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REVIEW ON THE DEVELOPMENT OF ITER LIKE DIVERTOR PLASMA FACING COMPONENT

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Abstract- *Development of plasma facing component of divertor for the world's largest experimental tokamak nuclear fusion reactor. Reviewer has studied different plasma facing components and also highlighted the development process of plasma facing component. Divertor is one of the key component of fusion reactor and its function is to extract heat and helium ash.*

At the beginning, CFC is selected as a lower portion of OVT(outer vertical target) and IVT(Inner vertical target) for its superior thermal shock resistance and high thermal conductivity and tungsten(W) is selected as an armor material for Dome, flat plate and upper region of OVT and IVT. Design and development of monoblock is describe and different techniques for joining the heat sink and cooling tube were discussed. Fishing scale are provided for edge protection of Monoblock. Development of small, medium and full scale plasma facing component with actively cooling system. Full set of divertor is ready and provided to ITER. More R&D work is require on full W divertor component.

Keywords: *OVT, IVT, CFC, Divertor, ITER, Monoblock, Tokamak*

I. INTRODUCTION

ITER is an international nuclear fusion research and engineering megaproject, which is currently building the world's largest experimental tokamak nuclear fusion reactor. The divertor is one of the key components of the ITER machine. Situated along the bottom of the vacuum vessel, its function is to extract heat and helium ash (both products of the fusion reaction and other impurities from the plasma) in effect acting like a giant exhaust system.

The ITER divertor is made up of 54 remotely-removable cassettes, each holding three plasma-facing components or targets. Divertor is mainly divided in to three parts- vertical target, dome and flat plate. The targets are situated at the intersection of magnetic field lines where the high-energy plasma particles strike the components. Their kinetic energy is transformed into heat. The heat flux received by these components is extremely intense and require active water cooling. The choice of the surface material for the divertor is an important one.

II. EXISTING LITERATURE

A. Literature on plasma facing materials

V. Barabash 1996[1]. Beryllium, carbon fiber composites and tungsten are under consideration as candidate materials for ITER plasma facing components. The final selection of the materials to be used in ITER, at least for the first operation phase, will be done only at the end of EDA and will be based on the results of the R&D program which includes the study of the features of plasma surface interaction, investigation of physical and mechanical properties, thermal fatigue/thermal shock resistance, including neutron irradiation and technological studies (joining, high heat flux response, etc.).In comparison with Be and W, carbon materials have been widely used as armour in present tokamaks and their performance is well established. Among the various different carbon based materials the carbon fiber composites (CFC) have been selected as a reference materials. The main reasons are good mechanical strength, superior thermal shock resistance, and high thermal conductivity.

J.W. Davis 1998[2]. Tungsten has been selected as armor for the divertor upper vertical target, dome, cassette liner, and for lower base because of its unique resistance to ion and charge-exchange particle erosion in comparison with other materials. The issues related to the use of tungsten in ITER are described in this paper. The different tungsten grades (pure, dispersion strengthened and cast alloys) which are being considered as candidate materials are evaluated. A comparative analysis has been made of the mechanical properties of the various tungsten grades in different thermo mechanical conditions, including the impact of irradiation effects. Three different tungsten grades have been evaluated for possible use in ITER plasma facing components. Key in material selection are the features of the operational conditions and design requirements for ITER.. Generally, there is a very limited database for the tungsten grades within the temperature range of interest to ITER. Therefore, the material selection must be based mainly on limited information. Among the different tungsten grades, W±1% La2O3, W-13I, and pure

W in the cold-worked condition are all considered candidates. For the final selection, additional R&D is needed.

H. Bolt 2004[3]. Tungsten shows the highest promise as plasma-facing material. Experiments in the ASDEX, upgrade tokamak indicate that plasma operation is feasible with walls and divertor surfaces mostly covered with tungsten. Results from large scale application of W as PFM in the ASDEX upgrade tokamak prove that plasma operation with W as PFM is possible in a divertor tokamak without major constraints on the operational flexibility. In a reactor operating with very long pulses and a sufficient clearance between wall and plasma the wall erosion should not be governed by plasma ions, but by the comparatively low flux of energetic charge exchange atoms which would result in very low wall erosion values.

Mario Merola 2010[4]. Study on ITER plasma facing components and described its 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 tokamak, 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).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 vapour shielding effects at high power fluxes tend to enhance W erosion compared with C. The power handling parts of the upper region of the VTs and the DOME will be manufactured with W because of its high threshold for physical sputtering, low tritium retention, high melting temperature and good thermal conductivity.

