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

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Abstract. Novel control schemes have been implemented at DIII-D to test and optimize heat-handling capabilities and burn regulation for advanced tokamaks. The world's first real-time Snowflake Divertor (SFD) detection and control system was implemented on DIII-D in order to produce and stabilize this configuration. The algorithm calculates the position of the two null-points in real-time by locally expanding the Grad-Shafranov equation and controls shaping coil currents to achieve and stabilize various snowflake configurations. SFD experiments achieved a 2.5 times increase in the flux expansion and a 2.5 reduction in peak heat flux for many energy confinement times without any adverse effect to the core plasma such as confinement in advanced tokamak scenario with β_N=3.0 and H98(y,2)$\cong$1.35. In addition, a new detachment and radiation control algorithm was implemented at DIII-D. The algorithm uses divertor temperature measurements from real-time Thomson diagnostics and a line ratio measurement to compute the detachment level, and a real-time bolometer diagnostic to determine core and divertor radiation. The new system was used to test the feasibility of the envisioned ITER partial-detachment operation using divertor Thomson measurements on DIII-D. A dedicated partial detachment control was also implemented to control the detachment front location using divertor temperature measurements from real-time Thomson diagnostics while minimizing the effect of the detachment on the core by fixing the core density independent of the detachment control. The control stabilized the detachment front fixed at the user-defined location between the strike point and the X-point throughout the shot. Finally, as a new approach to burn control, it was demonstrated that experimentally simulated fusion power could be controlled by the application of non-axisymmetric fields using in-vessel coils. 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 and kept the stored energy (proxy for fusion power) constant.

1. Introduction

Heat flux management of the divertor target is a major issue for ITER and future fusion reactors, which will need robust control. In order to manage the heat flux at the divertor, the control of the Snowflake Divertor (SFD), an alternative to the standard divertor configurations, and the control of divertor detachment were successfully demonstrated at DIII-D. 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.

Results of the development and implementation of these two heat flux reduction control methods are presented in the first two sections of the paper.

A major burning plasma relevant issue is the regulation of the burn rate (fusion reaction rate) of a tokamak reactor. As 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 at DIII-D. The control development and results are presented in the final section.

1. Advanced Magnetic Divertor Configurations Control

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 [1]. The topological instability of the SFD configuration motivated implementation of a control system to sustain the SFD. The feedback system uses a fast real-time snowflake identification algorithm based on local expansion of the Grad-Shafranov equation to locate the two X-points [2,3]. Then, poloidal field (PF) coil currents are modified by the algorithm to obtain the desired SFD configuration [Fig. 1(a)].

Formation of SFD was enabled by this control with varying σ, the distance between the X-points normalized to the minor radius, ranging from 0.08 to 0.5 in various scenarios. Figure 1(b) shows an almost exact SFD obtained with this control. The SFD control is turned on at 3 seconds (red line) and is controlled to a few cm (approximately the grid resolution of the real-time reconstruction) until the end of the shot. Broadening of the heat flux profile at the outer strike point is observed as the SFD is approached. Figure 2 shows the shape comparison of the DIII-D AT scenario with the standard divertor and the successful integration of SFD. Figures 3 and 4 show the heat flux comparison for these two different divertor configurations. A 2.5x reduction in peak heat flux was obtained at the outer target for many energy confinement times (2–3 s). This was achieved without adversely effecting the core plasma and while keeping β_N = 3.0 and $H_{98(y,2)}$≅1.35.

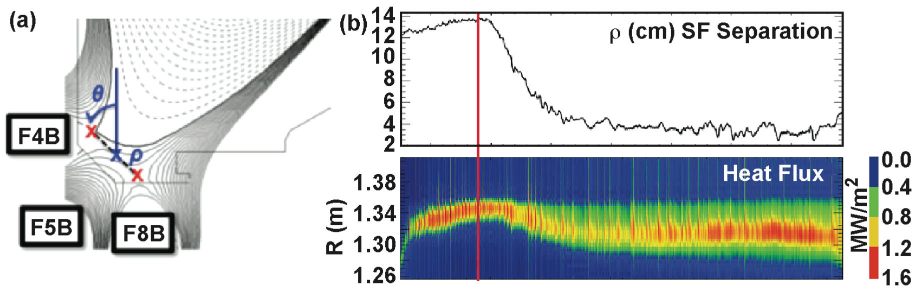

Fig. 1. (a) PF coils used in SD control and the definition of SD configuration parameters. (b) Plasma controlled to almost exact SD. The SD control starts at 3000 ms (shown with red line). The lower panel shows the heat flux at the outer strike point (155478) [3].

