TOKAMAK CONCEPT IMPROVEMENT RESEARCH IN DIII–D*

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The DIII–D program objective is to develop the science basis for improvement of tokamak fusion systems with a long term goal of demonstrating a high confinement, high β, steady-state plasma with power plant relevant heat and particle handling capability. Increased β allows higher power density, improved confinement and steady-state reduce capital and operating costs, and together they make a fusion power plant economically more attractive. This benefit can be realized only if the added challenge in heat and particle handling can be simultaneously met. In DIII–D, a quantitative goal is to simultaneously demonstrate a normalized beta (β_N) of six, a confinement enhancement factor (H) of four, and bootstrap fraction (f_{bs}) › 70% with ITER-relevant power and particle exhaust. An important near term objective is to carry out dedicated experiments, in collaboration with other tokamaks when appropriate, to provide key scientific inputs for optimizing ITER design. Results from theory and experiments have suggested several paths for improving stability and confinement in DIII–D. The improvements will come from shape optimization and control of internal profiles including current, pressure and rotation, as well as divertor heat and particle control. Separate improvements in key quantities have been demonstrated, accompanied by important physics understanding obtained through extensive diagnostic measurements and theory validation. With the availability of new tools such as FWCD, ECCD and a new radiative divertor in the near future, DIII–D is entering the integrated demonstration phase of its concept improvement program

The strong shaping capability of DIII–D has been used to make great advances in high β and stability research. Values of $\beta \le 12.5\%$ and $\beta_N = 6$ have already been achieved in strongly shaped plasmas. Theoretical calculations show two distinct regimes for achieving improved stability: 1) negative central shear (NCS) produced with a hollow current profile — NCS allows access to the second stable regime to ballooning modes, stability to low n kink modes with wall stabilization and well-aligned bootstrap current; 2) high internal inductance (ℓ_i) — produces strong magnetic shear which is favorable for first regime stability. DIII–D has focused mostly on NCS discharges although some current ramp high ℓ_i discharges have been studied and more are being planned. A variety of NCS discharges have been produced under L–, H–, and VH–mode edge conditions, using early beam injection during current ramp up. These discharges have achieved $\beta_N \le 4$ with excellent confinement, $H \le 4.5$. Without active control to maintain the favorable profiles, these discharges would eventually be terminated by MHD instabilities. Density peaking was only observed in NCS L–mode. While peaked density is favorable for high neutron yield, these discharges typically ended in hard disruptions. From theoretical studies, we believe the disruptions were caused by tearing modes destabilized by the negative shear and peaked pressure profile. NCS H–mode has broader density profiles and eventually ended with a β-limiting instability associated with the build-up of edge bootstrap current. The DIII–D plasma control was used to control the peakedness of the pressure profile by toggling the discharge between L– and H–mode, thereby avoiding the hard disruption and achieving record neutron yield for DIII–D of $2.4 \times 10^{16}\ s^{-1}$. In addition, we have been able to improve the performance over a wide range of plasma shape using the NCS configuration, including a JET and ITER-like SND configuration. This offers promise for achieving higher performance for these machines. The β value for a standard ITER demonstration discharge has also been shown to be limited by the onset of resistive MHD instabilities.

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A wide range of data supports the hypothesis that a single physical mechanism, turbulence suppression via E×B shear flow is playing an essential, though not necessarily unique, role in reducing turbulence and creating a transport barrier in many improved confinement regimes in DIII–D, including NCS and high ℓ_i. For NCS discharges, a power threshold has to be exceeded before turbulence and transport are reduced, in agreement with ExB shear suppression theory. NCS is not necessary for the maintenance of high performance, although it may be necessary for the transition to high performance. It is possible that early formation of NCS allows access to the second stability regime and thus larger ∇p, which when coupled with increased toroidal rotation and smaller interior poloidal field causes the turbulence shearing rate [$\propto \nabla (E_r/RB_\theta)$] to exceed the stability threshold. An ion transport barrier is observed in all NCS improved performance discharges with ion thermal diffusivity reduced to below neoclassical values. An electron transport barrier is most clearly demonstrated in combined NBI and FWCD NCS discharges, and a density transport barrier is typically seen with NCS L–mode.

Most of the DIII–D high performance regimes are non-steady-state. NCS for example is produced by temporarily "freezing" the current profile using high power neutral beams. The profile evolves slowly according to the current diffusion time scale back to a monotonic profile. To maintain the favorable profile, active control of the current and pressure is required. DIII–D has begun a program of active current profile control using FWCD. NCS is particularly amenable to radiofrequency current drive because of its high temperature and low target density. The total plasma current driven in DIII–D has exceeded 250 kA, and the current drive figure of merit is consistent with theory, in particular with the ITER scaling for FWCD. Profile control using counter-current drive with fast waves has succeeded in prolonging and enhancing the shear reversal in NCS discharges which resulted in improved stability to MHD modes.

Success in achieving higher power density (β) in advanced tokamak regimes increases the challenge of power and particle handling in the divertor beyond even that of ITER. We are seeking divertor solutions of the detached type with strong divertor radiation, low core Z_{eff}, and effective fuel and helium exhaust. Our closest approach to meeting the ITER requirements has been with D_2 puffing (partially detached divertor or PDD) with divertor peak heat flux reduced by factors of 3–5, density control from the cryopump, and $H_{ITER-89P} \sim 2$ in ELMing H–mode. Tomographic reconstruction of bolometer and visible data show that the radiation is primarily in the divertor and from carbon. A new Divertor Thomson scattering system has provided 2-D maps of divertor temperature and density. Electron pressure conservation along the field lines is clearly seen in attached plasmas and pressure drops of a factor of ten in detached plasmas. We have found $T_e < 2$ eV during PDD operation. Modeling of these plasmas with UEDGE has shown that volume recombination is a dominant process and that the coupling of the inner and outer legs by neutrals is important in the detachment process. Techniques to concentrate impurities in the divertor are being studied with D_2 puffing and active exhaust. To date, we see modest (2–3x) enrichment of neon in the pump plenum.

We have designed a new double-null slot divertor and cryopump for DIII–D (the Radiative Divertor) which is estimated to reduce the core ionization source from divertor recycling by nearly a factor of 10. This is expected to improve the core confinement, control (lower) the core density for efficient current drive operation, and provide a test of the ability of the SOL and divertor codes to predict details of the wall geometry and recycling. The Radiative Divertor is a flexible design; the slot width and length can be easily changed. The first phase of the Radiative Divertor, together with 3 MW of ECCD power for off-axis profile control, will be used in the near future for the first integrated demonstration of advanced performance regimes.