

Power Handling in ITER: Divertor and Blanket Design and R&D

M. Merola 1), D. Loesser 2), R. Raffray 1) on behalf of the ITER Organization, ITER Domestic Agencies and ITER Collaborators

1) ITER Organization, Route de Vinon sur Verdon, F-13115 St Paul lez Durance, France

2) Blanket Integrated Product Team, Plasma Physics Laboratory, Princeton University, Princeton, NJ 08543 USA

E-mail contact of main author: mario.merola@iter.org

Abstract. The Divertor and the Blanket System directly face the thermonuclear plasma and cover an area of about 850 m². The main function of the divertor is minimizing the helium and impurity content in the plasma. The main functions of the blanket system are: (1) to exhaust most of the plasma thermal power, (2) to contribute in providing neutron shielding to superconducting coils and (3) to provide limiting surfaces that define the plasma boundary during startup and shutdown. The development of suitable technologies for the Divertor and Blanket power handling of the stationary thermal loads has required an unprecedented R&D effort by all the concerned Domestic Agencies over the last two decades. This paper summarizes the status of their design and procurement, the related R&D and technology qualification effort, and gives an outline of the concerned schedule.

1. Introduction

The ITER Project is an international effort aimed at demonstrating the scientific and technological feasibility of nuclear fusion energy. The fusion reactions occur within the volume of the ITER vacuum vessel, which contains the thermonuclear plasma. One of the most technically challenging components of the ITER machine are plasma-facing components (PFCs), which directly face the thermonuclear plasma and cover an area of about 850 m². They include the Divertor and the Blanket (*see Fig. 1*). The PFCs consist of a plasma facing material (the “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. The development of suitable technologies for the power handling of the stationary thermal loads has required an unprecedented R&D effort by all the concerned ITER Parties over the last two decades.

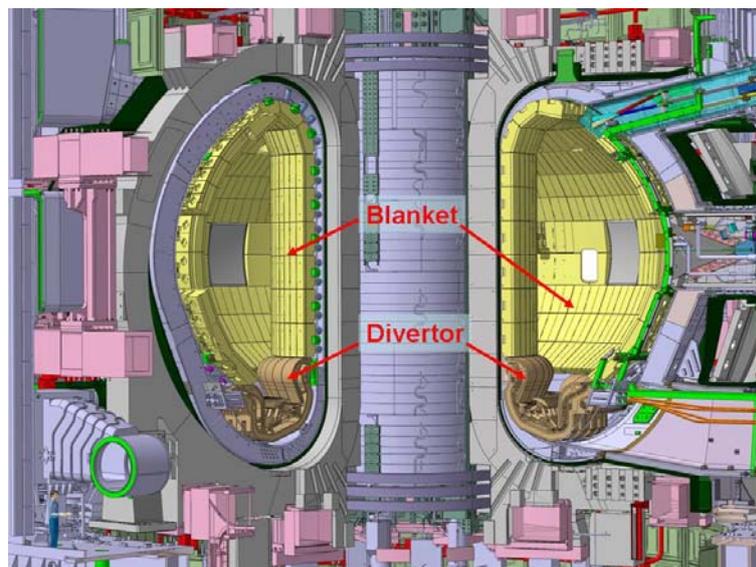


FIG. 1. ITER Plasma-Facing Components.

2. The Divertor

2.1. Main Function and General Layout

The main function of the divertor is minimizing the helium and impurity content in the plasma as well as exhausting part of the plasma thermal power.

The ITER divertor consists of 54 cassette assemblies, with a total mass of 470 tons, and a plasma-facing surface area of $\sim 210 \text{ m}^2$. This is based on the maximum size of a cassette assembly that can be handled via the Remote Handling ports and still have an integer number of cassettes per sector. Of the nine lower ports, three equally-spaced ports are allocated to divertor remote maintenance. The cassette assemblies are inserted radially through the ports and moved toroidally before being locked into position. Each cassette assembly includes one Cassette Body (CB) and three PFCs, namely the inner and outer Vertical Targets (VTs), and the Dome (DO) (*see Fig. 2*). The geometry of divertor is based on simulations obtained using the B2-EIRENE (SOLPS) code and by extrapolation from results from tokamak experiments[1].

