Advanced Tokamak research at the DIII-D National Fusion Facility in support of ITER

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Abstract. Fusion energy research aims to develop an economically and environmentally sustainable energy system. The tokamak, a doughnut shaped plasma confined by magnetic fields generated by currents flowing in external coils and the plasma, is a leading concept. Advanced Tokamak (AT) research in the DIII–D tokamak seeks to provide a scientific basis for steady-state high performance operation. This necessitates replacing the inherently pulsed inductive method of driving plasma current. Our approach emphasizes high pressure to maximize fusion gain while maximizing the self-driven bootstrap current, along with external current profile control. This requires integrated, simultaneous control of many characteristics of the plasma with a diverse set of techniques. This has already resulted in noninductive conditions being maintained at high pressure on current relaxation timescales. A high degree of physical understanding is facilitated by a closely coupled integrated modelling effort. Simulations are used both to plan and interpret experiments, making possible continued development of the models themselves. An ultimate objective is the capability to predict behaviour in future AT experiments. Analysis of experimental results relies on use of the TRANSP code via the FusionGrid, and our use of the FusionGrid will increase as additional analysis and simulation tools are made available.

1. Introduction

The primary goal of next-step fusion devices such as ITER will be demonstration of sufficient fusion performance to merit consideration of fusion as a viable energy source. "Conventional" scenarios such as the High Confinement mode of operation (H-mode) appear capable of fulfilling this requirement. However, reliance of these scenarios on a transformer makes them inherently pulsed rather than steady state as preferred for a power generating station. Advanced Tokamak (*AT*) research [1-4] focuses on developing steady-state scenarios with fusion performance comparable to the conventional H-mode.

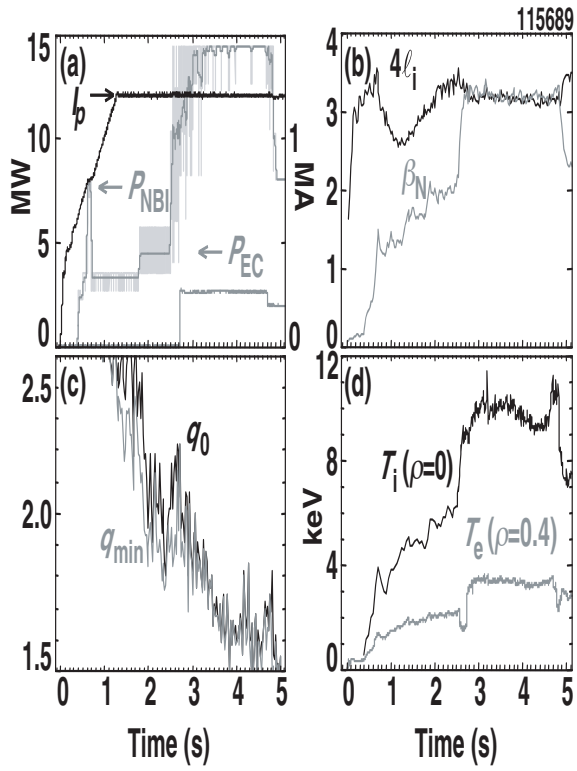
For steady-state operation, all plasma current must be supplied noninductively. To minimize dependence on external systems, the self-generated bootstrap current [5], I_{BS},, must provide most of the plasma current. The bootstrap current fraction $f_{BS} = I_{BS}/I_p \propto \beta_p \propto q\beta_N$, where $\beta_{P,T} = 2\mu_0\langle\pi\rangle/B_{P,T}^2$ is the ratio of the plasma pressure to the poloidal (P) or toroidal (T) magnetic field pressure, $\beta_N = aB_T\beta_T/I_p$ is the normalized beta (with minor radius a in m, toroidal magnetic field B in T, β_T in percent and plasma current I_p in MA). The remaining current is provided by external sources.

