Technology R&D Activities for the ITER full-Tungsten Divertor

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Abstract. To reduce costs, the ITER Organization (IO) has proposed to reduce by one the number of divertors planned during the ITER lifetime and to begin operations with a full-tungsten variant in place of the original first divertor which was armoured with carbon fibre composite in the high heat flux regions. The ITER Council has requested that the proposal be subject to an in-depth study during which the physics implications of a full tungsten start to operations be assessed, a design developed and a technology programme launched within the procuring domestic agencies to confirm that tungsten monoblocks with the required heat handling capacity can be produced to the required standards. A decision on which divertor to manufacture is scheduled to be taken by the end of 2013. A design and development work programme has thus been launched at the IO and the three concerned Domestic Agencies, Europe, Japan and the Russian Federation, to prepare for the procurement of a full-tungsten divertor, including the manufacturing and testing of small scale mock-ups, full-tungsten pre-qualification and full-size prototypes. This paper presents the status of progress of this development work programme.

1. Introduction

The current ITER Baseline foresees the use of carbon fibre composite (CFC) as armour material in the high heat flux (HHF) strike point regions of the first divertor to be installed for operation in the non-active (hydrogen and helium) phases. Tungsten (W) is used on all other plasma-facing locations and in fact dominates the total surface area of these components in the divertor [1]. As a result of the high levels of tritium retention expected when using carbon, this first divertor would be replaced with a full-W variant for the nuclear phase with deuterium and deuterium-tritium plasmas.

An extensive Research and Development (R&D) programme dedicated to fabrication technologies has been implemented over more than 15 years in the three Domestic Agencies (DA), Europe, Japan and the Russian Federation, who will supply the first ITER divertor components [2–4]. Having been established for the CFC/W divertor variant, the design solution for W-armoured components was optimized for the divertor baffle and dome regions (see FIG. 1). In these plasma-facing locations, the demands on heat handling are considerably lower than in the HHF regions, with a requirement for steady state heat handling of typically 5 MW/m², with short transient events (~2 seconds) up to 10 MW/m². A very positive outcome of this R&D effort has been that some fabrication technologies can achieve much higher performance, close to the slow transient conditions expected in the strike point regions during nuclear phase operation (i.e. 20 MW/m²).
In November 2011, the ITER Council endorsed a cost containment proposal from the ITER organization (IO) to eliminate the CFC/W divertor and begin operations with a full-W variant. This endorsement was accompanied by a request that a work programme be implemented to (i) develop a full-W divertor design, (ii) qualify the corresponding fabrication technologies and (iii) investigate critical physics and operational issues with support from the R&D Fusion Community. A decision is to be taken near the end of 2013, once the results of this work programme are known.

This paper presents the status of progress of the technology R&D work programme implemented by the DAs to prepare for the procurement of a full-W divertor.

2. Full-W divertor design

In order to minimize impacts on the divertor Cassette Body (CB) design, diagnostic interfaces, running Procurement Arrangements (PAs) and on the overall procurement schedule, the approach used by the IO [8] has been to leave the baseline CFC/W divertor design largely unchanged, making modifications only where necessary (for example, to mitigate as much as possible against melting during fast transient heat pulse events). As a result, the key features of the design remain as described in [1]. The CB routes the water into the plasma-facing components (PFC) and provides neutron shielding. The PFCs are the inner and outer vertical targets (IVT, OVT), the dome umbrella and reflector plates (see FIG. 1). Both the IVT and OVT retain the use of monoblocks bonded to single copper chromium zirconium (CuCrZr) pipes (see FIG. 2) with unchanged tolerances at plasma-facing surfaces and attachment concept. However, in the full-W divertor, the monoblocks are now all made from W material and cover the entire length of the targets, forming rows of poloidally extended plasma-facing units (PFU). The dome umbrella and reflector plates still use W flat tiles and the changes are limited to the tilting angle of the PFUs. The entire divertor is made up of 54 individual cassette assemblies carrying the PFCs.

The PFC tilting angle – already implemented in the current CFC/W design - ensures no leading edges from cassette to cassette during the assembly onto the divertor rails in the vacuum vessel. The introduction of W in the strike-point region requires the systematic shadowing of monoblock edges at the PFU scale (see FIG. 2 and [5, 8]). For this purpose, “fish-scaling” is adopted on the monoblock surfaces (2-D toroidal chamfer with 0.5 mm radial depth). The value of 0.5 mm ensures full protection of toroidal leading edges for the maximum local tolerance at the PFC surfaces (maximum toroidal step of 0.3 mm from monoblock to monoblock), while maintaining an acceptable wetted area on each monoblock (>74%). The same fish-scale scheme is applied for both the IVT and OVT.