The divertor CB is reusable to minimise activated waste; it provides neutron shielding, routes the water coolant and supports the different PFCs. The inner and outer VTs are the PFCs, which in their lower parts directly interact with the plasma scrape-off layer and in their upper parts act as baffles for the neutrals. The DO protects the CB from plasma radiation, baffles the neutral, particularly helium, allows transient movements of the strike points and provide improved operational flexibility of the divertor in terms of magnetic configuration. To maintain a high enough gas pressure to achieve detachment in the outer divertor channel, without recourse to excessive gas puffing, a large unrestricted opening in the private flux region of the DO is provided, beneath the umbrella to connect the inner and outer channels. This allows free re-circulation of neutrals from the inboard to the outboard private flux region and their re-ionisation in the outer divertor plasma.

The divertor plasma-facing material, is either carbon fibre-reinforced carbon composite (CFC) or tungsten (W). CFC is the reference design solution for the lower part of the VTs as far the first divertor set is concerned. This is considered to have advantages for the start of ITER operations given its proven range of compatibility with a number of plasma conditions in present devices, particularly at low densities with significant additional heating. CFC also promises to make easier the development of techniques for ELM control and disruption mitigation, taking advantage of the larger tolerance of the plasma to C versus W and its absence of melting. Finally, vapour shielding effects are more effective in protecting the CFC from further erosion at high power fluxes. Due to concerns on tritium retention, the CFC/W divertor will be replaced by a full-W armoured divertor prior to entering into the ITER nuclear phase.

2.2. Divertor Design Requirements

During normal operation a design heat flux of 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 up to 10 s duration for 300 events also has to be provided. As regards the upper segment of the VT, it is designed to sustain heat fluxes of up to 5 MW/m^2 in steady state to allow for configuration flexibility and broad power deposition profiles in L-mode conditions, as well as high plasma radiation densities in the vicinity of the X-point, which can occur when full performance plasmas operate in partially detached conditions.

The maximum total power to the divertor is 204 MW, of which 140 MW are deposited on the plasma-facing surfaces and 64 MW is due to the neutronic heat deposition. The cooling is ensured by demineralized water with an inlet temperature and pressure of 100 C and 4.2 MPa, respectively and a total flow rate of almost 1000 kg/s. The main driver of the cooling system is to maintain an adequate margin to the critical heat flux (CHF margin > 1.4) for each component at the design heat flux [2].

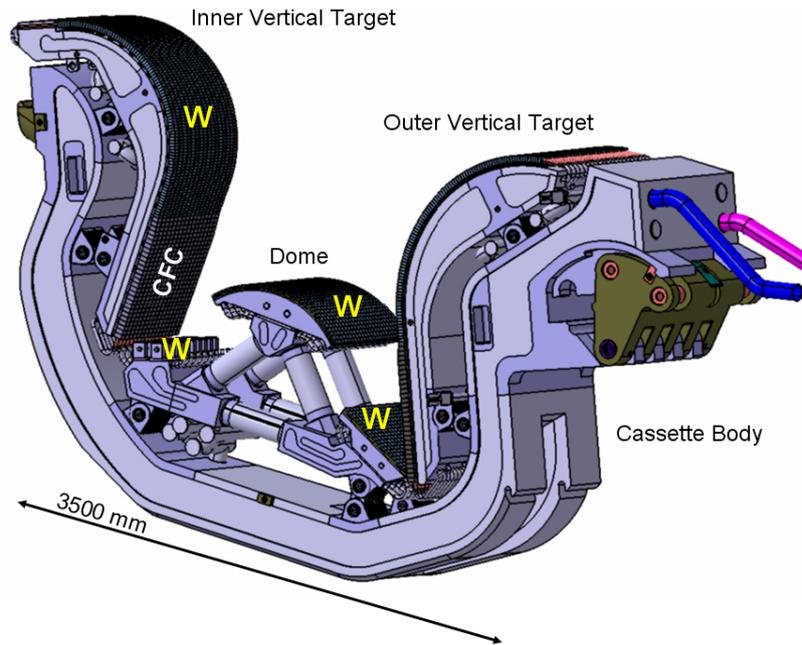


FIG. 2. ITER Divertor: Components and Plasma-Facing Materials for the first set (non-DT phase)

2.3. Divertor Manufacturing Technologies

The development of suitable technologies for the manufacturing of high heat flux components was one of the major efforts of the R&D programme. The main issues, which needed to be addressed were: the large thermal expansion mismatch between the armour and the heat sink; the requirement for two different armour materials (CFC and W) on the same component; and the preservation of the thermo-mechanical properties of the copper alloy CuCrZr during the whole manufacturing process. The joint interface-stress between the armour and the heat sink was mitigated by the introduction of a thin (1-2 mm) pure copper (Cu) interlayer.