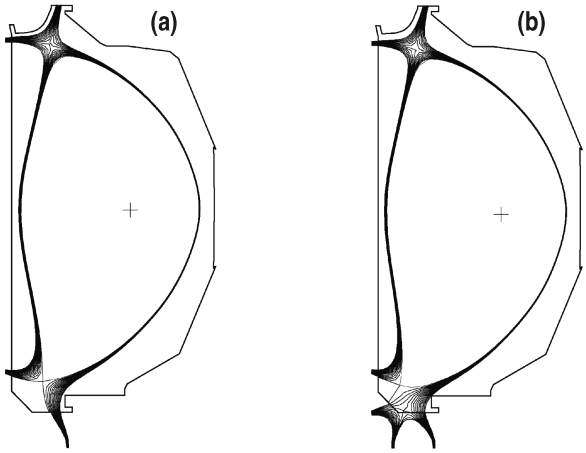

Fig. 2. The shape of the AT scenario with (a) the standard divertor double null and (b) the SFD (-) double null.

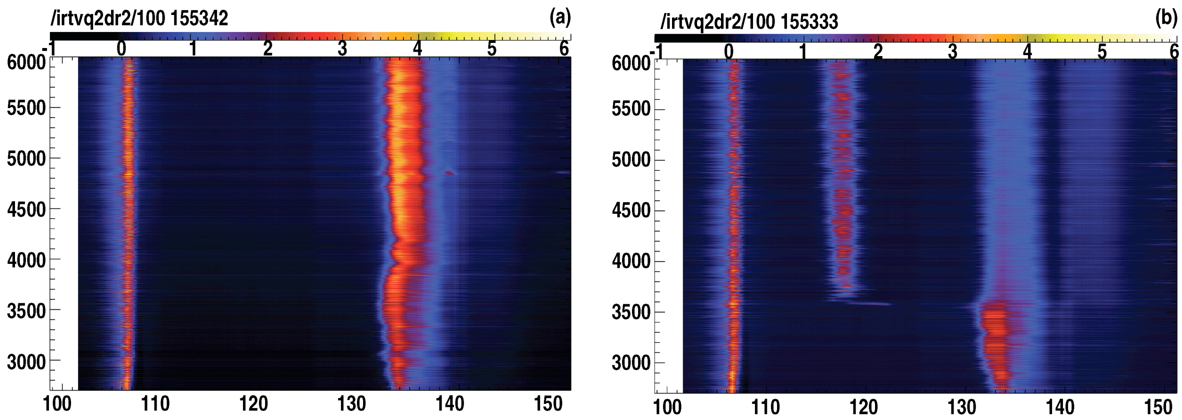

Fig. 3. The heat flux profile measured by the infrared television (IRTV) camera at the inner and outer strike points in time for (a) the standard divertor double null AT and (b) the SFD (-) double null AT.

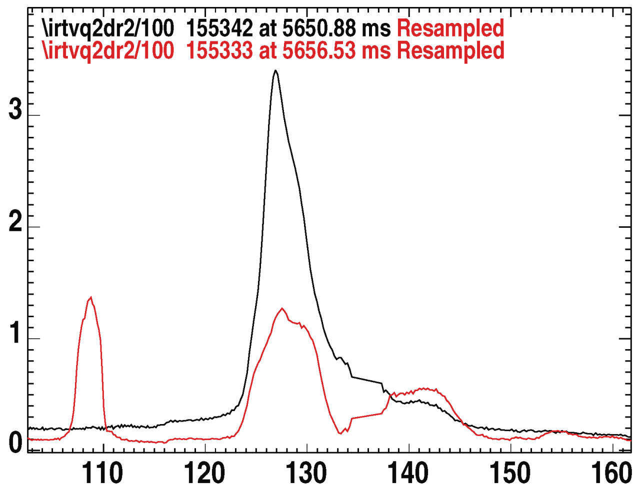

Fig. 4. The heat flux profile comparisons of the outer strike points for the standard divertor double null AT and the SFD double null AT.

3. Divertor Detachment Control

In further work, an integrated detachment control system was developed at DIII-D to locate and regulate the detachment front while at the same time keeping the core density constant in order to minimize the effect of the detachment on the core. ITER will require precise detachment control to manage divertor target heat loads without causing a MARFE thermal instability [4]. A new feedback control system was successfully implemented on DIII-D to regulate the degree of divertor detachment, where the plasma temperature drops to less than a few eV, and study this operating mode. It 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.

A divertor Thomson system with 1 eV resolution will be available on ITER [5]. We used the new detachment control system on DIII-D with divertor Thomson measurements as diagnostic inputs to test the feasibility of the envisioned ITER partial-detachment operation [3].