The intense transient energy loads which are expected on the OVT baffle region during downwards vertical displacement events (VDE) [5] raise a third shaping issue, which can be
reduced by adopting the same “roof-shaping” principle used in the ITER First Wall panel design [6]. Toroidal “set-backs” (typically 15 mm implemented on 3 PFUs) are under study on each side of the baffle to completely shadow any direct leading edge. Designing the OVT baffle region in this way introduces the need for a transition from global roof-shaping to local fish-scale shaping in the strike-point region and is one of the main difficulties of the current full-W divertor design. Moreover, this transition region needs to accommodate an armour thickness variation, from 8 mm on the flat target (HFF) region to 13 mm at the higher shoulder of the chamfered monoblocks in the baffle region (8 mm at the lower shoulder). Heat flux deposition analysis is underway to optimize the plasma-facing surface profile for both the increased steady state loads which will be experienced if the baffle is toroidally shaped and the heavy transient energies expected at the VDE.

The development of the detailed design of the full-W divertor includes 3 main phases [7, 8]:

- a pre-detailed design phase, to provide a design for the cost estimation by DAs, and the input for neutronic and thermo-mechanical analyses;
- a preliminary design phase to finalize the structural design;
- a final design phase to deliver the design with 2-D drawings according to the specified loads and based on the results from R&D in the DAs (see section 3.1).

The objective is to provide the final design on time for the armour material selection (i.e. by the end of 2013).

3. Technology R&D activities for the procurement of a full-W divertor

3.1 Full-W divertor qualification programme

Prior to starting the series production, each procuring DA must first demonstrate its technical capability to carry out the procurement within the applicable quality requirements by passing a qualification test programme. The full-W divertor qualification programme developed by the IO in agreement with the concerned DAs consists of two steps:

- Step 1 - Technology Development and Validation (R&D activity);
- Step 2 - Full-scale prototype demonstration.

The outcome of Step 1 will be used as input for the final design of the full-W divertor. In this step, the DAs must provide evidence that at least 3 small-scale mock-ups have the capability to withstand 5000 cycles at 10 MW/m² plus 300 cycles at 20 MW/m² in accordance with physics analysis which takes into account the predicted numbers of plasma pulses expected on the basis of the ITER Research Plan [9]. The monoblock attachment performance to the steel support structure must also be demonstrated through uniaxial tensile testing and cyclic zero-pull fatigue testing.

Step 2 demonstrates the feasibility of selected fabrication techniques by manufacturing and testing full-W full-scale PFU prototypes with similar level of performance as for the small scale mock-ups of Step 1 and in conformity with the ITER quality requirements.

3.2 R&D activities for the procurement of the IVT

Since the time of the ITER Engineering Design Activities (EDA) phase, the EU has implemented a significant R&D effort to develop fabrication technologies for the manufacturing of divertor vertical targets, in particular for W monoblock armoured components. The manufacturing processes are now fully defined and include two main phases: i) the preparation of W monoblocks and ii) the joining of the monoblocks to a CuCrZr alloy heat sink pipe. The first phase consists in the machining of W monoblock tiles followed by coating, mainly by casting, of the inside walls of the cylindrical hole passing through the monoblocks with a pure copper interlayer. In the second phase, the joining techniques
developed respectively by Plansee S.A. and Ansaldo/ENEA are Hot Isostatic Pressing (HIPing) and Hot Radial Pressing (HRP). In both cases, the joining process is critically dependent on the copper interlayer forming the interface between the W monoblocks and the CuCrZr pipe. The HIP process can be performed at both high (about 900°C) and low temperature (below 600°C). In the case of the higher temperature process, the CuCrZr heat sink is subsequently thermally treated (solution annealing + quenching + ageing) to recover the required microstructure and properties. The HRP process is a diffusion bonding based process activated by pressurizing the CuCrZr pipe at the selected temperature (usually ≤ 600°C) for 2 hours.

The design solutions for divertor W-armoured components have been optimized for low heat flux values, typically 5 MW/m², representative of the baffle region (see FIG. 3), but the results obtained in the EU have shown that the developed technologies also meet the ITER requirements for the strike point region. The performances obtained so far may be summarized as follows.