As regards the CFC to Cu joint, the European Domestic Agency (DA), which procures the inner VT, has developed the so-called “Active Metal Casting”. It involves casting Cu onto the CFC surface, which is first “structured” by a laser beam to improve the joint strength, and the surface is then “activated” to facilitate wetting. “Pre-brazed casting” is also used, where a brazing filler activates the CFC surfaces before casting Cu [3]. The preferred manufacturing route for the CFC/Cu of the Japanese DA, which procures the outer VT, is based on high temperature brazing, which is followed by a rapid cooling of the component and an ageing thermal treatment to recover the thermo-mechanical properties of the CuCrZr [4]. The Russian DA, which procures the DO, has developed a brazing technology to obtain the W/Cu joint. Alternatively, casting Cu onto W was also demonstrated [5].

2.4. Divertor Maintenance

Due to the erosion/damage of the divertor PFCs and the possible need for improving the design of critical items, the replacement of the divertor is foreseen three times during the 20

years of ITER operation. Therefore the divertor is allocated a Remote Handling (RH) Maintenance Class 1. Divertor maintenance is performed via “off-line” refurbishment (*see Fig. 3*). This requires that a complete spare set of CBs is kept in readiness for the installation of the new PFCs thus removing the refurbishment process from the critical path of a shutdown. In fact, these (new or refurbished) cassettes can be prepared and fully tested in parallel with the operational time [6]. The integration of the first divertor set, as well as of the second set, will be carried out using only “factory built” components. Similarly, the third divertor set will use factory built PFCs and a non activated cassettes. As a consequence, a full *hands-on* procedure can be envisaged. The RH refurbishment is required only for the last divertor refurbishment, which will be carried out only if the ITER operation is extended over the presently planned 20 years. Therefore, the possibility of a so-called “no-refurbishment” option is also being considered, where each replacement will be carried out using only factory built components. This would require a third set of CBs but would allow only hands-on refurbishment operations.

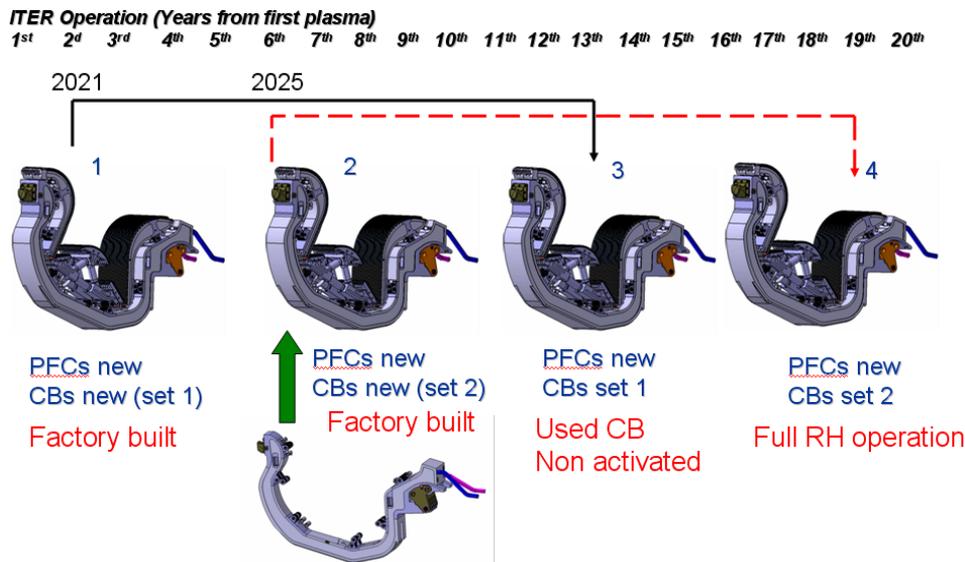


FIG. 3. ITER Divertor Maintenance Strategy

2.5. Divertor Schedule

Following the successful completion of the qualification phase of all the concerned ITER DAs, the construction of the Divertor started in June 2009 with the signature of the “Procurement Arrangements” for the outer VT and the DO with the Japanese and Russian DAs. In March 2010 the procurement of the inner VT was also launched in the European Union. The start of the fabrication of the CB is scheduled during the second half of 2011.

3. The Blanket System

3.1. Main Function and General Layout

The main functions of the blanket system are: (1) to exhaust part of the plasma power, (2) to contribute in providing neutron shielding to superconducting coils and (3) to provide limiting surfaces that define the plasma boundary during startup and shutdown.

