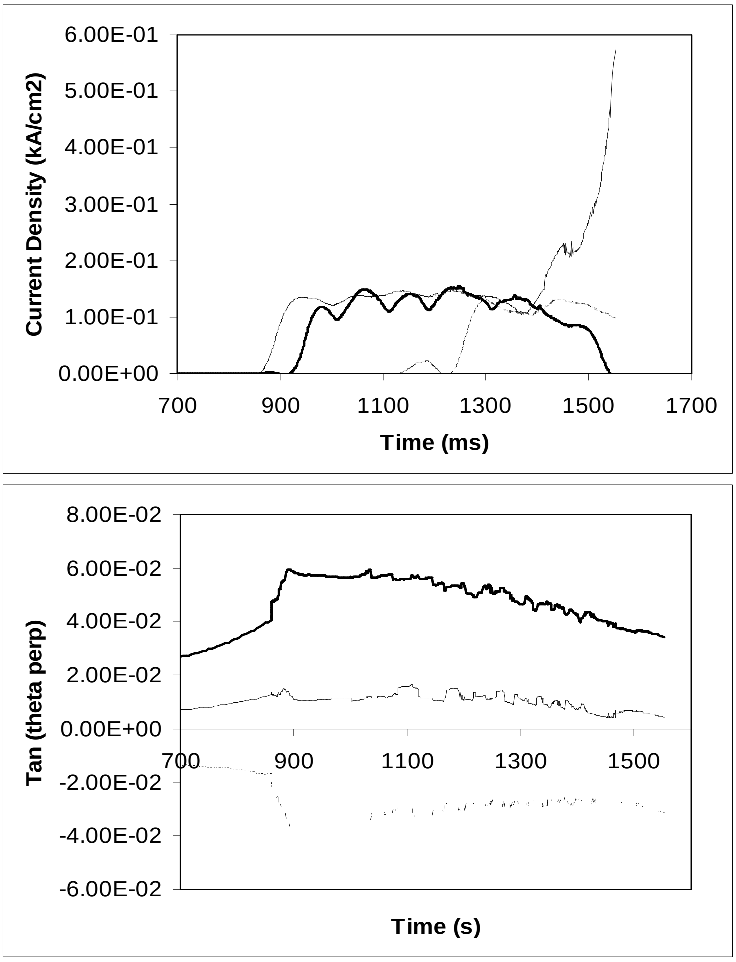

FIG. 1. The local toroidal current density in the halo, and the angle of toroidal penetration of the magnetic field for 3 points on the First Wall panel close to the second X-point for an upward VDE. The current density rises dramatically as the field angle approaches zero.

3. The Blanket Design Strategy.

The primary element of the strategy for design modifications to the blanket is related to the fact that it is a single component in a much larger system. For this reason any changes should have minimal impact on the interfaces with other components. In particular there is a large interface with the vacuum vessel which is one of the first components for which the manufacture must start, and the method of support of the blanket modules must remain unchanged in order to avoid significant delays to the project.

The design strategy adopted to address the concerns with respect to the power handling and remote handling capability of the blanket has two elements, which become interlinked.

For the RH, the major area of uncertainty associated with the original blanket design was in the cutting and rewelding of the hydraulic connector. Access for these operations was through a 30mm diameter hole in the FW panel whilst the tools had to be compatible with cutting and welding tubes of diameters 100mm and 30mm. Laser cutting and welding was proposed with a back-up of a special mechanical cutting tool. Debris produced during the cutting operation, the poor surface finish of the cut surface, and the high alignment requirements for welding were the main concerns. The strategy to simplify this operation is one of improving the access, and using as much as possible conventional and demonstrated techniques, thereby reducing the risk of failure of any required developments. Also given the importance of this operation and the consequences of its failure, a full end to end demonstration is considered a necessity. Increasing the access for this operation has implications on two fronts. First, a larger access hole leads to increased neutron flux with its associated He production limiting the re-weldability of the connector, and second, a large hole in the first wall would have to be shielded from direct plasma contact, reducing the ultimate power handling.

