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Burning Plasma Relevant Control Development: Advanced Magnetic Divertor Configurations, Divertor Detachment and Burn Control

PPC

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The world's first real-time Snowflake Divertor (SFD) geometry identification and feedback-control system was successfully implemented on DIII-D in order to obtain and stabilize various SFD configurations and integrate them with scenarios such as the Advanced Tokamak (AT) scenario. An integrated detachment control system was developed at DIII-D that calculates and regulates the detachment front while minimizing the effect of the detachment on the core by fixing the core density independent of the detachment control. Finally, in a new approach to burn control, it was demonstrated that the simulated fusion power could be controlled by the application of non-axisymmetric fields using in-vessel coils.

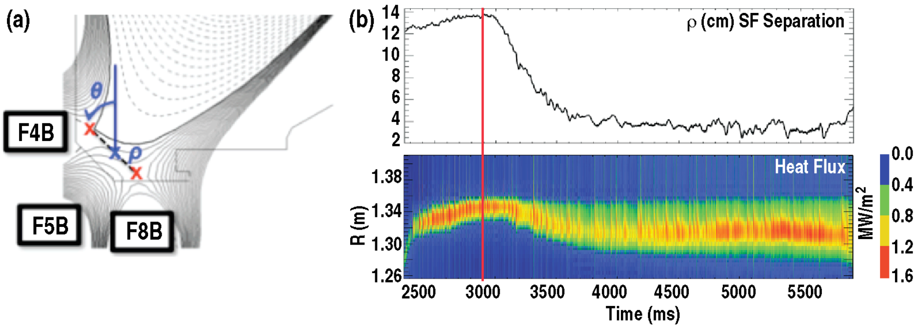

Fig. 1. (a) PF coils used in SFD control and the definition of the SFD configuration parameters. (b) Plasma controlled to almost exact SFD. SFD control starts at 3000 ms.

The new SFD control enables precise manipulation of SFD geometry, which greatly reduces peak heat flux through its high poloidal flux expansion, a large plasma-wetted area and extra strike points. SFD geometry requires a second-order poloidal field null created by bringing together two X-points. The feedback system uses a fast real-time snowflake identification algo-rithm based on local expansion of the Grad-Shafranov equation to locate the two X-points. Then, Poloidal Field (PF) coil currents are modified by the algorithm to obtain the desired SFD configuration [Fig. 1(a)].

This control enabled formation of SFD with varying σ, the distance between the X-points normalized to the minor radius, ranging from 0.08 to 0.5 in various scenarios. An example of an almost exact SFD obtained with this control is shown in Fig. 1(b), where the SFD control is turned on at 3 seconds (red line) and ρ is controlled to a few cm (approximately the grid resolution of rt-EFIT) until the end of the shot. Broadening of the heat flux profile at the outer strike point is observed as the perfect SFD is approached. SFD was successfully integrated with a DIII-D Advanced Tokamak (AT) scenario with β_N=3.0 and $H_{98(y,2)}\cong1.35$ (Fig. 2.), achieving a 2.5 reduction in peak heat flux at the outer target for many energy confinement times (2–3 s) without any adverse effect on core plasma confinement.

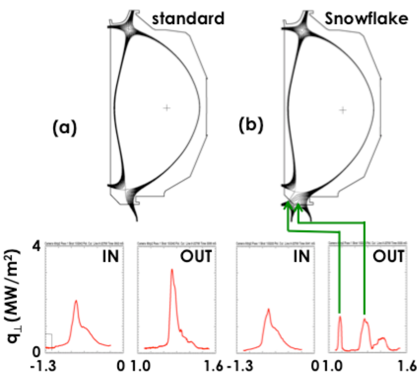

Fig. 2. Heat flux profile at the inner and outer strikepoint for (a) a standard divertor double null AT and (b) a SFD (-) double null AT.

