Session: **Session 9: Tungsten, tungsten alloys, and advanced steels and Technology and qualification of plasma-facing components**

I-18

Design development of inertially cooled tungsten first wall for ITER Start of Research Operation

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The revised ITER Research Plan accompanying the 2024 re-baseline includes a new ”Start of Research Operation” (SRO) phase, replacing the First Plasma and Pre-Fusion Power Operation phases of the 2016 Baseline [1]. A key feature of the new baseline is the switch from beryllium to tungsten (W) first wall (FW) armour [2]. To avoid the risk of damaging the complex, water-cooled panels while learning to avoid/mitigate disruptions and runaway electrons, an inertially cooled, temporary first wall (TFW) will be fitted to the blanket shield blocks for SRO only. This contribution focuses on the design of this TFW, which is constrained by the demands of the two principal SRO scenario objectives: deuterium H-mode plasmas at half plasma current and toroidal field (7.5 MA, 2.65 T) and hydrogen L-mode discharges at full field and current (15 MA, 5.3 T).

Mimicking the final, actively cooled FW (for DT operation) in material, geometry, and interfaces, two design concepts are under development, to be deployed at different locations on the FW dictated by local plasma thermal loads: a heavy design with bulk W mechanically connected to stainless steel (SS) support, and a light design with W coating on SS. The TFW panel is segmented into W bulk or coated tiles bolted to a steel frame with partial slitting, or with fully separate fingers to ensure proper management of thermal loads and electromagnetic stresses.

Given the limited industrial experience with W coatings on SS in tokamak environments, a detailed coating qualification plan is underway. This includes high heat flux testing using an electron beam facility, and exposure of coated samples across reasonably large surface areas in a tokamak with ITER-relevant plasma loading conditions. The hydrogen transport code MHIMS [3] is employed to analyze outgassing from the inertially cooled TFW SS components throughout the SRO campaign, using average temperature evolutions of TFW panel designs obtained from ANSYS, with heat loads for the 15 MA/5.3 T L-mode discharge. Results indicate that the inter-pulse hydrogen pressure is affected by outgassing and provides insights for optimizing the SRO pulse plan to mitigate its effect.

[1] P. Barabaschi et al., 33rd Symposium on Fusion Technology (SOFT) (2024)
[2] R. A. Pitts et al., Nuclear Materials and Energy 42 (2025) 101854
[3] E. Hodille et al., J. Nucl. Mater. 467 (2015) 424