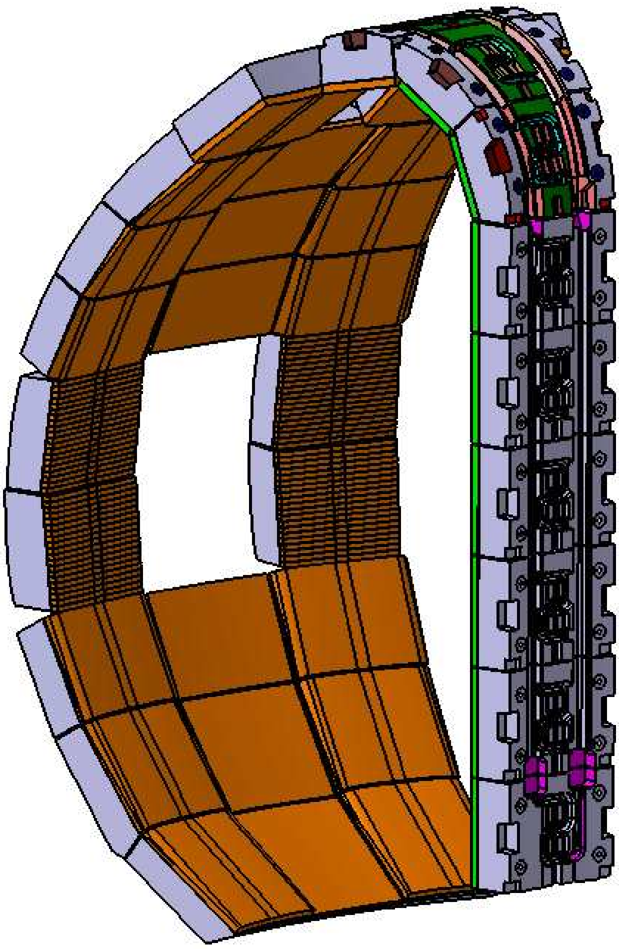

FIG. 2. View of the low field side first wall surface showing how the panels in line with the port openings are recessed with respect to those between. It also shows the shielded central section of the panels allowing for access to the mechanical and hydraulic connections.

For the power handling the major concern is the exposure of edges to the toroidal heat flux. This was a particular concern in the vicinity of the upper X-point, and the in-port structures which have many varying geometries. To shadow the first wall edges, and to provide shadowing for an increased access hole for the hydraulic connector detailed shaping of the first wall front surface is required, and has been well demonstrated on existing tokamaks. For the first wall of the port structures of the heating systems, diagnostics and test blanket modules the limited available surface area can add little to the overall power handling capability of the wall, especially when compared to the additional cost which would result from the diversity of their designs. For this reason it is proposed, on the low field side, to apply a radial offset between the blanket modules in line with the ports and those between, figure 2. As a result of the FW shaping strategy additional possibilities arise. The shielded access hole for the hydraulic connector can be extended, without a loss of power handling capability, to provide frontal access to a mechanical attachment between the FW, and the blanket shield block, such that the FW could be exchanged without removing the shield, and without requiring additional assembly and disassembly operations in the hot cell. In addition, the improved power handling capability, along with the reduced requirements resulting from the early X-point formation may permit the first wall to be used for plasma start-up and ramp-down, which further reduces the parallel power by virtue of its greater effective poloidal length, and may eliminate the need for movable port limiters.

4. The Blanket Design Modifications.

The detailed definition of the modifications to the ITER blanket is still in progress, but the outline proposals are now being analysed and optimised. Figure 3 shows the main features of the design. There is a single in-vessel removable FW panel formed of toroidal fingers mounted on a single central poloidal beam. That the panel is single is important in that it reduces the number operations required for its replacement, but has the negative effect of increasing the electromagnetic loads, which have to be reduced by using fingers.

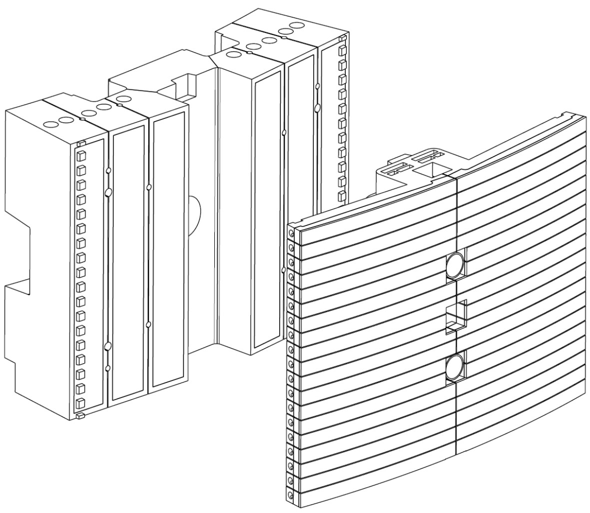

FIG. 3. Outline design of the proposed modifications to the blanket module design. An inner wall module is shown.

Removing the FW panel in-vessel has the advantage, beyond those mentioned previously, that the access holes for the shield module to vacuum vessel attachments, with their inherent reduction in power handling, are no longer required. Toroidal fingers have been chosen to reduce the loads due to halo currents and to allow the fabrication of smaller high heat flux components, compatible with other technologies such as hypervapotrons. The central poloidal beam is attached to the shield block by bolts. The number and size of these bolts is determined by many factors, such as the EM loads, thermal bowing of the structure, manufacturing tolerances, differential thermal expansion, and neutron induced effects. The poloidal moment on the FW panel is reacted by the bolts, and either the tips of the fingers, or the edge of the beam. Using the finger tips provides a longer moment arm, but requires a prestress in the fingers, and is strongly dependant upon the thermal bowing of the fingers. EM analysis of the FW panel during a disruption shows a large current circulating around the poloidal axis, despite its finger geometry. This current crosses the toroidal magnetic field at the finger tips and would lead to an unacceptable poloidal movement. Figure 4. It is therefore proposed to incorporate a series of insulated keys between the shield block and finger tips to react this force.

