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Preliminary results on long term test of tungsten tile with ITER castellation in KSTAR

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1. Introduction

Plasma-surface interaction is one of major parts for the successful operation of ITER. ITER will have a metal wall configuration consisting of Beryllium and Tungsten. JET and ASDEX Upgrade have employed these beryllium and full tungsten first walls for ITER-relevant experiments (ITER-Like Wall, ILW) [1, 2]. Recently, Tore Supra has started the "WEST" project, the abbreviation of "W Environment in Steady-state Tokamak", which includes the change of major machine configuration into a tungsten (W) divertor machine [3].

JET experiments with these ILW configuration shows a remarkable step towards the success of fusion reactor such as 1) a better wall condition for start up and the low impurity levels, 2) the decrease of fuel retention by a factor of 10, 3) reduction of carbon impurity by a factor of 20 while maintaining good confinement [4]. Nevertheless, the behavior of tungsten armor under ELMy H-mode and long pulse operation has to be investigated.

It is planned that plasma-facing components of the main chamber and the divertor armor in ITER will be castellated for the improvement of the thermo-mechanical stability and to limit forces caused by induced currents. The results from pioneering experimental researches at TEXTOR [5] and DIII-D, a new shape of castellation is proposed (Fig.1). The thermal response analysis by using the ANSYS code demonstrates that the new shape allows operation of even misaligned shaped castellation at 20 MW/m^2 of steady-state thermal load. Furthermore, the PIC code SPICE2 [6] predicts full ion flux suppression in the gaps of shaped castellation, and Monte-Carlo 3D-Gaps code modeling [7] results at least a 20-fold decrease of beryllium content in the gaps.

For the validation of the findings, multi-machine experiment within the IEA-ITPA Joint Experiments program task DSOL 27 aiming at testing the new shape of castellation is initiated. In this paper, we report preliminary results obtained from the test of the long term tungsten tile with castellation gap structures on it.

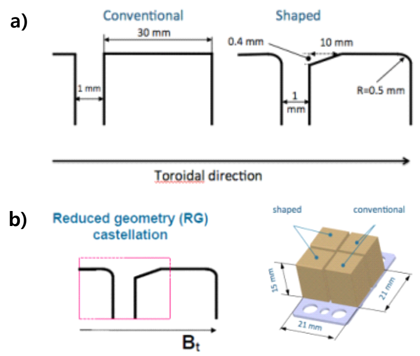

Fig. 1. (a) Conventional and optimized shaped castellation for a new W-divertor for ITER (b) reduced geometry of castellation proposed for an experiment.

2. Experimental setup

We have manufactured a ITER grade bulk tungsten tile which has the same geometrical shape as a carbon tile at the central diverter. The castellation gaps were carved by wire cutting EDM (electric discharge machining). The gap distance is 1 mm, and the depth of the gap is 15 mm. The tile is then installed at a position of lower outer strike point on central divertor at K port of KSTAR as shown in Fig. 2. The tile is exposed to various plasmas including D_2/He cleaning discharges.

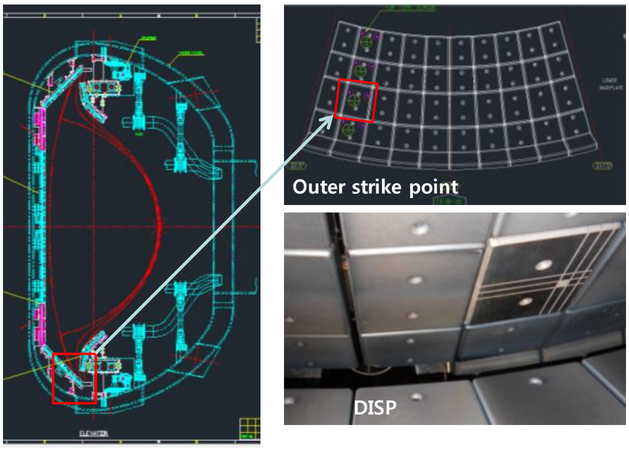

Fig. 2. Mounting and exposure location of the tile with castellation gap structure in KSTAR.

3. Results

3.1 Temperature evolution during ELMy H-mode shots

Fig. 3. shows the temperature evolution of tungsten tile during five successive ELMy H-mode shots. Three different temperature evolutions for three different tiles were depicted: tungsten tile, two carbon tiles at nearby location and one at the same poloidal location. As clearly seen, the temperature increase of the bulk tungsten tile is the largest due to the high heat capacity of tungsten. Furthermore, the "background" temperature of carbon tile at nearby position increases gradually due to the thermal radiation from the tungsten tile. This would be a critical point for KSTAR with bulk tungsten divertor/wall operation without active cooling of PFCs.

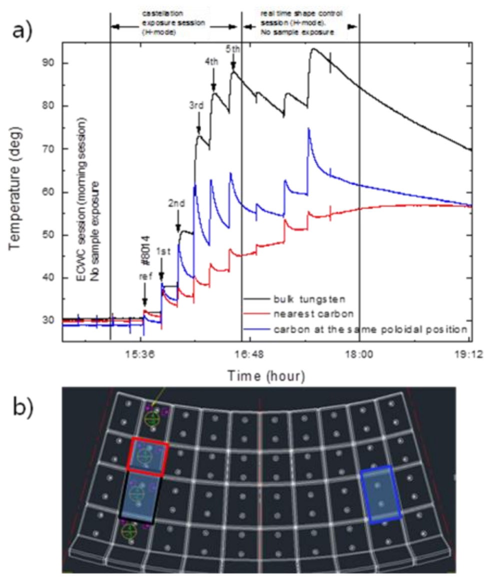

Fig. 3. (a) temperature evolution of tungsten tile during five successive ELMy H-mode shots, (b) the location of tiles.

3.2 Deposition inside the gap

Fig. 4. shows the deposition inside the gap. Since ions follow the magnetic field lines, the deposition at entrance is dominated by them. The neutrals, however, can penetrate deep inside of the castellation gaps, so that the deposition inside the gaps is dominated by charge exchange neutrals. This is consistent with similar experiments in DIII-D. Further quantifications of fuel retention, amount of carbon deposition are underway.

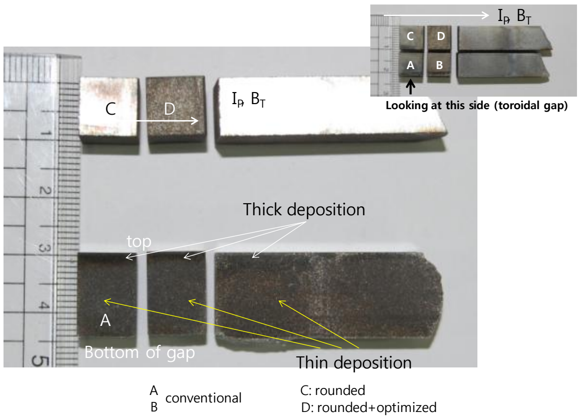

Fig. 4. Deposition inside the gap structure.

4. Summary

We have exposed and tested a bulk tungsten tile to various plasmas. The tungsten tile was examined after the 2012 campaign. The thermal behavior of the tile is measured and the deposition inside the gap shows different contribution of ions and charge exchange neutrals.

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