B. Literature on development of plasma facing component

M. Merola 2000[5]. Working on manufacturing and testing of a prototypical medium scale divertor vertical target for ITER. A medium-scale vertical target prototype has been successfully manufactured with all the main features of the corresponding ITER divertor design. Different joining techniques have been used during the manufacturing process, namely: Active Metal Cast and brazing for the CFC tiles, metal cast and EB welding for the W tiles, HIP'ing for the DS-Cu heat sink, EB welding for the integration of the high heat flux part onto the steel back plate, EB welding for the DS-Cu to stainless steel tube transition via a nickel adapter, TIG welding for the coolant connections.

M. Missirlian2005[6]. Developed full-scale vertical target prototype of ITER divertor. Full scale prototype is 1000mm long and respects the main points of ITER divertor design. Four units (A–D) having a full monoblock geometry, obtained by drilling a hole into each block armour, were assembled in parallel and actively cooled. The lower part of the prototype (blocks 1–32) has a carbon fibre reinforced carbon (CFC) armour, grade NB31.The upper part of the prototype (blocks 33–64) is covered by tungsten (W). A commercial grade was used (WL10) which contains 1% of La_2O_3 and has easier machining, lower cost and higher re-crystallisation temperature when compared with pure W. Active Metal Casting is used to join Cu onto the CFC armour, which has been previously activated to enable the wetting and laser structured to improve the joint strength . The W/Cu joint was obtained by a conventional casting process.

T. Hirai 2010[7]. Design and materials for first set of Divertor and the integration into the ITER tokamak were presented. The design was completed and procurement of divertor components started in Japanese, European and Russian Domestic Agencies (DA). Manufacturing routes of these components shall be selected by each DA. To meet ITER requirements, numbers of NDTs at the various manufacturing stages of components are mandatory for qualification. First divertor set is in production. The paper describes about plasma facing component design and its manufacturing process.

Eliseo Visca 2011[8]. Describes on manufacturing, testing and post-test examination of ITER divertor vertical target and tugnsten (W) small scale mock-ups. The capability to manufacture W monoblock high heat flux components by hot radial pressing (HRP) was already assessed in previous fusion activities and this experimental study confirms that this manufacturing technique can be successful applied to W monoblock components. The paper also describes the geometry of the tested mock-up was different from the W mock-ups manufactured under different contracts and tested up to 18–20MW/m2:

- The axial thickness of the monoblock has been increased from 4mm to 12mm.
- The monoblock are 28mm large instead of 23mm.
- The interlayer is 0.5mm instead of 1mm.
- The copper tube is 15mm OD 1.5mm thick instead of 12mm OD 1mm thick.

The decreasing of the Cu interlayer thickness is always found under the W monoblock and precisely in the centre of the zone faced to the heat flux loading. The results obtained from the thermal fatigue testing and the destructive examination indicate that the testing conditions are very close to the maximum thermal limits of these mock-ups. The presence of melting zones in the W surface and also in the copper interlayer zone need to be better investigated by further FE calculations and thermal load tests.

R.A. Pitts 2011[9]. Describes on the design of plasma facing component on physics basis. The complete ITER divertor consists of 54 fully remote handling compatible cassette assemblies, each weighing around 9 tonnes and consisting of a cassette body, and three PFCs, the inner and outer vertical targets and the dome.

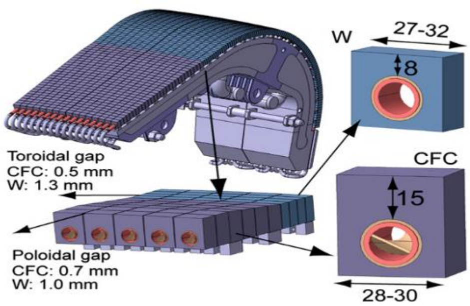

Fig.1 Monoblock of Vertical target

The inner and outer targets on each cassette are split into two similar halves, to reduce electromagnetic loads. They include a supporting structure, made of austenitic steel onto which an array of individual PFU is mounted, comprising multiple individual CFC or W monoblocks joined onto a continuous CuCrZr water cooled heat sink (via a copper interlayer) running the entire length of the poloidal target contour. Fig.2 provides details on the region of the inner target CFC–W transition, illustrating the monoblock dimensions (which differ from CFC to W and which must also vary poloidally from one part of the target to the other due to the toroidal geometry). This design allows steady state power handling of 10MW/m^2 on the CFC and around half of that for the W monoblock.

