Investigation of ITER – like tungsten tile mock-up with modified surface in Globus-M tokamak

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One of the key issues to be resolved for fusion energy production is testing and development of advanced plasma facing materials capable to withstand the extreme heat loads expected in fusion reactors. Tungsten is favourable as a plasma-facing material due to its high melting temperature, low physical sputtering and no chemical sputtering. Budget restrictions have forced the ITER Organization to reconsider the baseline divertor strategy, in which operations would begin with carbon (C) in the high heat flux regions, changing out to a full tungsten (W) variant before the first nuclear campaigns. Substantial cost reductions can be achieved if one of these two divertors is eliminated. The new strategy implies not only that ITER would start-up on a full-W divertor, but that this component should survive until well into the nuclear phase.

CFC as a material for the non-nuclear phase is well established and does not carry any new risks. The main issues to investigate for tungsten surface morphology changes are: blistering under H/D/T bombardment, He bubbles formation under He bombardment, material mixing (in particular Be-W alloy formation with decreased melting temperature compared to pure W) and damage (cracking/melting) under large number of pulsed heat loads (ELMs).

Most likely ITER will be confronted with thin melt layers in the order of 100 microns. If these melt layers are produced by ELMs then we may have a large number of melting and solidifying cycles. The result will be a very thin brittle layer because of re-crystallization effects. The question is if we restart the plasma on such surfaces will there be problems mainly by flaking and after how many cycles we will see flaking?. This is important in order to build up a validated construction of plasma facing components, preparing for ITER operation and providing information in support of the development of demonstration power plant - DEMO.

The goal of this work is to study the interaction of the Globus-M deuterium plasma with full scale mock-up of ITER-like tungsten tile, previously irradiated with TSEFEY-M Facility (Efremov institute) and plasma gun device (Ioffe Institute) at the power densities comparable to emerging during ELMs. To safe bulk structure properties, the tiles irradiated with the electron beam have to be actively cooled to achieve temperature close to the expected in ITER tungsten elements. The modified tungsten properties should be compared with virgin tungsten tiles after/during irradiation at the outer strike-point of Globus-M lower divertor (Fig. 1).

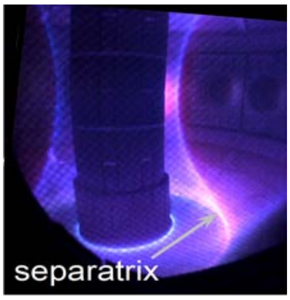

Fig. 1. Lower divertor of Globus-M tokamak

Globus-M experiments showed that in the divertor, where separatrix contacts with the tiles, deposited energy density reaches about 1 - 2 MW/m^2 (Fig. 2) [1]. The basic design characteristics of Globus-M are: major plasma radius R = 36 cm, minor plasma radius a = 24 cm, A = R/a = 1.5, toroidal magnetic field at the vessel axis B_T = 0.2 - 0.5 T, plasma current I_p= 0.1 - 0.35 MA, average plasma density n_e = (1 - 10) 10^{19} m^{-3}, pulse duration $\tau_{pulse} \leq 0.14$ s.

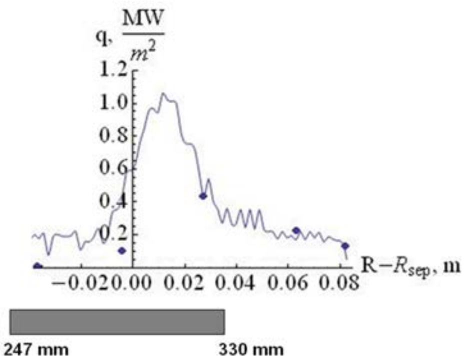

Fig. 2. Radial power density distribution on divertor plate in the OH regime

The sector of tungsten protection consists of 4 no cooled ITER – like tungsten tile mock-ups and 18 plates of 3-millimeter tungsten sheet, which will be attached to the existing stainless steel substrate. This sector will be located near intersections of the outer branch of the separatrix with the divertor tiles (Fig. 1). Radial dimension of the sector is 83 mm. ITER-like uncooled W/Cu element consisting of 20.5 × 23.9 × 10 mm W-tiles brazed to 2 mm copper substrate will be mechanically fixed to the existing stainless steel substrate. The rest of the graphite plates (18 pieces) will be replaced with tiles from 3 mm tungsten sheets fixed on the substrate.