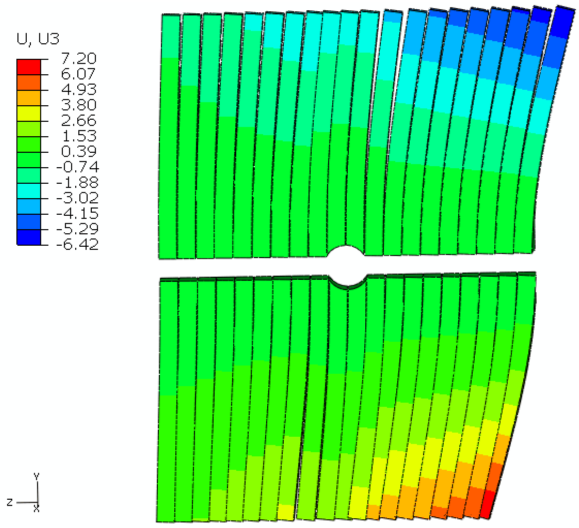

FIG. 4. The calculated displacements of unsupported fingers during a major disruption as simulated by DINA (36ms linear decay). The deflections have been magnified for clarity [9].

Two possible hydraulic connector strategies are being pursued, a coaxial connector and separate connectors for the flow and return. The tube diameter is a balance between access for the cutting and welding tools, and the neutron flux at the point of the weld. By keeping all welds the same diameter both cutting and welding should be simplified. The preferred option for cutting is to use an internal wheel cutter, and test will be started soon to demonstrate this procedure in the exact conditions required. Laser welding is preferred for its reduced heat affected zone, but small diameter tubes, which can be thinner, can be TIG welded at similar He levels, whilst requiring less stringent alignment. Similar connectors will be used both between the FW and shield block, and between the shield block and the blanket manifold.

5. First Wall Shaping.

As previously described, on the low field side only half (18 of 36) of the first wall panels can be effectively used due to the presence of the ports. In principle it is possible to use all 36 at poloidal positions in between the ports, but the ultimate capability would be limited by the port positions, and the transition would add complication to the design. A determining factor in the power density to the FW is the radial setback. This is the distance at which the toroidally facing edge of the FW must be located behind the last closed flux surface to ensure that the power to the edge does limit the power handling. The setback is a function of the plasma configurations, and the difference in the relative radial positions of the FW panels due to manufacturing and installation tolerances. To minimise the latter the blanket modules will be installed using custom machined mountings following a full survey of the interior of the assembled vacuum vessel. Figure 5 shows the calculated heat flux pattern for plasma ramp-up, using a setback which allows for both a 5mm misalignment and for a 100mm wide central access for the hydraulic connector and mechanical attachment. Even though the setback avoids the over exposure of the toroidal facing edges, there is still an increase in the peak power density to the FW surface due to misalignments. At the 5mm limit the power density to the advanced FW panel is increased from 1.32 MWm^{-2} to 1.96 MWm^{-2}.

In the vicinity of the upper X-point the FW must be designed to be compatible with both divertor and limiter configurations, and as such cannot take the form of the optimum uni-directional divertor. In X-point configurations where the poloidal field can be perpendicular to the wall the relatively long e-folding length of the ELM power gives an optimum shape that is not far from a roof top, with an effective area of around 13 m^2, and a resulting time averaged ELM power density of $2MWm^{-2}$.