Tests during the ITER EDA phase showed that W monoblocks (of reduced size) cladding 10 mm inner diameter (ID) cooling pipes successfully survived thermal fatigue testing up to 18-20 MW/m² for 1000 cycles. Similar positive results were also obtained on irradiated W mock-ups, which sustained 18 MW/m² for 1000 cycles after being neutron irradiated at 200°C to 0.1 and 0.5 dpa [10].

![FIG. 3: CFC-W pre-qualification prototype](image)

![FIG. 4: Full W mock-up (EUDA)](image)

More recently, W monoblock mock-ups (see FIG. 4) armouring 12 mm ID cooling pipes (current baseline geometry) were successfully tested up to 15 MW/m² for 1000 cycles and 20 MW/m² for 300 cycles [11]. These findings indicate that the baseline W monoblock design could be acceptable for the new requirements for a full-W vertical target, as reported in Section 3.1. By increasing the number of cycles at 20 MW/m², further experiments revealed i) a progressive W microstructure degradation leading to W surface microcracking and melting and ii) macrocracking of the monoblocks without armour-heat sink joint damage (see FIG. 5).

In collaboration with the Efremov and TRINITI Institutes (RF), a study has been performed of the behaviour of W monoblocks under a series of exposures to steady state and transient heat fluxes, with the latter resembling the heat pulses expected in the ITER divertor during edge localized mode (ELM) activity [12]. This was achieved by alternating the exposure of the mock-ups to ELM-like loads in a quasi stationary plasma accelerator (QSPA) with HHF cycling (1000 cycles at 10 and 20 MW/m²) in an electron beam test facility. The principal conclusion of this study was that, notwithstanding the heavy cracking produced in the W material by plasma exposure and electron beam thermal fatigue testing, all the mock-ups survived the planned testing campaigns without degradation of their power handling capability.
Based on these promising results and to prepare for the work now specified in the IVT PA, the EUDA has launched an extensive program to demonstrate the fabrication feasibility of a full W divertor. These activities include:

- the manufacturing and HHF testing of small-scale mock-ups with different W monoblock geometries and of pre-qualification prototypes. This activity aims at increasing the experimental database for thermal fatigue performance, including testing of alternative W material grades, and to demonstrate that the PFU design fulfils the ITER requirements of the strike point region.

- the manufacturing and testing of full-size, full-W IVT prototypes, a necessary step to qualify industry and be allowed to start the series production.

This EU development work programme is planned to be completed by the end of 2014.

3.3 R&D activities for the procurement of the OVT

In the Japanese DA (JADA), W-armoured divertor components have been developed both for the baffle region of the ITER divertor OVT [1] (see FIG. 6) and for DEMO divertor components [13]. In addition, JADA has initiated voluntary R&D in 2012 on the full W-armoured ITER divertor target, based on the IO proposal to begin operation with a full-W divertor. In 2012 it has manufactured 12 small-scale W-armoured divertor mock-ups for HHF testing (see FIG. 7).

All the W monoblocks constituting these mock-ups are manufactured of pure tungsten in accordance with the standard ASTM B760-86. The cooling pipes made of CuCrZr (UNS C18150: 15mm outer diameter, 12 mm ID) are brazed onto a pure Cu interlayer with 1 mm thickness using Ni-Cu-Mn braze filler material. Prior to the brazing, the pure copper interlayer has been bonded onto the W monoblock using 3 different methods: 1) direct casting of pure copper, 2) HIP bonding and 3) uni-axial diffusion bonding. Moreover, W monoblocks with different thicknesses and longitudinal lengths have been selected in terms of reduction of the monoblock surface temperature and the thermal stresses at the bonding interface generated by the thermal loads (Table 1).

After the bonding of the pure Cu interlayer onto the W monoblocks, the latter are brazed onto the CuCrZr cooling pipe and aged in a vacuum furnace. The brazing temperature is 980 °C for 30 min, similar to the solution annealing temperature of the CuCrZr alloy. Massive nitrogen gas injection into the furnace is then rapidly performed up to a pressure of 200-300 kPa, followed by fan cooling to increase the cooling rate up to 5°C/s. This fan assisted rapid
cooling aims to achieve over-saturation of the precipitates in the CuCrZr alloy. Sufficient mechanical strength can then be recovered subsequent ageing heat treatment (480°C for 2 hours). Following the brazing and ageing processes, ultrasonic testing (UT) performed on these mock-ups revealed no defects at either bond interface (W/Cu and Cu/CuCrZr) exceeding the acceptance criteria of 70° defect angle at the circumference. These mock-ups will be mainly tested in JA-DA’s HHF test facility (JEBIS) in 2012-2013.

