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The Alcator C-Mod Program

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To introduce this special issue, this paper gives a brief overview of the key themes in the Alcator C-Mod tokamak research program.

I. Introduction

Alcator C-Mod[1] is a high magnetic field tokamak with strong shaping capabilities. While compact in physical dimensions, C-Mod produces plasmas which overlap, in dimensionless parameters and absolute performance, with those produced in much larger devices. Auxiliary heating and current drive systems for C-Mod exclusively employ radio-frequency tools (Ion-Cyclotron and Lower Hybrid), naturally decoupling the heating, fueling and momentum sources. Routine operation at high absolute density, enabled by large B/R, allows C-Mod to explore regimes with fully equilibrated electrons and ions. Compactness also yields very high power densities and particle fluxes, expanding the available parameter space for scrape-off-layer and divertor physics and technology studies. C-Mod has always and exclusively used high Z metallic plasma facing components for all high heat flux regions. C-Mod produces the highest absolute pressure plasmas (at the ITER magnetic field and β), with the current tokamak record for volume average plasma pressure (1.8×10^5 Pa).

II. History

C-Mod is the third in the series of high field tokamaks built and operated at MIT. The first, Alcator A,[2,3] which operated from 1972 through 1980, was a circular, limiter tokamak, which pioneered the compact high field approach. Alcator C,[4] also circular and limited, was scaled up in size from Alcator A, with smaller aspect ratio, and operated from 1979 to 1987. High current tokamak plasmas on Alcator C-Mod, which added advanced shaping and divertor capabilities, were first produced in 1993.

III. Parameters

Alcator C-Mod provides physics data and understanding of several critical issues facing tokamaks on the path to a viable fusion reactor. C-Mod is the only divertor tokamak in the world operating at (and above) the ITER design field of 5.3 T; the maximum C-Mod field of 8 T is prototypical of future advanced reactor designs. Therefore C-Mod is able not only to operate at the same non-dimensional plasma parameters as the burning plasma (except for gyro-size), but also at the same absolute plasma pressure and power density. Some of the main parameters of the C-Mod facility are listed in table 1.

Alcator C-Mod has an extensive diagnostic set,[5] for detailed measurements of plasma parameters, supporting all aspects of the experimental program.

As a consequence of its high magnetic field (B ≤ 8 tesla) and compact size (~1/10 of ITER[6], figure 1), C-Mod operates with higher plasma density than larger, lower field devices. A direct result of this fact is that over most of its operating space, the product ν_{ei} x τ_E is large, such that electrons and ions are always closely coupled and equilibrated.

Such close coupling, resulting in $T_e \sim T_i$, is characteristic of a reactor, or a burning plasma experiment such as ITER. While other tokamaks can achieve near-equality of temperatures by careful adjustment of the relative ion and electron heating, C-Mod achieves this condition naturally, as would a reactor. The T_e/T_i ratio is fundamentally important to the transport physics[7].

IV. Radio Frequency Heating and Current Drive

C-Mod utilizes RF (ICRF and LH) exclusively as auxiliary heating and current drive sources[8]. This choice, in contrast to neutral beam auxiliary power, separates the heating and current drive from the particle and momentum sources. In this sense, the heating in C-Mod is prototypical of the alpha particle heating which will be dominant in any burning plasma. This decoupling of source terms facilitates design of unique transport experiments. In particular, the lack of momentum sources has allowed for the detailed investigation of intrinsic rotation[9].

ICRF minority heating studies have shown high overall heating efficiency (up to 90%) for the strong single-pass absorption hydrogen minority, deuterium majority regime, with somewhat lower efficiency for the low single-pass absorption ^{3}He minority, D majority regime.[10] Mode conversion to both the Ion Bernstein and Ion Cyclotron waves

has been observed[11,12] using detailed measurements with the Phase Contrast Imaging diagnostic, confirmed by the results of numerical simulation.[13,14]

C-Mod has begun using lower hybrid current drive for control and sustainment of the current profile. This is the most efficient non-inductive technique available, particularly for deposition far off-axis to optimize advanced scenarios. C-Mod is the only diverted tokamak in the world (and thus with H-Mode edge profiles) testing LHCD at the density and field range of ITER, which are key parameters determining the RF physics, including accessibility and current drive efficiency.[15]

V. Boundary Science

The high power and particle density of the C-Mod divertor and scrape-off layer exceed those achievable in all other tokamaks by factors of two to five and provide access to conditions closest to those projected for ITER. C-Mod studies of divertor detachment[16] are performed at ITER-relevant parameters, providing key data on reduction of power flux to material surfaces and the effects of volume recombination for reducing particle fluxes. Both plasma and neutral densities in the C-Mod divertor are similar to those predicted for ITER, providing a unique capability to study issues of radiation transport and trapping, and ionization balance under divertor conditions (n_0L) approaching those of ITER. The similarity in C-Mod and ITER of normalized neutral mean free path (scrape off layer transparency) is important for fueling, plasma-wall interactions, impurity screening and transport. C-Mod features extremely high parallel heat flux in the Scrape-Off Layer (SOL), equal to that expected in ITER (~ 1 GW/m^2).

Power handling will be a particular challenge in long-pulse, advanced regimes since efficient current drive is favored by lower edge densities which may preclude radiative detachment. This may well prove one of the most challenging aspects of applying such scenarios. Near term experience on C-Mod, both with optimizing physical parameters and in testing advanced divertor materials, will be particularly important in establishing feasibility of long pulse advanced scenarios in these regimes.