The system comprises two different sub-systems that cover the ITER inner vessel wall. The wall-mounted blanket modules covering 620 m² and the port-mounted blanket modules

covering 40 m², which are not described in this paper. The wall mounted blanket modules (BMs) consist of two major components, a plasma-facing First Wall (FW) panel, with a beryllium (Be) armour, supported by a Shield Block (SB).

The BMs are segmented into 18 poloidal locations, BM 1 to 6 cover the inboard region, BM 7 to 10 the upper region and BMs 11 to 18 the outboard region (*see Fig. 4*).

The inboard and upper modules (except BM10) are segmented toroidally into 18 equal modules, and the outboard modules (except BM14 and 15) are segmented into 36 modules. In the upper and equatorial port region (BM10, 14 and 15), the modules are located between ports and therefore segmented into 18 modules.

The main function of the SB is to provide nuclear shielding and supply the FW panel with cooling water. To maximize the nuclear shielding performance, a 85/15 % SS/water ratio is required. This ratio is achieved by optimizing the number of poloidal cooling channels and their size within the SB. 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.

The FW panel consists of two sets of toroidal fingers, which are assembled into the two lateral sides of a central support beam. The two sets allows a central slot in the FW panel, which is shadowed from the direct plasma contact, where the access holes for welding the cooling pipes and the gripping points for the RH of the panel are located.

The FW surface of each panel is defined by the toroidal shaping of the individual fingers. Each finger comprises a series of Be armour tiles, a copper alloy (CuCrZr) water-cooled heat sink and stainless steel (SS) support structure. The FW panel is constructed as fingers to reduce eddy current-related loads. The fingers are assembled on the central support beam in a toroidal orientation to equalize as much as possible the distribution of thermal load and to minimize the mechanical loads due to halo currents.

Each BM 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 to the BM is supplied by manifolds supported off the vacuum vessel behind or to the side of the SB. The cooling water is fed to and from the SB through a coaxial connector and branch pipes. The branch pipes provide sufficient flexibility to cope with the differential thermal expansion of the vacuum vessel and manifolds during operation.

Following to the ITER Design Review in 2007, the blanket has been subject to a substantial redesign. The main features of the new design are: (1) elimination of the Port Limiters and startup and shutdown of the plasma onto the FW panels, (2) shaping of the FW panels to protect leading edges and (3) in-situ removable FW panels to minimize the number of cut and rewelding operations in case of refurbishment of the BM, to avoid large holes in the FW panels for the RH of the SB, and to minimize the space requirement in the Hot Cell.

3.2. Blanket Design Requirements

The FW has to withstand the plasma heat load during operation. The most important design cases are:

Start-up and ramp down: The plasma is limited by the wall before and after the diverted phase of the scenario. Both inboard and outboard contacts are possible. The power entering the scrape off layer is 7.5 MW, a fraction of that of the burning plasma, but the separatrix (last closed flux surface) contacts the wall. The parallel field lines carry up to 50 MW/m². The inboard contact concentrates on rows 3 and 4, the outboard contact on rows 14 to 17.

Flat top: The burning phase last 400 seconds. The major part of the power passing through the scrape-off layer goes to the divertor. The first flux surface reaching the wall is 40 mm outside the separatrix. It can carry up to 33 MW/m² along the field lines, because of the ELMs

which add to the far SOL power. This heat flux reaches the upper panels 7 to 10. There are localized heat loads associated with additional heating systems (sheath heat fluxes for ICH, shine through for HNB).

The heat flux deposited on the FW panels is unevenly distributed and, therefore, there are two sets of FW panels: the “normal” heat flux panels and the “enhanced” heat flux panels with a design load of $0.5\text{--}2\text{ MW/m}^2$ and $3.4\text{--}4.6\text{ MW/m}^2$, respectively. There are 234 normal FW panels (at poloidal rows 1#, 2#, 6#, 10-13# and, partially, 18#) and 216 enhanced FW panels for a total of 440 wall-mounted BMs [7].

The maximum total power to the blanket is 736 MW, of which 596 MW is due to the neutronic heat deposition. The cooling is ensured by demineralized water with an inlet temperature and pressure of 100 C and 3.0 MPa, respectively and a total flow rate of about 3300 kg/s.

