Development of Burning Plasma and Advanced Scenarios in the DIII-D Tokamak*

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The DIII-D research program emphasizes development of integrated scenarios for burning plasma experiments and investigation of elements in those scenarios that are critical for the success of proposed devices such as ITER. In this light, stationary discharges have been developed which exceed the ITER baseline design figure of merit ($G \equiv \beta_N H_{89}/q_{95}^2$) by 50% for durations much longer than the current relaxation time ($t_{dur} > 4\ \tau_R$). Looking beyond the initial stage of burning plasma experiments, steady-state operation of a tokamak is attractive. DIII-D has demonstrated the first fully noninductive discharges that project to high fusion gain in an ITER-sized tokamak. This requires simultaneously high β ($\beta_N > 3$), high confinement ($H_{89} > 2$), and high bootstrap fraction ($f_{BS} > 0.5$), plus the capability in DIII-D to drive current where required to maintain a stable q profile using electron cyclotron current drive (ECCD). Supporting these efforts is a vigorous program of basic physics studies in the areas of stability, transport, plasma boundary, and current drive physics. Highlights of recent research in these three broad areas, burning plasma scenarios, advanced scenarios, and basic physics studies, are detailed below.

Projections from present-day experiments to next-step devices such as ITER should be based on the performance of stationary plasmas. Discharges in DIII-D with $q_{95} = 3.2$ achieve $G > 0.6$ for longer than $4\tau_R$, compared to $G = 0.42$ for the ITER reference scenario. Access to this higher performance comes through modification of the plasma initiation to obtain current profiles that are stable to n=1 tearing modes. Using the same method, the ITER G value can be obtained at reduced current ($q_{95} = 4.4$). Reduced current in ITER would lessen the impact of a disruption, and the flux savings would allow operation at this level of performance for longer than an hour, compared to a 500 s operation goal of the baseline scenario. Thus, the core ITER design may be capable of achieving both its high fusion gain and high fluence missions simultaneously, building a solid case for a demonstration power station.

Research in DIII-D is addressing many critical issues for burning plasma operation. One such issue is confinement scaling. Scalings derived from the ITER global database indicate a strongly unfavorable scaling with β. Recent experiments in DIII-D see no degradation while scanning β a factor of 3 in H mode, confirming and extending previous experiments on DIII-D and JET. An H-mode confinement scaling relation has been developed using the global database, but constrained by the DIII-D dimensionless scalings, with little increase in residual compared to the standard fit.

Energy deposition by ELMs in the divertor is a critical issue for divertor life. The energy available to an ELM is determined by the pedestal height. Assuming the pressure gradient is limited by intermediate-n MHD stability, a model for the width is needed to determine the height. A density width model based on neutral penetration, presented at the previous IAEA meeting, has been further validated by joint experiments with JET. This model predicts very narrow density width in ITER. Analysis of the scaling of the temperature width is continuing. ELMs have been suppressed in DIII-D by use of stochastic magnetic fields at the plasma edge without loss of energy confinement. Stationary ELM-free discharges developed in DIII-D (QH mode) have been reproduced in joint experiments on AUG, JET, and JT-60U.

Another significant issue is the retention of tritium in plasma facing components (PFCs), due to co-deposition of tritium with carbon. This raises doubts as to the suitability of carbon for PFCs. Studies using carbon-13 methane puffing to simulate wall sputtering sources show that carbon from the main wall is swept to the inner divertor by a strong scrape-off layer (SOL) flow, resulting in co-deposition in this region. Experiments will be done in the coming

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year inserting a heatable test tile to determine the effects of tile temperature on co-deposition, especially in regions shadowed from direct plasma contact.

The large plasma stored energy in ITER at full current represents a potential for damage in a disruption. The DIII-D team has developed methods both to prevent disruptions and to mitigate their effects. The m=2/n=1 tearing instability is the most significant mode leading to disruption. This mode is suppressed by ECCD using closed-loop feedback to maximize the suppression rate. The driven current is the dominant means of suppression rather than heating. Ongoing modeling of the existing data should be able to differentiate between the effects of axisymmetric and helical currents. If a disruption does occur, a high-pressure gas jet system has been demonstrated to radiate the plasma energy without generation of runaways. Increased jet pressure provides more effective mitigation. A jet with a tenfold pressure increase over previous experiments is now available. Newly installed cameras will be employed to image the jet and validate models of its effects.

DIII-D experiments have made a significant step forward in the development of steady-state scenarios. Fully noninductive operation with the figure of merit $G > 0.25$ has been sustained in DIII-D for ~1 s. This projects to high fusion gain in an ITER-sized device. Operation above the no-wall n=1 limit is achieved by maintaining rotation through active feedback to reduce the effects of error fields. Off-axis ECCD is necessary to maintain a stable current profile. These discharges achieve $\beta_N > 3$, $H_{89} > 2$, and $f_{BS} > 0.5$ simultaneously. Experiments are ongoing to optimize these discharges, including replacement of NBI with fast wave heating for better alignment of the noninductive current profile with the total current. Part of the optimization is to understand how the ideal MHD β limits vary with shape and q_{min}. Shape optimization is critical since particle control by divertor pumping is essential for self-consistent scenarios in DIII-D. More highly shaped plasmas, measured by (I/aB) q_{95}, are found to have slightly higher β_N limits. The maximum β in long pulse discharges decreases as q_{min} increases, leading to an optimum value for maximum f_{BS}. Measurements of the no-wall n=1 limit indicate that the maximum sustained β in long pulse discharges and the no-wall limit have a similar dependence on q_{min}.

Without rotation, active stabilization of the n=1 resistive wall mode (RWM) is required to operate above the n=1 no-wall limit. Previous experiments on DIII-D with external coils indicated that modeling of stabilization using a rigid plasma model described well the observed effect of the coils on the RWM. Guided by these predictions, internal saddle coils have been installed in DIII-D in order to approach more closely the ideal wall β limit, independent of plasma rotation. Initial experiments show stabilization for many growth times in the absence of rotation by active feedback of the saddle coil currents. These experiments will be extended by enhancements in the power supplies and control algorithms. The internal coils have also been used to probe the plasma response to external magnetic perturbations. Analysis of these experiments and others with rotation braking should lead to a better understanding of the dissipation mechanisms within the plasma. Knowledge of the dissipation is critical to prediction of the requirements for stabilization in future tokamaks.

Advances in physics understanding in DIII-D have been driven by new diagnostic capabilities. In the boundary, a clear picture of impurity expulsion by ELMs is given by active charge exchange recombination (CER) spectroscopy measurements of the impurity temperature, density, and rotation with time resolution down to 300 µs. These measurements show the ELM flattens the plasma potential and pushes impurities as far into the SOL as can be measured. A comprehensive set of cameras monitoring the plasma cross section have identified leakage of carbon up the centerpost from the inner divertor leg as the dominant source of carbon to the core plasma. In the stability area, the fast CER measurements (here in the plasma center), combined with motional Stark effect measurements of the internal magnetic field and the unique shaping capability of DIII-D, provide insight into the dynamics of the sawtooth instability. Using shaping, the Mercier stability criterion can be moved from below to above q=1. Changes in the sawtooth behavior are observed, along with differences in response to localized heating with ECH. For core transport, a key issue is the existence of an instability threshold. Previous experiments with modulated ECH showed no evidence for threshold behavior in the electron channel, and new experiments, using a more direct technique developed on FT-U, again show no evidence for nonlinearity. First results from turbulence measurements in the short wavelength regime have been obtained and will be extended in the coming year.