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Progress of Qualification Testing for Full-Scale Plasma-Facing Unit Prototype of Full Tungsten ITER Divertor in Japan & Progresses on WEST Platform Construction towards First Plasmas

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A. R&Ds for starting operation with a full- tungsten (W) ITER (INB-174) divertor have been enhanced by recommendation of the ITER council since 2011. Japan Atomic Energy Agency (JAEA) as Japanese Domestic Agency (JADA) and the ITER organization (IO) have been actively working on the development and demonstration on the full-W ITER divertor under the framework of the task agreement. JAEA is in charge of technology development and demonstration for manufacturing the Outer Vertical Target (OVT) together with Japanese industries. In 2013, as the first phase of the qualification program, JAEA demonstrated the armour heat sink bonding technology with small-scale mock-ups. A high heat flux (HHF) testing for the mock-ups was carried out in the ITER divertor test facility in Efremov Institute, Russia. JAEA succeeded in demonstrating the durability of the W monoblock joint to the Cu-alloy cooling tube against the heat load of 10 MW/m2 × 5000 cycles and 20 MW/m2 × 1000 cycles which are three times higher than a requirement (300 cycles). This result provided one of sufficient materials for the decision to start with the full-W ITER divertor in the baseline. Since 2014, as the second phase, the full-scale plasma-facing unit (PFU) prototypes have been manufactured to demonstrate the scale-up manufactural technology. In this paper, JAEA reports progress of R&Ds on the full-scale PFU prototypes of a full- W ITER Divertor OVT. Under a framework of a W divertor qualification program, JAEA manufactured 7 full-scale PFUs as prototypes. Through the manufacturing, (i) all joint surfaces in four PFUs with a casting Cu interlayer successfully passed the ultrasonic testing and (ii) the surface profile in target part of PFUs stayed within a tolerance. (iii) Moreover JAEA succeeded in demonstrating a durability for the HHF testing of the repetitive heat load of 10 MW/m2 × 5000 cycles and 20 MW/m2 × 1000 cycles under close collaboration with the IO and the Efremov Institute. These results demonstrated the ability of Japanese industries to produce the PFU of full-W ITER divertor enough to meet the technical requirements.

B. The WEST platform, which is a major evolution of Tore Supra towards a steady-state tungsten diverted tokamak, is targeted at minimizing risks for ITER divertor procurement and operation. This paper presents an overview of the status and relevant technical issues for the new platform. At the time of the writing, the 4 meter diameter thick casing of the upper and lower divertor in-vessel coils have been manufactured, assembled inside the torus and accurately positioned. The in-situ winding of the water cooled copper conductor requiring about 140 brazing is underway. The complex assembly sequence as well as the resin epoxy impregnation has been simulated and validated on a full scale mock-up. The power supplies which will feed the divertor coils have been produced. Factory acceptance test have been performed and the two power supplies will be installed at Cadarache this summer. The procurement of the ITER-like divertor plasma facing units (PFUs), using the ITER tungsten monoblock technology, is ongoing in collaboration with the European and Japanese Domestic Agencies in charge of providing ITER divertor vertical targets. Prototypes are in preparation and will be tested in WEST before launching series production. Tungsten-coated technologies have been developed and qualified on various substrates to cover the other high heat flux plasma facing components. In particular, inertial graphite PFUs with improved CMSII tungsten coating (15 μm) have been qualified and manufactured in order to complement the ITER-like prototypes of the WEST lower divertor for the first phase of operation. The new CW ELM-resilient ICRH antennas are in manufacturing and the first one will be assembled in spring 2016. The existing LHCD launcher front faces have been reshaped to match the new plasma geometry. The overall diagnostic layout is finalized. Key diagnostics are being upgraded to allow for a proper monitoring of the divertor plasma facing units, the tungsten sources and transport. A new plasma control system prototyping ITER requirements is being implemented. WEST is presently scheduled to be operational in late 2016.

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Primary author: Dr SEKI, Yohji (National Institutes for Quantum and Radiological Science and Technology)

Co-author: Dr BUCALOSSI, Jerome (CEA)

Presenter: Dr SEKI, Yohji (National Institutes for Quantum and Radiological Science and Technology)

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