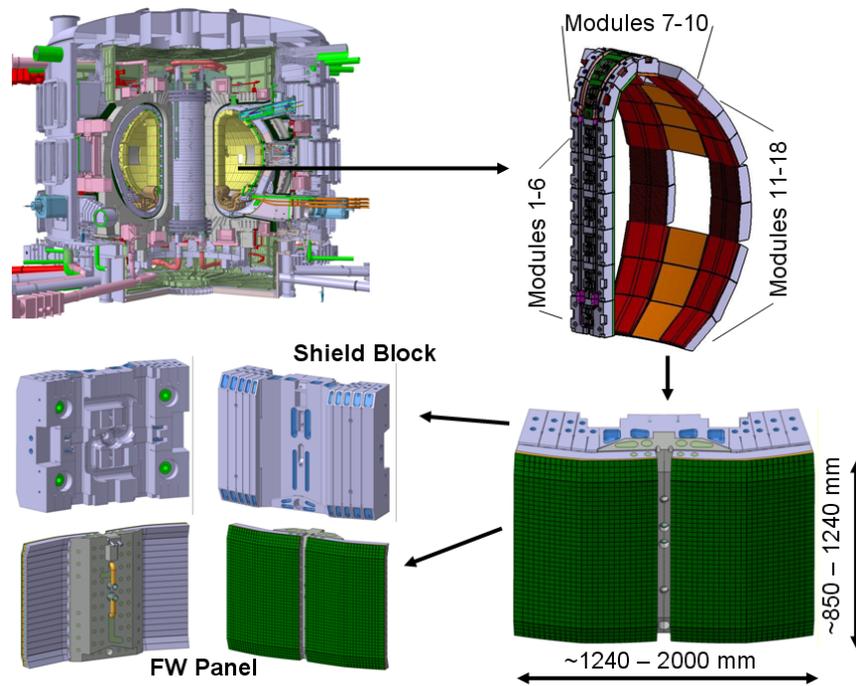


FIG. 4. ITER Blanket Overall Layout

3.3. Blanket Manufacturing Technologies

The Be to CuCrZr joint represent the most technically challenging step of the BM manufacturing process. Hot Isostatic Pressing (HIP'ing) is the reference technology developed by the European [8] and Chinese DA [9], whereas brazing is the preferred route for the Russian DA [10].

Prior to start the procurement phase, each concerned DA shall be subject to a qualification process to demonstrate their capability to manufacture FW panels according to the ITER design requirements. This qualification is carried out in two steps: the first one consists in the manufacturing and testing of small-scale mock-ups to demonstrate the “fitness for purpose” of the proposed Be joining technology (see Fig. 5). The formal test program was comprising 12,000 normal cycles at 0.875 MW/m^2 and 1000 MARFE cycles at 1.4 MW/m^2 . This phase was successfully completed early in 2010 and the FW qualification program will move into a second phase requiring larger components with a shaped FW profile used in the current design. Successful completion of this program will qualify the DAs for the follow on Procurement Arrangement.

A Be qualification test campaign was also completed in order to qualify new Be grades for ITER. Several Russian and Chinese materials have been tested in the electron beam facility JUDITH-1 (Juelich, Germany) and compared to the reference material grade S65C. Beside thermal shock experiments (simulation of disruption- and ELM-like loading) each DA delivered two actively-cooled mock-ups for VDE simulation testing.

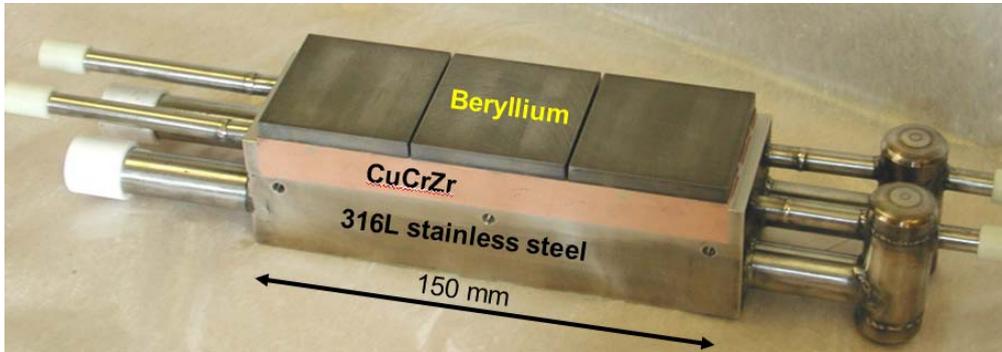


FIG. 5. ITER Blanket: FW Qualification Mock-ups

3.4. Blanket Maintenance

The BMs are designed to survive for the 20 years of ITER operation from a fatigue and erosion stand point. However, as regards the FW panels, the possible loads related to transient events (runaway electrons, VDEs, H-L transitions), may lead to a higher failure probability than once in 20 years (based on RAMI analysis). As a consequence, there is a requirement to allow exchange and repair of FW panels due to operational damage, or a desire to change the first wall material. As a consequence, the FW panels are RH Class 1 components.