In further work, an integrated detachment control system was developed at DIII-D that locates and regulates the detachment front while minimizing the effect of the detachment on the core by fixing the core density independent of the detachment control. ITER will require precise detachment control to manage divertor target heat loads without causing a MARFE thermal instability. A new feedback control system was successfully implemented on DIII-D to regulate the degree of divertor detachment and study this operating mode, and was successfully used to control and stabilize the detachment front at a given location between the strike point and the X-point throughout the shot. The new system was used to test the feasibility of the envisioned ITER partial-detachment operation using divertor Thomson measurements on DIII-D. (ITER will have a divertor Thomson with the diagnostic capability to measure as low as 1 eV.) This control regulates both deuterium fuel and impurity gas injection rates via a valve close to the strike point, which maintains the detachment front (where the plasma temperature drops to less than a few eV) at a pre-set distance from the divertor target using the real-time electron temperature measurements. The far-away valve keeps the core density stationary using interferometry measurements. A comparison of two shots with and without detachment control in L-mode is shown in Fig. 3. This control stabilized the detachment front fixed at the mid distance between the strike point and the X-point throughout the shot (Fig. 4). This partial detachment reduces the radiation peak from the strike point and spreads it across the detached area.

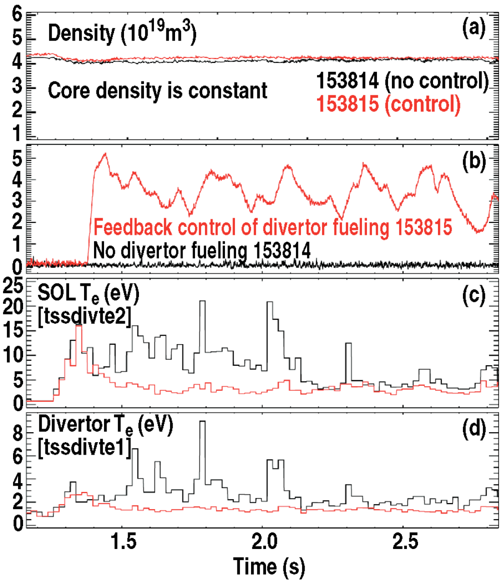

Fig. 3. Data showing feedback control of divertor detachment. Detachment feedback control on (red), detachment control off (black) (no divertor fueling). (a) Line-average core density. (b) Gas fueling rate. (c) SOL electron temperature at ~20 cm above divertor. (d) Electron temperature just above divertor plate.

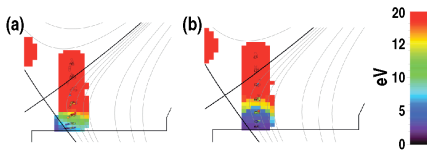

Fig. 4. 2D Projected divertor Thomson temperature measurements for DIII-D: (a) shot without detachment control (#153814) shows no detachment, (b) shot with partial-detachment control (#153816) achieves detached cold front region shown in purple and blue.

For the first time, simulated fusion power at DIII-D was controlled using non-axisymmetric magnetic field (n=3) in-vessel coils. For negligible auxiliary heating, the fusion power scales approximately as $H_{IPB98(y,2)}^{5.3}$, thus an actuator that modifies the confinement time can be used to adjust the fusion power with modest power requirements. To test the feasibility of this approach in DIII-D experiments, alpha-heating excursions were simulated with transient increases in neutral beam power. The burn control algorithm compensated the increased heating power by increasing the I-coil current, which reduced the energy confinement time to keep the stored energy constant (Fig. 5). These experiments demonstrated that for ITER and future reactors, non-axisymmetric magnetic fields are feasible as burn control tools

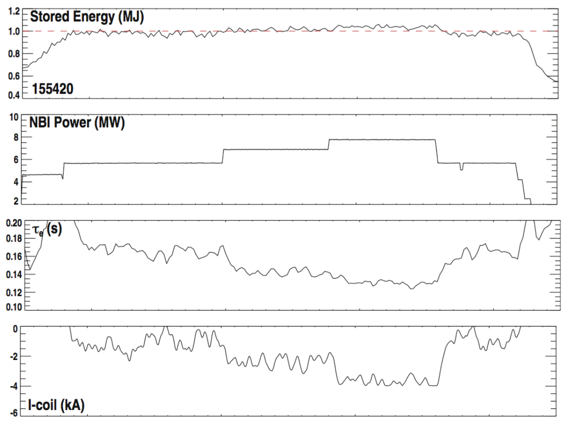

Fig. 5. Simulated burn control with in-vessel coils: The I-coil feedback loop compensated for the increased neutral beam power to control the stored energy.

Thus, DIII-D has demonstrated that with good control critical heat flux reduction strategies can be effectively achieved.

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