The design could handle thermo-mechanically essentially twice as much steady state power, but the limitation comes in terms of the steady state CFC surface temperature, which would lead to severe material erosion at the highest heat fluxes. The dome can withstand steady power flux densities of up to 5 MW/m^2 and a few seconds at twice that value. In total, the divertor will consist of some 3900 PFUs and around 320, 000 individual monoblock and flat tiles. The total PFC surface areas for all 54 cassettes are 14.5 m^2 (CFC) and 30.7 m^2 (W) at the inner target and 20.4 m^2 (CFC) and 45.9 m^2 (W) at the outer (the dome has a total area of 42.9 m^2 for a total W area of 119.5 m^2.

T. Hirai 2013[10]. Describes on ITER tungsten divertor design development and qualification program. The pre-detailed design phase was completed by providing design for the purposes of cost estimation by DAs and their suppliers. It is being used as an input for the neutronic, structural and stress analysis

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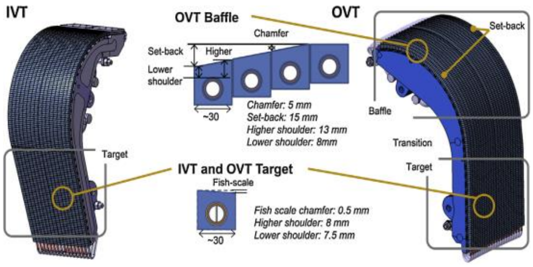

Fig.2 Full tungsten vertical targets

The preliminary design phase aims finalizations of the design of the structural component parts of the divertor assembly. The final design phase aims to deliver the design and 2D drawings of the structural and plasma-facing part (plasma-facing units, PFUs) in accordance with the results from DAs' R&D for technology development and validation show in Fig 2. A thin armour thickness is therefore favourable, although larger armour thickness is advantageous in terms of the erosion lifetime. In the present design, 8 mm has been selected, though the final choice will be determined on the basis of armour lifetime estimation and technological feasibility according to DAs' R&D. Unlike the case of carbon, which does not melt, any leading edges in a W divertor are likely to suffer melting under heavy disruptive or edge localized mode (ELM) transients on ITER. Such localized melting can pose a significant risk to subsequent plasma operation if the damaged areas are subject to high steady heat fluxes. It is thus mandatory to arrange for edge protection in all high heat flux areas. To accommodate this requirement, fish scaling will be adopted on the monoblock surfaces and at present is restricted to a simple, 2D 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), with maintaining an acceptable wetted area on each monoblock (>74%).The same fish-scale scheme is applied for both OVT and IVT areas where the divertor strike points will be located.

R.A. Pitts 2013[11]. Describes the current design status of full W divertor in ITER. The ITER organization has proposed to modify its baseline divertor strategy, replacing the current approach of beginning operations with CFC in the strike point regions followed by a switch to full-W for the nuclear phases, with a single, full-W divertor. This component would then be expected to survive some 25,000 pulses, through the non-active phases, DD operations and up to the end of the first DT campaign. Undoubtedly the biggest challenge to operation with W targets in comparison with CFC is the issue of melting, which may occur at the thermal quench of many of the several hundred unmitigated disruptions/VDEs which are expected in the ITER H/He campaigns. Here the magnitude of the problem, both in terms of the extent and nature of melting and the consequences for subsequent operation remain uncertain. Although much new R&D has recently been conducted, helping to benchmark numerical models which are the only realistic means by which to extrapolate to the ITER scale, what is still missing is a true dynamic melt experiment, conducted in a relevant tokamak environment, involving a rapid transient. A proposed experiment on JET for 2013 would seem to be the only significant chance of acquiring new information of this kind before a decision on the new divertor strategy will have to be taken. Whatever this and other studies might conclude, if a full-W divertor is installed from the beginning, it will be mandatory for ITER to develop highly reliable disruption detection, avoidance and mitigation techniques from the beginning of non-active operation.

III. CONCLUSION

ITER divertor's plasma facing component are made of CFC and W material. CFC have been selected as a armor for lower part of vertical target because of good mechanical strength, superior thermal shock resistance and high thermal conductivity. 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 and describe different W grade for armor materials. Different joining technology for Monoblock and heat sink and heat sink to Cucrzr cooling tube are define for development of plasma facing component. Dimension of monoblock are finalized and for protection of edge, fish scaling is adopted in monoblock surface. The inner and outer targets on each cassette are split into two similar halves, to reduce electromagnetic loads. Design and materials for first set of Divertor and the integration into the ITER tokamak were presented. R&D work started on implementation of full W vertical target for ITER divertor.

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