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DEVELOPMENT IN THE DIII-D TOKAMAK OF ADVANCED OPERATING SCENARIOS AND ASSOCIATED CONTROL TECHNIQUES FOR ITER

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The DIII-D research program is focused on providing solutions to issues critical to the future success of ITER, both in achieving its basic mission goal of Q=10 operation and in enhancing the ITER physics program through development of operating regimes capable of sustaining higher Q values. Significant progress has been made in the ability to control key plasma features and using such control to expand the operational limits of stationary and steady-state tokamak operation. Recent experiments have demonstrated the capability to suppress the key plasma instabilities of concern for ITER (ELMs [Fig. 1(a)], neoclassical tearing modes (NTMs) [Fig. 1(b)], and resistive wall modes (RWMs) [Fig. 1(c)]) by external means, techniques for mitigating the effects of disruptions, and control of the current profile evolution. The use of these techniques has allowed an expansion of the envelope of viable, stationary tokamak operation, highlighted by the demonstration of sustained (~2 s) operation of β_N~4 (50% above the no-wall stability limit) as well as fully noninductive operation with β_T~3.5%. This development is supported by a vigorous basic physics program, which also addresses several key ITER issues. These include edge carbon transport and tritium co-deposition on plasma facing surfaces, fast-ion instabilities and their effects on the fast ion population, and identification of the underlying mechanisms responsible for transport in a tokamak plasma. Highlights of recent research in these areas are detailed below.

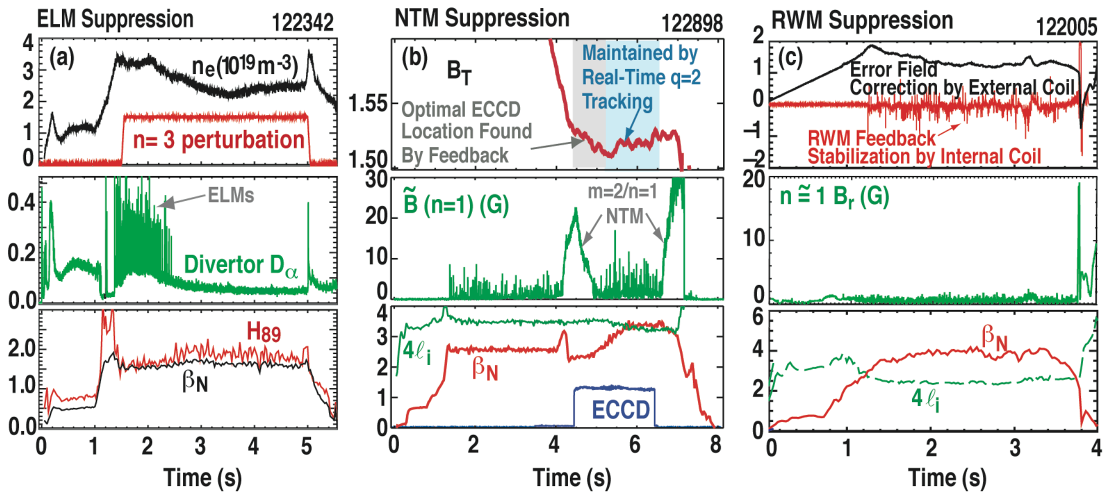

Fig. 1. Example discharges from DIII-D demonstrating the suppression of (a) ELMs using n=3 RMPs, (b) NTMs by ECCD at the q=2 surface, and (c) RWMs by feedback control with non-axisymmetric coil sets.

The erosion of plasma facing surfaces due to repetitive ELM heat pulses is a critical issue for the divertor lifetime in ITER. A possible solution to this issue has emerged from recent experiments on DIII-D. Applying edge resonant magnetic perturbations (RMPs) with n=3 symmetry completely eliminates ELMs in ITER-like edge conditions [Fig. 1(a)]. Detailed analysis has shown that the observed ELM suppression when using RMPs and in passively ELM-free (QH-mode) plasmas results from changes in edge particle transport such that the operational point is slightly below the peeling-ballooning stability limit. In both cases, this is accomplished with negligible reduction in energy confinement.