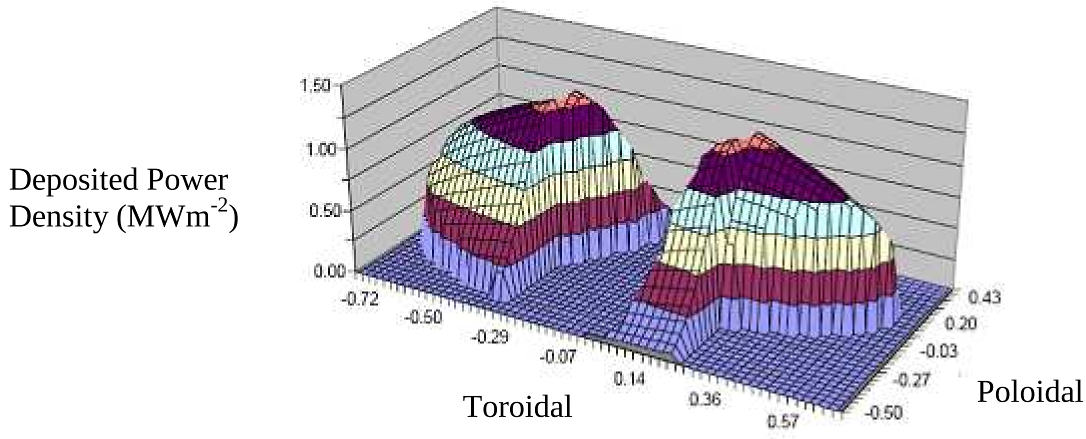

FIG. 5. The calculated maximum deposited power profile on one of the advanced first wall panels during plasma ramp-up[10].

6. High Heat Flux Technology.

The heat flux technology foreseen for the ITER FW panels was one of stainless steel tubes in a copper heat sink with beryllium bonded to plasma facing surface. This technology was consistent with the previously assumed surface heat flux of $0.5MWm^{-2}$, and can potentially withstand $2MWm^{-2}$ for a limited number of cycles. However, it is incompatible with such high power densities during each pulse, due to fatigue. Hence, other options for the heat exhaust technology are being considered. Table 1 summarises the typical limits for some different technologies. Reducing the wall thickness of the stainless steel tube improves the maximum possible power density, but less than halves the thermal bowing. Introducing copper as a water facing material, especially with a thin front surface, as in a hypervapotron significantly improves the power handling, and can reduces the bowing by almost an order of magnitude.

Technology	Maximum Power Density (MWm^{-2})	Comments
SS tube (1mm) in Cu	1 MWm^{-2}	Fatigue lifetime 30000 cycles
SS tube (0.5mm) in Cu	1.8 MWm^{-2} 2 MWm^{-2}	Critical heat flux at (8kg/s) Fatigue lifetime 30000 cycles
Cu tube in Cu	1.8 MWm^{-2} 2.8 MWm^{-2}	Critical heat flux at (8kg/s) Surface temperature 700C
Hypervapotron	4.5 MWm^{-2}	Surface temperature 700C

Table 1. Maximum power densities for various high heat flux technologies

Because of the critical nature of these components, and the specialised nature of the manufacturing process [11], it is necessary that the six involved parties demonstrate their ability to reliably and reproducibly manufacture such components. This process, which started immediately the ITER agreement was in place, was foreseen to be a 2 stage process. The first phase was to produce 2 small elements using the proposed manufacturing technique, which will be fatigue tested in 2 facilities in order to establish the reliability of the beryllium to copper joint. The second phase will be the manufacture a fraction of a full size first wall panel, including all of the important features, and will be subjected to various testing to establish the entire manufacturing technique. It will be essential in this second phase to include the heat exhaust method chosen for the final design.
To ensure that the final products meet the requirements the manufacturing process will have to be closely monitored. In this area the divertor, for which the design is well established, is far more advanced. Specific testing methods have been developed for easily monitoring the integrity of the bond between the carbon monoblock and the copper pipe. By manufacturing samples with know flaws at the interface it has been established which type and level of defects can be detected by which methods. This information, combined with the level of acceptable defects is essential to develop acceptance criteria for the manufacturing stage.

7. Acknowledgements.

This report was prepared as an account of work by or for the ITER Organization. The Members of the Organization are the People's Republic of China, the European Atomic Energy Community, the Republic of India, Japan, the Republic of Korea, the Russian Federation, and the United States of America. The views and opinions expressed herein do not necessarily reflect those of the Members or any agency thereof. Dissemination of the information in this paper is governed by the applicable terms of the ITER Joint Implementation Agreement.

8. References.

[1] Kukushkin, A., et al., Nucl. Fusion **45** (2005) 608.
[2] Loarte, A., et al., Nucl. Fusion **38** (1998) 331.
[3] T. Petrie, D. Buchenauer, D. Hill, et al., J. Nucl. Mater. **196-198** (1992) 848.
[4] M. Laux, K. Buechl, A. Carlson, et al., Controlled Fusion and Plasma Physics (Proc. 20th Eur. Conf. Lisboa, Portugal, 1993), vol. 17C.
[5] A.Loarte, M. Sugihara and M.Shimada, ITER Document, ITER_D_2FHGTA.
[6] A. Loarte, this conference IT/P6-13
[7] Prog. on ITER Phys. Basis, Nucl. Fusion **47** (2007) S203.
[8] M.Sugihara, ITER Document, ITER_D_2EXW4B.
[9] J.Kotulski, ITER Document, ITER_D_2E7APN.
[10] P.Stangeby, ITER Document, ITER_D_2FHFRE.
[11] J.Chen, this conference IT/P7-10