Fusion gain increases with the triple product $nT\tau_E \propto \beta_T\tau_E B^2$, where n is the plasma density and T is the temperature. Maintaining high gain requires high β_T and energy confinement τ_E. Simultaneous optimization for both bootstrap current and gain depends on maximizing $\beta_P\beta_T \propto \beta_N^2$, This ne-

cessitates operation near the pressure limit. This requires control of magnetohydrodynamic (MHD) stability and transport processes in the plasma.

Requirements for *AT* operation [1,6,7], specifically the DIII-D approach [8], have been described elsewhere in more detail. Here, we briefly summarize this approach, which relies on neutral beam (NBCD) and electron cyclotron current drive (ECCD). Following this approach, we have demonstrated fully noninductive conditions with $\beta_N = 3.6$ for several τ_E and nearly noninductive conditions for longer than one current relaxation time τ_R.

Simultaneous integration of the individual scientific elements into steady-state high performance scenarios remains the greatest challenge, due largely to the many couplings between the different scientific elements. A comprehensive integrated modeling effort is carried out along with these experimental efforts. Experiments are both planned and interpreted in light of these simulations. This benefits both the experiments and the models, since the results guide development of both. This supports a major goal: to develop a predictive capability for application to the design of advanced scenarios in next-step burning plasma experiments.



Experimental data analysis relies heavily on the TRANSP [9] code, provided as a service via the FusionGrid [10]. We anticipate using other such services for future analysis, modeling and simulation.

2. The physics elements of Advanced Tokamak research

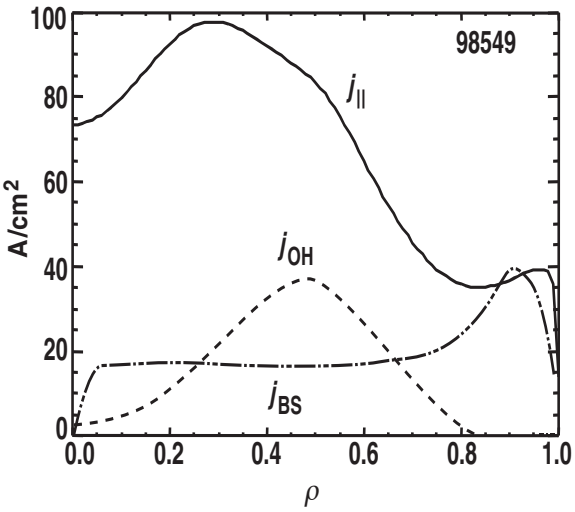
Our approach is to build physics understanding and control capability of the scientific elements and then combine the knowledge and tools to produce *AT* configurations. The elements fall into four general categories: (1) Facilitating operation at high β; (2) noninductively modifying and maintaining the current profile; (3) modifying and controlling transport; and (4) controlling particles and energy exhausted through the boundary. We briefly touch upon the first two; full detail is given in [8].

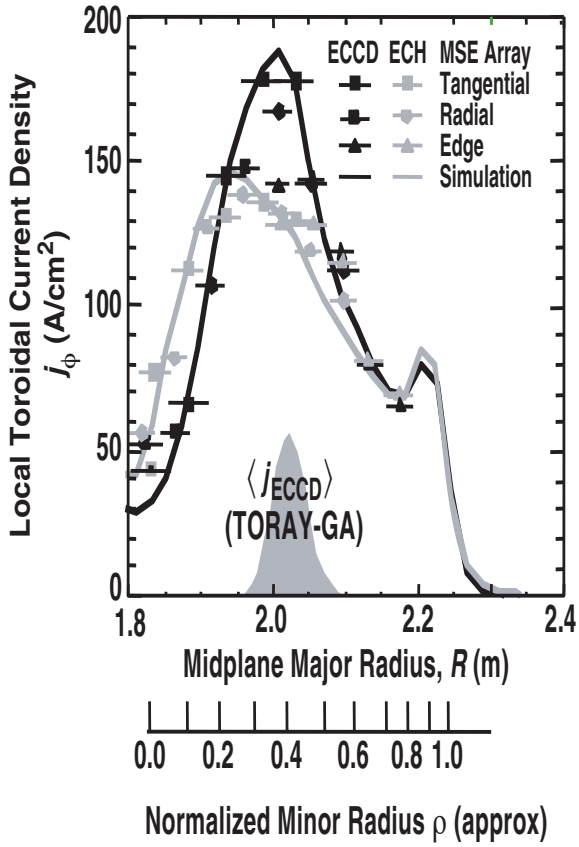
These experiments make use of variations on a typical DIII-D *AT* discharge (Fig. 1). Early in the current ramp, a momentary increase in the heating power triggers an early transition to H–mode. This broadens the temperature and density profiles and, with the addition of an H-mode pedestal, results in a hot core plasma, slowing the resistive evolution of the current profile and allowing access to plasmas with high q_{min} after the plasma current has reached flattop. The high power phase is timed to coincide with the desired value of the minimum safety factor q_{min}, typically 1.5–2.0 s for $q_{min} \geq 2.5$ and 2.5–3.0 s for $q_{min} \geq 1.5$.