Because of the potential risk of damage resulting from a full-current disruption in ITER, methods are required to prevent disruptions and to mitigate their effects. The m=2/n=1 NTM

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is expected to be the most significant mode leading to disruption in ITER. Experiments on DIII-D show that highly localized electron cyclotron current drive (ECCD) at the q=2 surface can stabilize the m=2/n=1 NTM. By actively controlling the alignment of the ECCD with the q=2 surface using real-time equilibrium reconstruction, the plasma pressure has been increased to the n=1 no-wall stability limit (β_N~3.2) [Fig. 1(b)]. Disruption mitigation studies on DIII-D are focused on the use of high-pressure impurity gas injection. These studies indicate that the thermal quench is primarily due to impurity mixing and heat transport out of the core caused by a global MHD event as the jet-initiated cold front reaches the q=2 surface. Preliminary experiments using a 96% H_2/4% Ar gas jet suggests it may be possible to tailor the neutral delivery time scale by entraining the radiating species in a lighter background gas.

The development of the physics basis for the choice of plasma facing materials in ITER is ongoing. The primary drawback for carbon is tritium co-deposition with carbon. Material exposure studies in DIII-D that show a sizeable reduction in D co-deposition in exposed samples when heated to $\gtrsim$100°C suggest that such tritium retention may be controllable. These results, combined with results from carbon-13 transport studies that show preferential deposition of carbon in the inner divertor, suggest that minimal heating of the inner divertor substrate in ITER may be sufficient to reduce tritium retention to an acceptable level.

Through the development of integrated scenarios with improved stability and transport properties, DIII-D is providing the basis for an enhanced physics program on ITER. In recent years, stationary operating scenarios at q_{95}=3 (advanced inductive regime) and q_{95}>4 (hybrid regime) have been developed on DIII-D that project to higher fusion gain than the ITER baseline. An expansion of the operating space of these regimes has allowed studies of the ρ^* scaling of energy transport, which show the local core transport to have gyroBohm scaling. Recently developed theories for maintaining $q_0 \gtrsim 1$ (including NTM/ELM coupling, NTM reduction of neutral beam current drive, and NTM generation of kinetic Alfvén wave) are now being tested. Finally, the compatibility of hybrid operation with high levels of radiation (P_{rad}/P_{input}=60%) and impurity enrichment (η_{Ar}>30) has recently been demonstrated.

The credibility of high β, steady-state, tokamak operation has been further strengthened by recent DIII-D experiments in which the plasma pressure has been maintained 50% above the no-wall stability limit (with β_N~4) for ~2 s [Fig. 1(c)]. These discharges are characterized by a broad current profile with ρ_{qmin}~0.6 and an internal transport barrier in the ion thermal channel. The non-axisymmetric coil sets on DIII-D were utilized to both enhance RWM rotational stabilization (by reducing the intrinsic error field via the external coil set) and stabilize the RWM directly (by feedback stabilization via the internal coil set). Along with other DIII-D studies of basic RWM physics (in particular the critical rotation velocity for rotational stabilization and the RWM dissipation mechanism), this research is providing the basis for a design for RWM stabilization on ITER.

Separate experiments have demonstrated fully noninductive operation with β_T>3.5% utilizing off-axis ECCD to sustain a weak negative central shear current profile. Simulations based on these results and using transport and current drive models benchmarked against DIII-D data indicate that Q=5, steady-state operation should be possible with ITER's initial heating and current drive complement. Real-time current profile control using real-time equilibrium analysis has been developed. Studies have shown that the current profile can be reliably controlled using either ECH or neutral beam heating during the current ramp phase.

Diagnostic advances on DIII-D are enabling the benchmarking of state-of-the-art computational tools, which are critical for predictive simulations of ITER plasmas. Of particular note is a suite of fluctuation diagnostics that provide measurements of fluctuations over a wide range of scale lengths ($0<k<40$ cm^{-1}). Comparisons with gyrokinetic transport codes such as GYRO have begun. One interesting result is the GYRO prediction of equilibrium zonal flow generation near rational surfaces, which may provide an explanation for the observed transport reduction near such surfaces. The fluctuation diagnostic set, along with a prototype fast ion profile diagnostic, have enhanced the study of fast ion instabilities in DIII-D, allowing identification and characterization not possible with external measurements.

Experiments in the coming year will take advantage of extensive upgrades including the reorientation of a neutral beam to allow counter- and low-rotation plasmas, a new lower divertor for density control in double null plasmas, and increased EC power. It is anticipated that these upgrades will increase flexibility for control and optimization of plasmas in DIII-D.