Armor of the elements consists of the W/Cu tiles (8 mm W + 1 mm oxygen free high conductive Cu) having planar size of 23.9 × 20.5 mm, brazed to substrates by the Stemet®1108 filler ($Cu_{base}Sn_{12}In_7Ni_3$, wt. %) (Fig. 3). This filler is approved for brazing of armor of the ITER Divertor Dome [2]. All armoring tiles were cut out from the same batch of W-Cu plates (produced by casting) which will be used by Efremov Institute for production of armor for the Divertor Dome, but thickness of the copper sub-layer was machined from standard 2 mm down to 1 mm to fit the thickness of surrounding graphite armor of the Globus-M divertor. Thus, tungsten and copper used for production of the armoring tiles

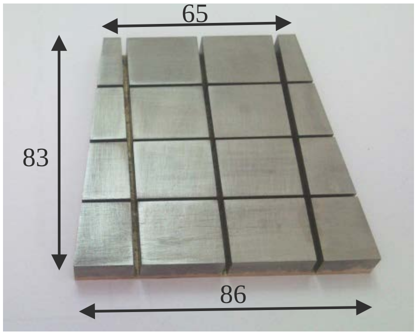

Fig.3. ITER–like tungsten tile mock-up

completely meets Material Specifications ITER_D_2EDZJ4 v 1.3 and ITER_D_2DQKNC v 1.2 respectively. Brazing of tungsten tiles onto substrates has been performed in industrial vacuum furnace "Super Series 12/24-14MD Vie" (Efremov Institute) which also will be charged armoring of plasma facing units (PFUs). Parameters of brazing where: T=900° C, t=300 s, P~5×10^{-4} Pa. Special fixture and rigs for proper armor fixation and clamping were developed and used.

First two elements were brazed onto 2 mm-thick OFHC Cu substrates. Prior to installation in Globus-M one of these elements will be irradiated by the plasma gun in at the Ioffe Institute. The second an unirradiated one will be set in the divertor as a reference.

Tiles for other two elements were brazed onto actively cooled CuCrZr substrate. Obtained in this way mock-up is planned for preliminary irradiation with surface loading parameters sufficient for multiple re-melting of the near-surface armor layer (as a simulation of ELMs). Various level of surface damaging to different armor portions may be provided. Then armor and thin CuCrZr sub-armor layer of this mock-up will be cut by means of Electrical Discharge Machining (EDM) into 3rd and 4th plasma facing elements for installing into Globus-M divertor outer strike-point region. Therefore four tungsten-armored elements with various pre-damage histories will be prepared for further exposition in the Globus-M tokamak plasma.

The tungsten tile treatment by electron beam will be made in TSEFEY-M facility [3], upgraded several years ago up to 200 kW e-beam power. The test bench has coaxial electron-beam accelerator with power of 200 kW and an accelerating voltage of 40 kV. Symmetrical two-coordinate deflection and power control of the accelerator permits to apply very wide range of heat loads to the testing object. The maximum density of the heat load can reach 1 GW/m^2. Varying these parameters one can choose the desired program for the formation of a thin melted layer on the tungsten surface. To minimize the hot tungsten surface oxidation the tests will be carried out at a pressure of $\leq 5\ 10^{-5}$ mbar. Irradiation of the samples will be made under active cooling. A few hundreds pulses providing re-melting of thin surface layer are planned. Necessary beam parameters will be selected by preliminary shooting of small-sized tuning mock-up.

We suggest the approach based on the original plasma gun device which can be used for simulation of ELM influence on the tungsten target surface. The device could be able generating plasma jet up to energy density ~ 1 MJ/m^2. The plasma gun pulse duration is ≤ 15 μs. It is clean, highly ionised pure hydrogen plasma with density (0.5 – 5) × 10^{22} m^{-3}, total number of the accelerated particles (1 – 5) × 10^{19} and flow velocity 100–200 km/s [4]. Advantages of the suggested gun are high kinetic energy and clean hydrogen plasma jet. Stored capacitor energy of the gun is about 2 kJ. It is supposed that damage factor can be in the range ε_{ELM} = 77 - 123 $MJm^{-2}s^{-1/2}$ for ELM-events in the ITER. Melting parameter for tungsten is ε_{melt} ~ 50 $MJm^{-2}s^{-1/2}$ [5]. The power density of the gun could reach 100 GW/m^2 (the damage factor ε_{gun} = 300). Power density achieves by a few orders of magnitude greater than the values achievable in tokamaks.

Fig. 4. Fragment of mock-up after irradiation with plasma gun; 100 Shots; 0.7 MJ/m^2; ε_{gun} = 230 MJ $m^{-2}s^{-1/2}$

Before installing in the tokamak the tungsten tile mock-up (Fig. 3) is cyclically irradiated by plasma jet on the plasma gun test bench. Fragment ¼ of the surface of the tile after a 100-fold exposure by plasma jet with the damage factor ε_{gun} = 230 $MJm^{-2}s^{-1/2}$ is presented in Fig. 4. Traces of melting of the tungsten surface with depth of a few micrometers are observed. Repeated irradiation of the four parts of the surface of the tile by plasma with damage factors of ε_{gun} = 80; 230 $MJm^{-2}s^{-1/2}$ is done [6].

All damaged tungsten elements have to be tested before and after installation in the Globus-M divertor. Chemical composition of surface and sub-surface layers (up to 10 μm in depth) will be studied by combined AES, XPS, and Dynamical SIMS profiling. The Multitechnique PHI-5500 AES-XPS facility (Physical Electronics, USA) and magnetic-sector SIMS IMS 7f (Cameca, France) will be used. The roughness of the surface will be studied using scanning electron microscope JSM-7001F (JEOL, Japan) and atomic-force microscope Dimension 3100 (Bruker, Germany).

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