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Investigation of plasma wall interactions between tungsten plasma facing components and helium plasmas in the WEST tokamak

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ITER will be equipped with a tungsten divertor, which is planned to operate over a significant period of time, from the Pre-Fusion Power Operation phase into the Fusion Power Operation phase 1. As was shown in a number of linear devices as well as in tokamaks, tungsten (W) exhibits pronounced surface morphology changes under helium (He) plasma exposure, which can affect its thermal and mechanical properties (see 2 and references therein). In particular, He induced nanostructures, such as He nanobubbles or the so-called W fuzz, can be formed at the W surface, depending on plasma conditions and surface temperature. He operation is planned in the ITER Pre-Fusion Power Operation phase, in order to demonstrate H mode operation without activating the in-vessel components 1. He will also be present during the Fusion Power Operation phase, as ash from the D-T reaction. Investigating interactions between W plasma facing components and He plasmas in a tokamak environment is therefore a key point to consolidate predictions for the ITER divertor performance and lifetime.

This paper reports on a dedicated He campaign performed to investigate these issues in the full tungsten tokamak WEST. The experiment was run at the end of the first phase of operation of WEST, before dismantling the lower divertor components for post-mortem analysis. During this phase, WEST lower divertor was composed of a mix of inertial W coated plasma facing units (PFU) and ITER like actively cooled PFU. The experiment was designed in order to reach the conditions for W fuzz formation in the low field side outer strike point (OSP) area, namely: incident particle energy Einc>20 eV, He fluence >1024/m2, surface temperature > 700℃ 2. For this purpose, a robust long pulse scenario was developed in He (L mode operation with plasma current Ip = 300 kA, Lower Hybrid (LH) power = 4 MW, average density 4 1019 m-3). The plasma flattop duration was adjusted to ~30 s, in order to reach a surface temperature above the threshold for W fuzz formation over a significant area around the OSP on the inertial PFUs. Repetitive He long pulses (in the range 20-30 s) were run, cumulating ~2000 s of plasma exposure and 4.4 GJ of energy coupled to the plasma over 1 week of operation.

Local divertor plasma conditions were recorded with an array of Langmuir probes (LP), while the divertor surface temperature was monitored by embedded thermocouples (TC) / Fiber Bragg Gratings (FBG) and infrared (IR) cameras. It is shown that the 3 criteria mentioned above for W fuzz formation were met in the OSP area. Indeed, typical electron temperatures measured by the Langmuir probes were Te~20 eV at the OSP as shown in figure 1, which corresponds to an
incident He energy Einc > 100 eV. The surface temperature is estimated from IR measurements (see figure 2), assuming an emissivity ⍰ of 0.15± 0.03 at the OSP, as deduced from the TC/FBG measurements 3. As shown in figure 2, the temperature threshold for W fuzz formation is typically reached after 20 s at the OSP. The total He fluence reached at the OSP is assessed from cumulated Langmuir probe measurements to be ~4 1025 He/m2. Combining IR and LP measurements, it is shown that even in the worst case scenario (highest emissivity assumption), the threshold for W fuzz formation in terms of He fluence (~1024/m2) is reached in an area of ~1 cm width around the OSP, with surface temperature exceeding 700℃ for ~100 s.

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Electron temperature profile (1a) and cumulated He fluence (1b) measured by Langmuir probes as a function of monoblock number along the divertor. The location of the inner and the outer strike points are shown.*

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Infrared view of the lower divertor for shot #55953 (t = 27s), with the maximum temperature in the OSP area circled in black (2a) and time evolution of the maximum OSP surface temperature during the same shot (2b). Temperatures are shown for black body (⊠=1) and real OSP emissivity ⊠ = 0.15± 0.03. The threshold for W fuzz formation (700 ℃) is indicated with a dashed line.*

In-vessel inspections using the WEST Articulated Inspection Arm (AIA) were performed before and after the He campaign. They did not reveal any macroscopic signs of W surface modification in the OSP area of inertial PFU, such as blackening of the surface reported when thick W fuzz is formed. Post mortem analysis of the components is however necessary before concluding on the type of He induced nanostructures formed during the experiment, and is ongoing at the time of writing. It will be reported in the paper. These results underline that in tokamak conditions, the complex balance between erosion (in particular from impurities) /redeposition (from W eroded from the main chamber) and W fuzz formation needs to be taken into account, as was shown in other tokamak experiments 4 [5] [6]. The data obtained will be used to consolidate the experimental database supporting the modelling effort for predicting W fuzz formation and growth in ITER, such as in [7].

1 ITER Research Plan within the Staged Approach, ITR-18-003, 2018, 2 G. De Temmerman et al., Plasma Phys. Control. Fusion 60 (2018) 044018, 3 J. Gaspar et al., submitted to PSI2020, 4 S. Brezinsek et al., Nuclear Materials and Energy 12 (2017) 575–581 [5] A. Hakola et al, Nucl. Fusion 57 (2017) 066015, [6] S. Brezinsek et al., submitted to PSI2020, [7] G. De Temmerman et al, Nuclear Materials and Energy 19 (2019) 255–261256

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