This system regulates both deuterium and impurity (Ne and Ar) injection rates via a primary valve near the strike point and a deuterium fueling rate via secondary valve further away. The aim of the primary control loop is to keep the detachment front at a user-specified distance from the divertor target plate using the real-time electron temperature measurements. The second control loop holds the core density stationary using interferometry measurements as diagnostics. Two consecutive shots with and without detachment control in L-mode are displayed in Fig. 5. The figure shows the temporal effect of the control. When the partial-detachment control is turned on at 1.25 seconds, the divertor temperature reduces to 1-2 eV and is kept stationary throughout the shot, while the core density is controlled to be the same as the no-detachment control case. The spatial effect of the control is illustrated in Fig. 6, which shows 2D Thomson temperature projection. The location of the detachment front is fixed at the mid-distance between the strike point and the X-point during the control phase. The partial detachment also changes the radiation profile, spreading the radiation across the detached area and thus reducing the peak radiation from the strike point location. This control facilitates the investigation under reproducible conditions of the plasma-surface interactions and the physics of detachment onset.

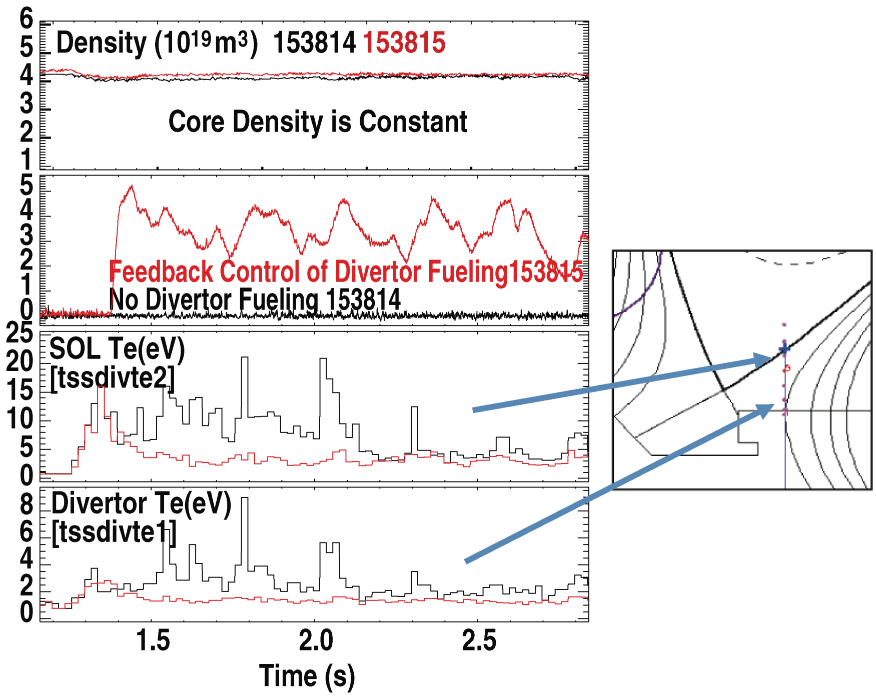

Fig. 5. Data showing feedback control of divertor detachment. Detachment feedback control on (red) and detachment control off (no divertor fueling) (black). (a) Line average core density, (b) gas fueling rate, (c) SOL electron temperature at ~20 cm above divertor and (d) electron temperature just above divertor plate. The inset shows the divertor geometry.

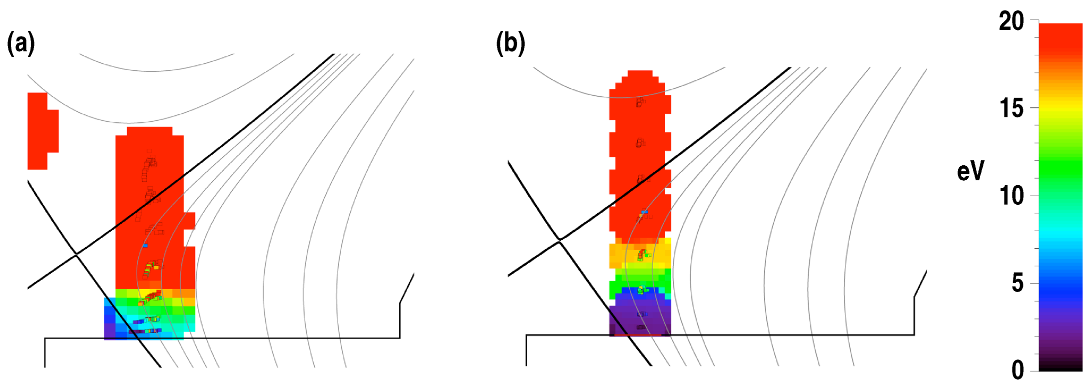

Fig. 6. Divertor Thomson temperature measurements with a 2D projection for DIII-D: (a) detachment is not observed in no control discharge (153814), (b) a detached cold front region shown in purple and blue shot is observed in discharge with partial-detachment control (153815).

