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DIII–D PROGRAM RESULTS AND FUTURE PLANS

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This presentation summarized recent research results and future plans for the DIII–D National Fusion Program. The full set of transparencies for this presentation can be found at http://fusion.gat.com/pubs-ext/presentations/Stambaugh_FPA00.pdf

The DIII–D research is carried out by an extensive international team composed of 9 U.S. national laboratories, 16 U.S. universities, 17 foreign laboratories, 9 foreign universities, and 11 industrial companies.

The DIII–D Program works with other programs internationally to optimize the tokamak approach to fusion energy production. The main focus is the Advanced Tokamak (AT) research line, but the program also seeks to resolve key enabling issues for next steps toward fusion energy and to advance the science of magnetic confinement on a broad front.

A brief listing of outstanding recent research results is given below.

Advanced Tokamak

1. Good progress on AT scenarios ($\beta_N H_{89P}$ ~ 9 for 2 seconds).
2. First results using smart conducting shells for wall stabilization.
3. Increased physics understanding of edge and internal plasma instabilities.
4. Exploration of internal transport barriers with counter injection and pellets.
5. Exciting new work affecting turbulence using impurity atoms.

Next Steps

1. New discovery — ELM-free H–mode without impurity accumulation or density buildup.
2. New discovery — H–mode confinement quality above the Greenwald density limit with gas fueling and pumping.
3. A scientific basis for the choice of the optimum shape of the plasma.

Broad Science

1. Measurement of the complex 2-D flow patterns in the edge plasma.
2. Studies of self-organized criticality.
3. Movies of edge plasma turbulence from plasma fluctuation measurements.

The DIII–D Advanced Tokamak Plan to the Fusion Energy Services Advisory Committee (FESAC) checkpoint in 2004 envisions expansion of the 110 GHz gyrotron set to eight units of the MW class, with an option to extend the power to 9.5 MW at 10 seconds through 1.5 MW units being developed by the Virtual Laboratory for Technology. With the completion of the upper divertor, the fueling and edge control systems will remain the same until a possible divertor improvement in 2004. The present 6 coil resistive wall mode feedback stabilization system will be expanded to 18 coils in 2002. Major new diagnostics are the lithium beam based edge current density profile system and future initiatives in electron transport diagnostics (high wavenumber turbulence measurements) and diagnostics for 3-D equilibria.

The primary integrated AT scenario uses off-axis electron cyclotron current drive (ECCD) to sustain a hollow current profile, reversed shear. Significant improvement in long-pulse advanced tokamak performance has been achieved with $\beta_N H_{89P} \sim 9$ for 2 seconds. This level of performance exceeds the requirements for the Japanese SSTR reactor study and almost meets the requirements for ARIES-RS. The limiting factors in these plasmas are the uncontrolled rise in the density that must be cured with divertor pumping and the inward diffusion of the current profile peak that must be arrested with off-axis ECCD.

The gyrotron set for DIII–D consists of three Russian gyrotrons of about 0.7 MW, 2 second capability and two diamond window gyrotrons from Communications Power Industries of nominal 1 MW, 10 second capability. Two of the Russian gyrotrons were purchased from the TdeV Program in Canada. The CPI gyrotrons have achieved 0.55 MW for 10 seconds, the limit of the testing capability at CPI. Two more CPI units will be installed as part of an approved upgrade program. It is planned to acquire two more units to replace the shorter pulse Russian tubes, resulting in 5.7 MW of 10 second power. The long pulse power could be brought to 9.5 MW in the 2003 time frame with the acquisition of three 1.5 MW gyrotrons being developed by the Virtual Laboratory for Technology. Eventual buildout of the system will be 8 gyrotrons with 8 waveguide connections to launchers into DIII–D.

A new 2-D steerable electron cyclotron (EC) wave launcher from Princeton Plasma Physics Laboratory (PPPL) is now available on DIII–D. It enables precise co-counter current drive comparisons from shot to shot. Very spatially localized ECCD has been measured already. Previous to the steerable launcher, what now takes 2–4 shots had to be done over two run days. Electron cyclotron heating (ECH) has been able to produce a strong transport barrier in the electron channel, 15 keV electron temperatures, and to completely suppress a 3/2 neoclassical tearing mode.

The upper divertor of DIII–D was modified to include a baffle and a cryopump in the private flux region. Both the inner and outer strikepoints of highly triangular plasmas can now be pumped independently. The tiles in the new upper divertor were precision aligned to 0.1 mm tolerance, effectively eliminating edges. infrared (IR) images of these tiles show edge hot spots have been eliminated. Owing to the reduced erosion of edges, the carbon content of plasmas in 2000 has been cut in half compared to 1999. The pumps provide sufficient density control in ELMing H–mode plasmas for our AT scenarios.