2.1. High beta operation

The need to simultaneously maximize power density (β_T) and bootstrap (β_P) motivates operation near the pressure limit. Present experiments operate with $\beta_N \approx$ 3-4; higher values are anticipated in the future. We maximize the β limit through optimization of the plasma geometry and pressure profile [11–13], and active control of MHD instabilities such as the Resistive Wall Mode (RWM) [14–16].

Geometry (elongation κ and triangularity δ) and pressure profile shape both impact stability limits [11]. Both calculation and experiment [12,13] demonstrate that increasing δ and κ increases the ideal, low-n β_N limits that often limit performance. Pressure profile control is more difficult, since it relies on transport control, a less developed research area. We approach this by building an under-





standing of transport behavior and designing scenarios that are consistent with this underlying transport.

AT plasmas in DIII-D routinely operate above the no-wall β limit. This is allowed by rotational stabilization of the RWM, in turn facilitated by active control of error fields that might otherwise slow the toroidal rotation. Coils recently installed in the vacuum vessel are designed to provide direct RWM stabilization regardless of rotation. Efforts to exploit these coils in experiments have begun, with detailed results described elsewhere [15–17].

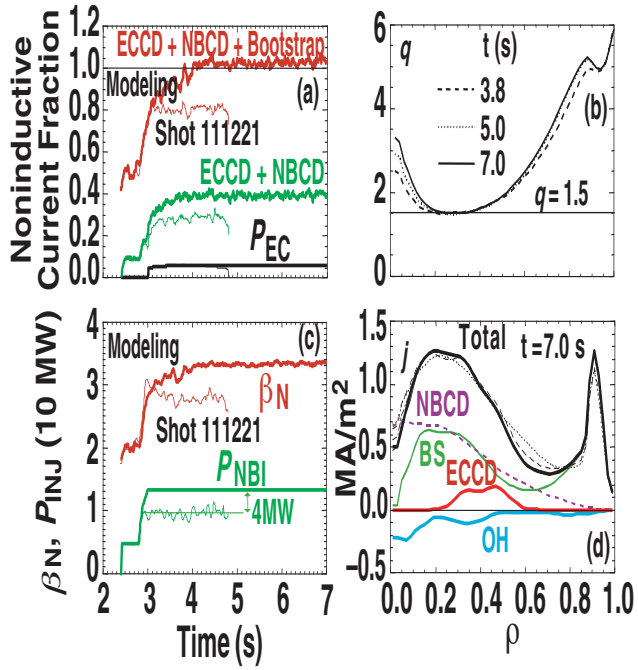
2.2. Noninductive current profile control

In conventional tokamak scenarios, most plasma current is driven inductively, with the plasma acting as the secondary of a transformer. These scenarios are inherently pulsed, decreasing their attractiveness for a power plant. The *AT* eliminates this constraint, allowing steady state operation without sacrificing fusion performance. Current is provided by means other than transformer action. The bootstrap current [5], driven by radial gradients in the kinetic profiles, provides most of the current in high β steady state plasmas. *AT* discharges in DIII-D typically have $f_{BS} \approx 50\%–70\%$, with simulations indicating feasible scenarios with $f_{BS} \approx 90\%$. In target plasmas (prior to ECCD activation), the remaining inductive current in the plasma amounts to about 25% of the total current, centered around the mid radius (Fig. 2) [1]. It is this current that must be replaced by additional noninductive sources.