The SBs are not expected to be damaged since they are protected from a direct plasma interaction by the FW panels. Therefore, major maintenance or upgrading is not anticipated. However the in-vessel coils and the blanket manifold, which are located behind the SBs, may require unscheduled maintenance. As a consequence, the SB has a RH Class 2 [11]. This would also enable their possible replacement in case of unexpected damage including leaks, and to change the shielding blanket to a breeding blanket as defined by the operational plan.

3.5. Blanket Schedule

Following the Conceptual Design Review (CDR) of February 2010, the focus moves from post-CDR work to preparing for the Preliminary Design Review (PDR) of the Blanket (to be held in mid 2011). A number of key modules have been identified for detailed design and analysis based on a combination of electromagnetic loads, heat loads, interface and design complexity. Design activities also include the blanket manifolds, the BM to vacuum vessel attachment system as well as RH experimental demonstrations. The PDR will be followed by the Final Design Review (FDR) late in 2012. In parallel, the blanket technology qualification programme will continue with the aim to be completed also late in 2012. Soon afterwards, the procurement of the Blanket system for ITER is scheduled to start.

4. Final Remarks

The Divertor and Blanket systems are one of the most technically challenging components of the ITER machine. Following to the successful completion of the R&D and qualification technology phase, the Divertor reached a stage of maturity to allow the start of its construction in 2009.

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 of the new reference solutions and now the effort is focused on finalizing the design work and the supporting R&D programme in view of the start of the procurement late in 2012.

5. Acknowledgement

The views and opinions expressed herein do not necessarily reflect those of the ITER Organization.

References

- [1] R. A. Pitts, A. Kukushkin, A. Loarte, A. Martin et al., “Status and physics basis of the ITER divertor”, *Physica Scripta* Vol. T138, 2009.
- [2] A.R. Raffray, J. Schlosser, M. Akiba, M. Araki et al., “Critical heat flux analysis and R&D for the design of the ITER divertor”, *Fusion Eng. Des.* Vol. 45, pp. 377-407, 1999.
- [3] M. Merola, J. Palmer, “EU activities in preparation of the procurement of the ITER divertor”, *Fusion Eng. Des.* Vol. 81, pp. 105-112, 2006.
- [4] S. Suzuki, K. Ezato, Y. Seki, K. Yokoyama et al., “Recent activities related to the development of the plasma facing components for the ITER and fusion DEMA plant”, *Physica Scripta* Vol. T138, 2009.
- [5] R. Giniatulin, A. Gervash, V.L. Komarov, A. Makhankov et al., “High heat flux tests mock-ups for ITER divertor application”, *Fusion Eng. Des.* Vol. 39-40, pp. 385-391, 1998.
- [6] J. Palmer, M. Irving, J. Järvenpää, H. Mäkinen et al., “The design and development of divertor remote handling equipment for ITER”, *Fusion Eng. Des.* Vol. 82, pp. 1977-1982, 2007.
- [7] R. Mitteau, P. Stangeby, C. Lowry, M. Merola, “Heat loads and shape design of the ITER first wall”, *Fusion Eng. Des.*, 2010, in press.
- [8] P. Lorenzetto, B. Boireau, C. Boudot, Ph. Bucci et al., “Status of the EU R&D programme on the blanket shield modules for ITER”, *Fusion Eng. Des.* Vol. 83, pp. 1015-1019, 2008.
- [9] J. M. Chen, X. Liu, J. H. Wu, L. R. Shen, et al., “ITER first wall fabrication technology in China”, 22nd IAEA FEC 2008, Oct. 13-18, Geneva, Switzerland.
- [10] V. Belyakov, I. Mazul, Yu. Strebkov, “Manufacturing and testing of large-scale mock-ups of ITER plasma facing components in Russia”, *Fusion Eng. Des.* Vol. 61-62, pp. 129-134, 2002.
- [11] N. Takeda, S. Kakudate, Y. Matsumoto, H. Kozaka et al., “R&D on major components of control system for ITER blanket maintenance equipment”, *Fusion Eng. Des.*, 2010, in press.