<table>
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Table 1: Main dimensions of the W monoblocks in small-scale mock-ups (JA-DA)

3.4 R&D activities for the procurement of the divertor dome

The divertor dome PFUs consist of 316L(N)-IG steel base structures joined by full penetration laser welding onto bimetallic CuCrZr alloy/316L(N)-IG steel covers (see FIG. 8). The bimetallic covers are manufactured from CuCrZr alloy plates and 316L(N)-IG steel plates by explosion bonding and have a hypervapotron design. The welding thickness for all PFUs is 6 mm. After welding the PFUs are machined into the required shape.

Armouring technology for the divertor dome is based on flat bimetallic 8 mm thick W / 2 mm thick Cu tiles of approximate dimension 23×24×(8W+2Cu) mm³ produced by a casting technique. The bimetallic W/Cu tiles are bonded onto a CuCrZr alloy substrate by brazing with STEMET® 1108 Cu-based braze alloy using industrial vacuum/gas quenching furnaces with ohmic heaters (see FIG. 9). The heating cycle combines a brazing temperature of ~970°C for ~60 mins with fast cooling at a rate higher than 40°C/min down to an ageing temperature of 480 °C for about 2 hours and then free cooling down to room temperature. Such a brazing cycle allows a joining process with subsequent recovery of the CuCrZr alloy properties. During the previous ITER CFC/W divertor qualification phase two medium-scale dome qualification prototypes were successfully manufactured and tested.

Considering the large number of W tiles (e.g. 58 per umbrella PFU) and that their joining onto the bimetallic structure is one of the final fabrication steps, a failure of this operation would lead to significant loss of time and resources. It is therefore important to develop a repair process that would limit this risk. Trials have therefore been performed in which brazed tiles are removed by electric discharge machining from already fabricated mock-ups, followed by re-brazing (according to the cycles above) of new W/Cu tiles at these locations, with the
neighbouring tiles in place. Since re-melting of the braze alloy in already brazed joints occurs at higher temperatures, the neighbour tiles did not detach during the repair process.

This repair technique was successfully demonstrated on five dome mock-ups with both flat and curved armoured surfaces (two umbrella mock-ups and three particle reflector plate mock-ups (see FIG. 10)). All survived 1000 cycles at 3 MW/m² plus 1000 cycles at 5 MW/m² without failure [14]. Some local overheating (up to 780°C compared with an average of 640°C) was observed in only a single umbrella PFU, after ~700 thermal cycles at 5 MW/m². Continuation and completion of the thermal cycling showed that this overheating remained stable, within 30% of the expected temperature, and its area did not expand. The test was therefore considered successful. UT of the overheated area revealed small defects in the brazed Cu/CuCrZr alloy joint not exceeding ~15% of tile joint area. This shows both the robustness of the fabrication technique and the tolerance to possible fabrication defects.

Dome R&D activity concerns the current divertor design but can be directly applied to the full W divertor for which the only difference is in the PFU tilting angle. This issue has already been addressed for the manufacture of reflector plates in the current Dome design. Tilting the umbrella PFUs should not raise big problems and will be validated with the manufacture of representative mock-ups when detailed drawings will be made available by IO.

4. Conclusion

The design of a full-W divertor is being prepared by the IO, leaving the baseline divertor design largely unchanged and making modifications only where necessary, e.g. ensuring no leading edges on the PFCs to mitigate as much as possible the effects of melting during transient events. This is considered one of the highest risk factors associated with beginning ITER operations on a full-W divertor.

The fabrication technologies developed by the procuring DAs are being adapted to the more demanding heat loading requirements for PFCs in high heat flux areas in comparison with the loads specified for the W components in the baseline CFC/W divertor. This is being
performed through the manufacture and testing of small scale mock-ups up to full-scale prototypes. The extremely promising preliminary results, approaching the revised heat load requirements for the full-W divertor, which have been obtained on small scale mock-ups, combined with good results obtained on larger scale components of the baseline design give confidence in the success of this fabrication development work.

If confirmed in the next months, such a successful outcome will be an important achievement, together with the resolution of other important physics and plasma wall interaction issues, in the input required to take a decision to begin ITER operations on a full-W divertor.

Acknowledgements

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

References