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DIII-D RESEARCH TO ADDRESS KEY CHALLENGES FOR ITER AND FUSION ENERGY

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GENERAL ATOMICS

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DIII-D Research to Address Key Challenges for ITER and Fusion Energy **OV**

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The DIII-D tokamak has addressed critical challenges in preparation for ITER and the next generation of fusion devices. The robustness and performance of ITER scenarios was expanded with edge localized mode (ELM) suppression demonstrated using a reduced coil set (Fig. 1), disruption heat load mitigation, physics based error field correction, and extension of promising scenarios such as QH mode to high Greenwald fraction. The path to steady state has been advanced using DIII-D's flexible heating systems to develop a range of scenarios, including a fully noninductive 1MA hybrid scenario (Fig. 2). High performance regimes are shown compatible with radiative divertor and also innovative new divertor configurations. A strong science program using high-resolution diagnostics and simulation to understand and project behavior underpins this work. For example, ECE imaging captures turbulence structure (Fig. 3) to confirm theory predictions of a trapped electron mode. Such work provides a framework to project and optimize behavior in future fusion devices.

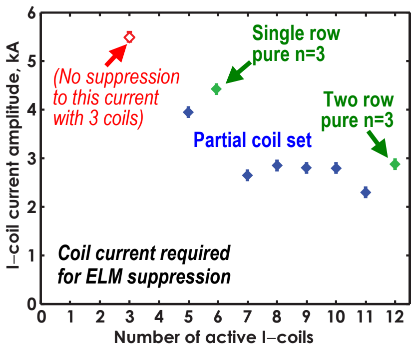

Fig. 1. RMP ELM suppression is maintained as coils turned off.

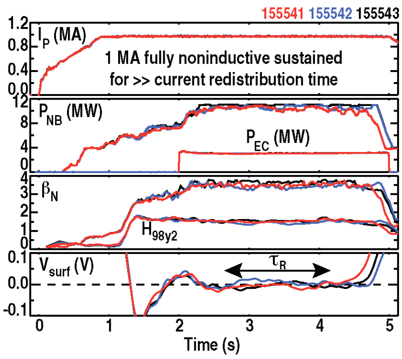

Fig. 2. Fully noninductive hybrid scenario.

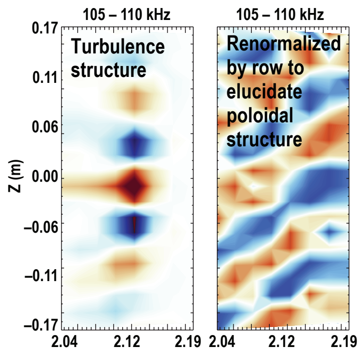

Fig. 3. Localization of TEM turbulence with ECE imaging.

Recent DIII-D experiments have developed improved confidence in ELM-free regimes. 'RMP' ELM suppression with 3-D fields is found viable even when several coils are turned off (Fig. 1), with the reduction in the primary n=3 harmonic compensated by additional sidebands to maintain suppression at similar coil current and confinement levels. A coherent physics picture is emerging with modeling and experiments showing enhanced plasma response to 3-D fields at the pedestal top, which is theorized to limit its width and maintain stability. ELM free QH mode has been extended to robust operation at 80% Greenwald density fraction, validating theoretical concepts that predict ITER can access the regime. QH mode has been sustained at the ITER baseline β_N, H_{89} and q_{95} for 18 confinement times, while the associated edge harmonic oscillation is found to flush impurities better than ELMs.

Disruption mitigation studies on DIII-D show that massive gas injection does not lead to significant localized heat loads due to radiation asymmetry, while measured runaway electron dissipation is much higher than theory predicts. Studies with toroidally and poloidally separated gas injectors showed <30% asymmetry in radiation during the thermal quench, irrespective of relative timing or number of injectors. An underlying 1/1 mode, theorized to cause radiation asymmetries, led to weak additional variations in radiation that could be phase controlled with 3-D fields. In the case of vertical displacement, mitigation of heat loads, forces and halo currents depended only modestly on injector location, though earlier mitigation gave clear advantages. A new slide-away technique to obtain runaway electrons in the ITER avalanche regime showed increased dissipation in the runaway population compared to Rosenbluth theory, leading to decay rather than growth.