4. Burn Control Using Non-Axisymmetric Coils

Fusion heat management in a reactor starts with the power obtained from the fusion reaction of the deuterium and tritium particles ("burn") at the core. It is critical to control the burn rate for the ITER project against power surges that may occur during the burn phase due to variation in the plasma and more specifically during burn entry and exit where the plasma conditions are unpredictable. Approaches to burn control using auxiliary heating control, density control and impurity injection have been proposed [6–8]. However, all these approaches have down sides. Burn control with auxiliary heating requires additional and costly heating power capability; density control is limited by the Greenwald density limit and the need to maintain a detached divertor; impurity injection may affect the plasma too slowly for effective control. We have developed a new feedback algorithm by compensating the increased heating power withy increasing the I-coil current, which reduced the energy confinement time to keep the stored energy constant (Fig. 7). Here, the excursions in the plasma stored energy, which can be used as a surrogate for the fusion power, were simulated with transient increases in neutral beam power. These experiments demonstrated that for ITER and future reactors, non-axisymmetric magnetic fields are feasible as burn control tools.

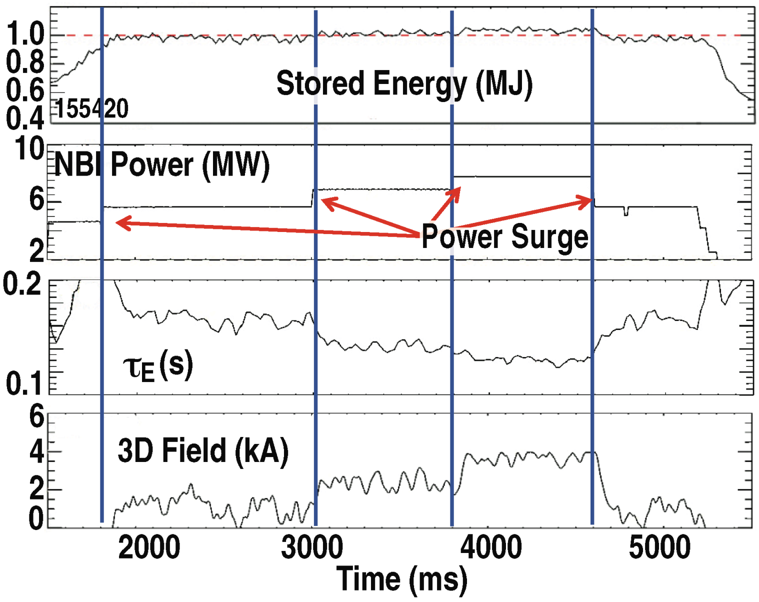

Fig. 7. Simulated burn control with in-vessel coils: the I-coil feedback loop compensated for the increased neutral beam power to control the stored energy.

In addition to regulating the simulated burn, pedestal density measurements were obtained using real-time analysis of Thomson scattering data and were used to feedback on the D_2 gas using a proportional-integral-derivative (PID) controller to keep the pedestal density constant. This control was able to counteract the density “pumpout” observed at DIII-D due to resonant field perturbations.

An advantage of the stored energy control using I-coils relative to the regular approach of using neutral beam injection (NBI) as actuators is that the standard deviation of the stored energy is 50% lower in the control loop time scale. This is due to the continuous control capability of the I-coil current between 0 and 4.2 kA versus the quanta of 2 MW in the NBI power control.

5. Conclusions

Control design is a critical component of the heat management solutions that must be addressed for the next generation of fusion reactors. At DIII-D, with optimal control design, we have shown that the magnitude of the heat management problem can be reduced substantially. Fusion heat management starts with the power obtained from the fusion reaction of the deuterium and tritium particles ("burn") at the core. We have demonstrated that, for ITER, the use of the non-axisymmetric coils as a possibly effective burn control approach, which minimizes the need for additional heating resources and impurity injection. Detachment of the plasma from divertor targets is necessary in order to avoid material erosion. We have also shown that detachment control on DIII-D can be achieved by employing a partial-detachment control system that uses divertor Thomson measurements. Finally, material technology constrains the maximum heat flux that the divertor plates can handle. We have demonstrated that, for future fusion reactors, the peak heat flux to the divertor can be reduced using advanced magnetic divertor control strategies. Control should be considered as an indispensible component of future fusion reactor designs, which may allow alternative solutions and lower costs.

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