Advancing Plasma-Facing Materials for Fusion Pilot Plants at DIII-D TEC

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A total of 17 novel plasma-facing materials (PFMs) from 8 institutions have been successfully tested at the DIII-D National Fusion Facility as part of an ongoing, two-year experimental thrust. This effort coordinates exposure and down-selection of promising PFMs through testing at DIII-D in an integrated, reactor-like environment. Here we present an overview of the 1st year experimental results, which were conducted across 3.5 run days in 2024 utilizing the Divertor Materials Evaluation System (DiMES). Repeatable reference discharges were developed to enable direct comparisons between experiments, including a new strike-point rastering/sweeping scenario to provide more uniform heat/particle flux across the DiMES surface area. This standard ELMing H-mode scenario had incident heat fluxes during and between ELMs of $q_{\perp,inter-ELM} \approx 2\ MW/m^2$, $q_{\perp,intra-ELM} \approx 6\ MW/m^2$, at $f_{ELM} \approx 40\ Hz$. Various DiMES and sample geometries were used to achieve FPP relevant fluxes, including samples angled 10-15° towards the incident plasma flux (increasing heat/particle flux by factors of 7-10) and heated DiMES up to 500 °C.

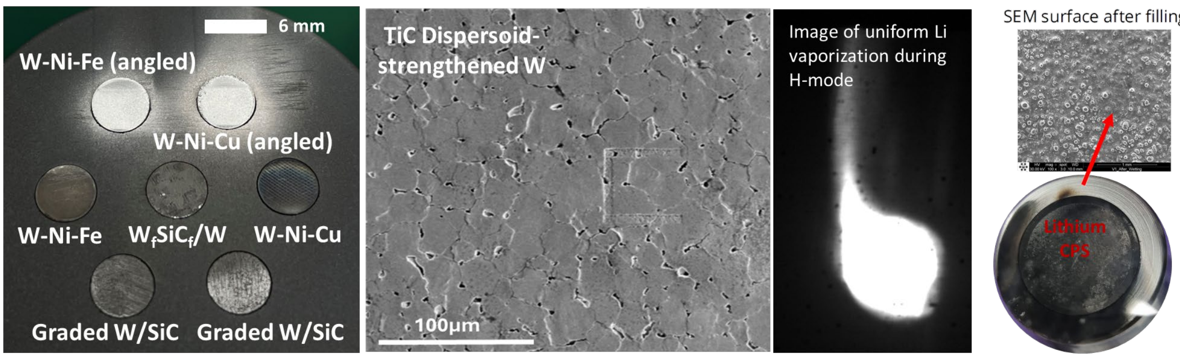

Figure 1 – Example post-exposure images of various PFMs tested under H-mode with ELMs. Left shows assembled DiMES holder with 6mm button samples (flush and 10° angled). Center shows microscopy of a DSW sample with dispersoid ejection. Right shows a post-exposure image of the Li CPS surface as well as in-situ diagnostic images of uniform Li vapor emission.

Dispersoid-strengthened tungsten samples (DSW) [1] with 1 wt% transition-metal carbides (TaC, TiC, ZrC) were tested under H-mode to compare surface stability and crack resistance with and without near-surface helium implantation (Figure 1). Ambient and heated (~500 °C) DiMES samples compare crack resistance for starting temperatures closer to W's ductile-to-brittle transition temperature (DBTT). No melting or cracking is observed on flush samples, regardless of composition. TEM analysis of flush He-implanted samples revealed nanobubble formation 10's of nm below the surface. Significant cracking and evidence of dispersoid ejection was observed for all varieties (as-received, helium implanted, and heated) of the ZrC and TaC DSW using a 10° angled geometry with $q_{\perp,inter-ELM} \approx 15\ MW/m^2$, $q_{\perp,intra-ELM} \approx 42\ MW/m^2$. The TiC DSW exhibited superior macroscopic surface integrity with only micro-crack formation, providing a clear down-selection.

The liquid Li experiment represents the first tokamak experiment assessing Li enclosed in an additively manufactured W capillary porous structure (CPS) under strike-point plasmas. The experiment evaluated lithium transport, redeposition, and the effectiveness of the CPS in preventing droplet formation and lithium leakage. The CPS was filled with 0.5g of Li and exposed in the lower divertor of DIII-D across a range of H-mode power conditions, resulting in $q_{\perp,inter-ELM} \approx 2-3\ MW/m^2$ and a Li atomic ejection rate up to 10^{21} s^{-1}. Operating at 350°C starting temperature ($T_{melt,Li} \sim 200$ °C) resulted in total suppression of Li droplets, with a uniform emission of Li vapor observed via a fast filtered camera as the Li dispersed toroidally. Li was not detected in the core, and minimal local splashing of the liquid Li layer was observed post-experiment.

Ultra High-Temperature Ceramics (UHTCs) TiB_2 and ZrB_2, candidate materials for first-wall and sacrificial limiter components [2], were exposed to L-mode conditions to characterize erosion and hydrogenic retention. Initial SEM results show minimal degradation, with more significant preferential erosion observed in the TiB_2 variety. Nuclear-grade SiC fiber composites (SiC_f/SiC) were exposed alongside CVD SiC (bulk and thin film) to the ELMing H-mode scenario. The SiC_f/SiC samples showed microscopic cracking and arcing on exposed edges, but all other SiC samples showed no signs of macroscopic or microscopic degradation.

W and SiC coatings deposited onto flush graphite substrates through atmospheric plasma spraying [3] were tested under the ELMing H-mode scenario as a potential in-situ wall armor replenishment technique. Both coatings survived with no macroscopic delamination. The SiC coating demonstrated granular material ejection during ELMs, particularly around the sample leading edges where 300μm buildup of amorphous SiC was measured. Surface roughness S_a at the strike-point locations for W (50μm) and SiC (34μm) was about double that of the central, lower heat flux regions. A variety of additional W-based alloys were also stress tested in H-mode. These include Ni-based W Heavy Alloys (WHAs) [4], W_fSiC_f/W composites, and functionally-graded W/SiC. Both varieties of 10° angled WHAs displayed edge melting and cracking, along with heavy local redeposition that then led to arcing (see Fig 1). All flush samples of WHAs, W_fSiC_f/W, and W/SiC showed no macroscopic damage.

This first year of tests quantified erosion rates, fuel retention, and material integrity of candidate PFMs under L- and H-mode conditions, leveraging DIII-D's extensive diagnostic and divertor exposure capabilities. Year 1 was designed to expose as many candidate materials as possible under similar FPP-relevant plasma conditions, exploring variations relevant to each material class (manufacturing method, alloy percentages, crystallinity, fiber density, dispersoid composition/size, coating technique, etc.). The thrust integrated materials from public and private institutions, with multiple private companies scheduled for experiments in Year 2. Further DIII-D testing to down-select materials and quantify key enabling properties will advance the TRL of the most promising PFMs, enhancing US and international capabilities to develop a viable Fusion Pilot Plant.

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