DIII-D is developing the basis to optimize and control scenarios for the new regime of burning plasma operation in ITER, with low rotation and electron heating. Transport studies showed how long wavelength turbulence rises as these conditions are approached, with transitions from ITG to TEM turbulence increasing particle transport and density peaking, requiring new optimizations be developed. For stability, real-time ray tracing calculations and new spectral identification techniques enabled electron cyclotron current drive to suppress tearing modes earlier, with reduced power and confinement impact. A physics based method for error field correction yielded better 3-D optimizations: for each toroidal mode number the plasma was found most sensitive to a single "least stable ideal kink" field component. However, in H modes additional neoclassical toroidal velocity (NTV) braking was observed from other orthogonal field components, which must therefore also be optimized.

DIII-D has demonstrated promising candidates for steady state tokamak fusion, exploring the optimization of heat and current distributions for ITER, FNSF and DEMO. A fully non-inductive hybrid scenario (q_{min}~1, 1.0 MA) was developed at β_N ~3.6 (Fig. 2), providing an attractive target for a nuclear science facility with efficient core current drive. For DEMO, scenarios with higher β_N and bootstrap current fraction potential were found with more off axis current and higher q_{min}. These had good thermal confinement, but increased fast ion transport due to Alfvénic modes, with behavior consistent with a critical gradient model. An alternative "high l_i" scenario with peaked core current, first demonstrated transiently with β_N~5, has been sustained at β_N=3.6 in ITER-like configuration, providing an attractive candidate for ITER steady state using its day 1 heating systems. DIII-D's flexibility also enabled development of scenarios for EAST, demonstrating a fully non-inductive regime with 80% bootstrap and lowered torque injection consistent with EAST capabilities.

DIII-D has explored the physics of improved boundary solutions and detachment. Radiative divertor operation was demonstrated in ITER baseline and steady state scenarios, giving high performance, reduced heat load and good impurity control through D_2 gas puffs. Steady-state scenarios were also compatible with a snowflake divertor, combining with radiative techniques to reduce peak heat flux by up to 70%. Upgraded divertor Thomson scattering measured T_e to below an eV and observe the recombination region in 2-D. Using this system and a dual gas injection technique, detachment control was possible while maintaining constant core density. Upstream scrape-off layer profiles exhibited a critical gradient behavior, increasing with ideal ballooning limits until detachment. Increasing connection lengths broadened the SOL and lowered divertor heat flux, highlighting the importance of cross-field transport. Finally DIII-D's carbon wall enabled critical tests of impurity dynamics, including prevention of high Z erosion by low Z gas injection.

Finally, DIII-D continues to expand the scientific basis of fusion to enable confident projection and optimization for future devices. For example, transport studies characterized turbulence in electron heated regimes (Fig. 3) to constrain predictive models of burning plasmas. Turbulence was also found to play a crucial role in L-H transition, leading to increased flow which in turn suppressed the turbulence. In addition, 3-D fields were found to redistribute energetic ions, sometimes leading to losses over a single poloidal orbit, which provided a way to diagnose this effect. Detailed measurements of H-mode pedestal structure could only be successfully simulated by including kinetic effects in full f simulations; optimization of the pedestal exploited a valley of improved stability to raise performance.

For the future, DIII-D will continue to enhance electron cyclotron and off-axis current drive capabilities to develop the basis for burning plasma physics operation in ITER and steady state solutions beyond ITER. DIII-D will target urgent ITER questions with new 3D coils and power supplies for ELM and MHD control, and new disruption mitigation technology. For the longer term, work on the boundary is focused on developing the physics basis for an improved divertor configuration to meet the challenges of future steady state fusion reactors.

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