ECCD can effectively provide this off-axis current. Simulations of a discharge with and one without EC indicate a difference in current density at the location where the EC waves are deposited (Fig. 3) [18]. Measurements, made by motional Stark effect (MSE) [19], are in good agreement with the simulation, and indicate that approximately 130 kA is driven by ECCD near the absorption radius.

3. Integrated self-consistent scenarios

Predictive modeling with the ONETWO [20] code, was carried out based on previously reported discharges (Fig. 4) [6]. The discharge forming the basis of this study has $q_{min} > 1.5$, $\beta_N^{max} \approx 3.1$, $f_{BS} \approx 55\%$ and $f_{NI} \approx 90\%$. The simulations indicate that increasing the neutral beam power by 4 MW would result in the plasma reaching $f_{NI} \approx 100\%$ at somewhat higher β. These calculations were repeated using the theory-based GLF23 model [21], with similar results.



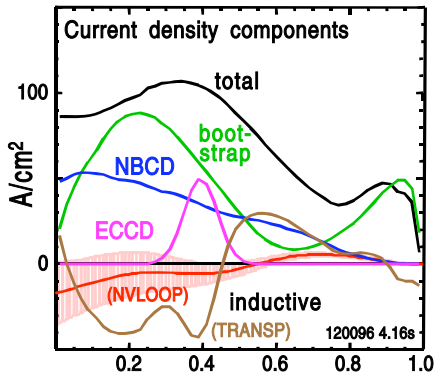


Fig. 5. TRANSP simulations are used to calculate contributions to the current profile from each source. The inductive current is calculated by subtracting these sources from the total current calculated by the EFIT equilibrium code. This is compared to an NVLOOP calculation.

TRANSP [9] is another code that is extensively used for modeling and analysis. Unlike ONETWO, this is run remotely using the FusionGrid [10]. This has proven very successful, since this code demands a large processor resource, and is therefore well suited to rely on a central processor farm such as the TRANSP cluster at Princeton Plasma Physics Laboratory (PPPL). This also eliminates the previous need for PPPL personnel to maintain TRANSP installations at remote labs, often on computer systems that are substantially different than their own.

Although TRANSP is extremely useful in its own right, it also serves as a model for other analysis, modeling and simulation tools to be deployed in the same way. Development is underway on several such codes. In the present work, TRANSP is used to interpret data (Fig. 5) and prepare data for simulations such as those discussed above.

Experiments based on the above simulations resulted in establishment of fully noninductive conditions (Fig. 5). The TRANSP inductive current profile is not fully relaxed, indicating here that the models used are not fully self-consistent. NVLOOP [22] calculates the same quantity based only on the temporal behavior of the EFIT equilibria, and in this case, indicates the inductive current has been eliminated throughout the plasma. $f_{NI} \approx 100\%$, $\beta_T = 3.5\%$ and $\beta_N = 3.6$ is maintained for 0.6 s, until the pressure profile evolves to an unstable state. A similar discharge was maintained with $f_{NI} \approx 90\%$–95% for 2 s, limited by available hardware duration.

4. Summary

AT research in DIII-D seeks to provide a scientific basis for steady state high performance regimes in next-step tokamaks. Our approach combines key advances in separate scientific areas into an integrated scenario. This has led to demonstrations of fully noninductive, high beta plasma states that will be directly applicable to experiments on ITER. Integrated modeling, primarily using ONETWO, is used to both interpret and plan experiments. TRANSP is used extensively for data analysis, and provides a demonstration of the potential power of the FusionGrid for providing computing resources for a variety of resource intensive tasks. We look forward to using other such tools as part of future *AT* research.

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