Complex uses are made of the plasma shape control capability in DIII–D to manage AT scenarios. Early in the discharge, an upper single null is operated but with the grad-B drift direction down to maintain a high L–H transition power and keep the plasma in L–mode and at lowish density while the current is being ramped up and the reversed shear profile formed. At the desired time of the L–H transition, the plasma is shifted up to an upper null configuration, abruptly lowering the L–H transition power. The strong edge temperature rise in the H–mode phase freezes in the reversed shear current profile. Then the plasma outer separatrix is moved to the outer pump throat to bring down the density (for eventual application of ECCD to maintain the off-axis current peak).

A program of study of the plasma edge has resulted in detailed measurements of the up-down divertor heat and particle efflux balance as a function of the distance between the in-general two separatrices of a divertor equilibrium. The heat flux balance shifts from top to bottom when the between separatrix distance is varied by the parallel heat flux scrape-off layer width, the anticipated result. The particle flux balance curve is much broader, implying that 2-D effects in the recycling of neutrals in the divertor are important. This information and other measurements from this study provide a basis for detailed divertor design in future machines.

It appears that a high density divertor solution is available. Recombining divertor plasmas with less than 1 eV temperatures at the divertor plate and much plate erosion nearly zero have been achieved. The peak heat flux is also reduced by a factor of five.

New physics is being investigated in the private flux region of the divertor. Flow reversals, the role of E×B drifts, and recombination are all being investigated.

The pumped divertor has enabled studies aimed at producing a highly radiating divertor by trapping impurity atoms in the divertor plasma with copious fuel ion flows down the divertor field lines. The large fuel ion flows are driven by large fuel gas injections into the scrape-off layer (SOL) and pumping of the plasma outflux in the divertor. Substantial purging of the core plasma of argon impurities has been demonstrated.

Wall stabilization and plasma shaping are essential for high performance advanced tokamak operation. Theory work shows that the achievable β_N can be approximately doubled with elongated, triangular shaped plasmas with stabilization of ideal kink modes with a nearby conducting wall. In reality, the growing mode diffuses through the finite conductivity metal wall. It has been suggested that feedback coils can counteract this leakage of the mode through the wall. Results with a 6 coil feedback system on DIII–D have shown extension of the duration of the plasma by use of the feedback system.

Detailed measurements and theory have led to an improved understanding of the physics of the steep pressure gradient region that forms just inside the separatrix of H–mode plasmas. The strong bootstrap current caused by the large pressure gradient opens second stable access to ideal infinite-n ballooning modes. Yet the plasmas still exhibit a pressure gradient limit. The answer is that finite toroidal mode number stability theory shows modes with n ~ 5 will still be unstable, albeit at a pressure gradient well above the infinite-n ballooning theory prediction.

DIII–D is performing research into methods of controlling the internal transport barriers that have so much promise for improved confinement. The central objective of this research is to move the transport barrier location where theory predicts much higher states of plasma pressure can be stable. The pressure gradient also must not become too steep. Counter injection has been shown to alter the profile of the radial electric field and to expand the radius of the transport barrier in a manner consistent with theory. The shear in the radial electric field is the active agent in suppressing plasma turbulence to allow transport barrier formation. Pellet injection has also been shown to cause internal transport barriers to form in the ion, electron, and density transport channels. Neon injection has also been shown to approximately double the energy confinement time. Theory calculations show this effect may derive from a favorable isotope effect on turbulence growth rates coupled with the E×B turbulence shearing mechanism.

DIII–D has been able to achieve discharges with H-mode quality confinement with the plasma density well above the empirical Greenwald density limit using just gas fueling and pumping in the divertor. This result was widely thought impossible. Most tokamaks, especially large tokamaks, shown a strong degradation of confinement as the empirical density limit is approached. DIII–D also sees a degradation of confinement early in these discharges when the density exceeds the empirical limit. However in one uses the pumping to prevent thermal collapse of these plasmas and waits patiently, the interior density profile slowly peaks up, restoring all of the confinement loss. Both the high confinement low and high density plasmas have in the end the same density profiles.

Using divertor pumping, an entirely new mode of operation, also widely thought impossible has been discovered. This mode is an H–mode plasma, but the plasma edge never develops the instability bursts discussed above and yet the density and the impurity content of the plasma do not increase. ELM-free H–mode phases had been seen before in tokamaks but only for a few hundred milliseconds and during such phases the impurity content and the plasma density rose steadily and uncontrollably. These ELM-free H–mode plasmas have lasted 2.5 seconds and are apparently in steady-state. Such a plasma edge is the nearly ideal plasma edge for tokamak reactors.

The long range Advanced Tokamak research plan for DIII–D points to the FESAC assessment point in 2004. The central AT approach is pursuit of the high bootstrap current fraction, reversed shear mode, first sustaining it for long pulse and then optimizing it. This central research line is supported by research thrusts in divertor physics for the necessary density and impurity control, ECCD and ECH physics for the necessary current profile control, edge stability studies for density control and preventing the collapse of AT states owing to excessively large edge instabilities, neoclassical tearing mode control by use of ECCD, resistive wall stability research to enable access to the theoretically predicted high beta states, and internal transport barrier research to eventually optimize the confinement.

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