Studies of turbulence, transport and flows in the C-Mod SOL reveal the presence of large asymmetries between the high and low field sides of the SOL, with resulting strong flows.[17] Anomalously large perpendicular transport in the SOL results from striation like turbulent structures, which exhibit $k_\parallel << k_\perp$, have characteristic perpendicular scale size of 1 cm and lifetimes in the range of 10's of microseconds.[18]

Present plans for reactor Plasma Facing Component (PFC) surface materials call for tungsten due to its low tritium (T) retention, capability to handle high heat fluxes with low erosion, and robustness to nuclear damage and activation.[19] ITER, to provide required operational experience for DEMO, will likely at some point in its lifetime operate with all tungsten high-Z PFCs. Alcator C-Mod experiments provide unique divertor tokamak operational experience with all high-Z (molybdenum) PFCs, comparing boronized and un-boronized surfaces. The results from these experiments indicate that high-Z PFC operation, without boronization, carries the risk of degraded confinement; boronization, or other low-Z wall coating, might be required with high-Z PFCs.[20]

Three regimes of H-Mode pedestal are seen on Alcator C-Mod[21,22,23]: ELM-free, Enhanced D-Alpha (EDA) and ELMy. Regulation of transport (particularly for particles, including impurities) through the edge transport barrier is different for the three cases,

with impurity accumulation for ELM-free. In the EDA case, a quasi-coherent mode drives the particle transport, and leads to steady high performance conditions, without impurity accumulation.[24] The onset of ELMs occurs at the highest edge pressure gradients and reduced collisionality (typically $\beta_N \geq 1.3$), and is well modeled using a coupled peeling/ballooning mode analysis.[25]

VI. Advanced Scenarios

The Advanced Tokamak program on Alcator C-Mod is an integrating thrust which combines aspects of transport physics, wave-plasma interactions, stability and plasma boundary physics. It focuses on the development of scenarios which feature control of current as well as kinetic profiles, leading to largely non-inductive current drive with high bootstrap fractions. Typically, this requires reversed shear current profiles which should be achievable on C-Mod by the simultaneous application of efficient off-axis LHCD, perhaps combined with modest central mode conversion or fast wave current drive, and intense ICRF heating to generate significant bootstrap current ($\sim$70%)[26]. Such scenarios could lead to more attractive fusion reactors, as discussed in the Aries RS and Aries AT studies[27].

The current relaxation time (τ_{CR}) is short compared to pulse length in C-Mod, due to its small minor radius ($\tau_{CR} \propto a^2$). C-Mod is perhaps the only divertor tokamak which routinely operates for pulse lengths exceeding the current relation time, which is the relevant time scale for maintaining quasi-steady current profiles. This situation prevails in C-Mod even at full heating power and with low Z_{eff}. Once significant fractions of the

current are non-inductively driven on C-Mod, pulse lengths of up to 5 seconds will be possible, corresponding to at least 5 current redistributions times (at $T_e = 6$ keV, $Z_{eff} = 1.5$) and exceeding the L/R time.

The tight coupling of electron and ion heat channels makes the formation and control of transport barriers, particularly in the core, more challenging: reducing transport in only one channel is not sufficient to guarantee the strong gradients required for large bootstrap current generation. Internal Transport Barriers (ITBs) have been produced in C-Mod plasmas using off-axis ICRF heating[28]. These ITBs, with high pressures (up to 4×10^5 Pa peak) and pressure gradients (to 2.5×10^6 Pa/m), are unique in that they are formed with no particle or momentum sources, and can occur with normal magnetic shear. The barrier foot location can be controlled by changing the toroidal magnetic field and/or the plasma current, and appears well correlated with changes in q_{95}, while the peak pressure (as well as impurity accumulation) can be controlled with additional on-axis ICRF heating power. In most ITB experiments on other facilities, core particle sources from NBI, as well as NBI momentum input to modify the local velocity shear, are thought to play important roles in the formation of core barriers.[29]

VII. Disruption Studies

Because of the high magnetic fields and current densities, disruption forces are very large on Alcator C-Mod. Extensive diagnostics were incorporated into the baseline set on C-Mod to look at disruption dynamics, and these measurements led to the discovery of large toroidal asymmetries in the halo currents that can dominate the forces on structures during disruptions.[30] Many of these results were incorporated into the ITER disruption

database, and have strongly influenced the design of that device.[31] More recently, extensive experiments have been carried out on disruption mitigation through the use of massive noble-gas puffing, first demonstrated on DIII-D.[32] The emphasis on C-Mod is to extend these studies to plasmas at high absolute plasma pressures energy densities (comparable to those expected in ITER); both parameters can be important, affecting neutral gas penetration, and effectiveness of radiative power dissipation, respectively.

VIII. Fast particle studies

Fast particle driven modes are studied on C-Mod using both passive and active techniques. Stable Alfven eigenmodes are probed using low power antennas;[33] by sweeping the drive frequency across the resonance, the q of the system, and thus the damping rate of the modes is measured. Typical damping rates are in the range from .5% to 5%. Toroidal Alfven eigenmodes are driven unstable in C-Mod plasmas by the ICRF minority tail ions under various plasma conditions, including low density discharges with normal shear, and in higher density plasmas with flat or reversed shear, during the current rise phase.[34]

Table 1: Alcator C-Mod Facility Parameters

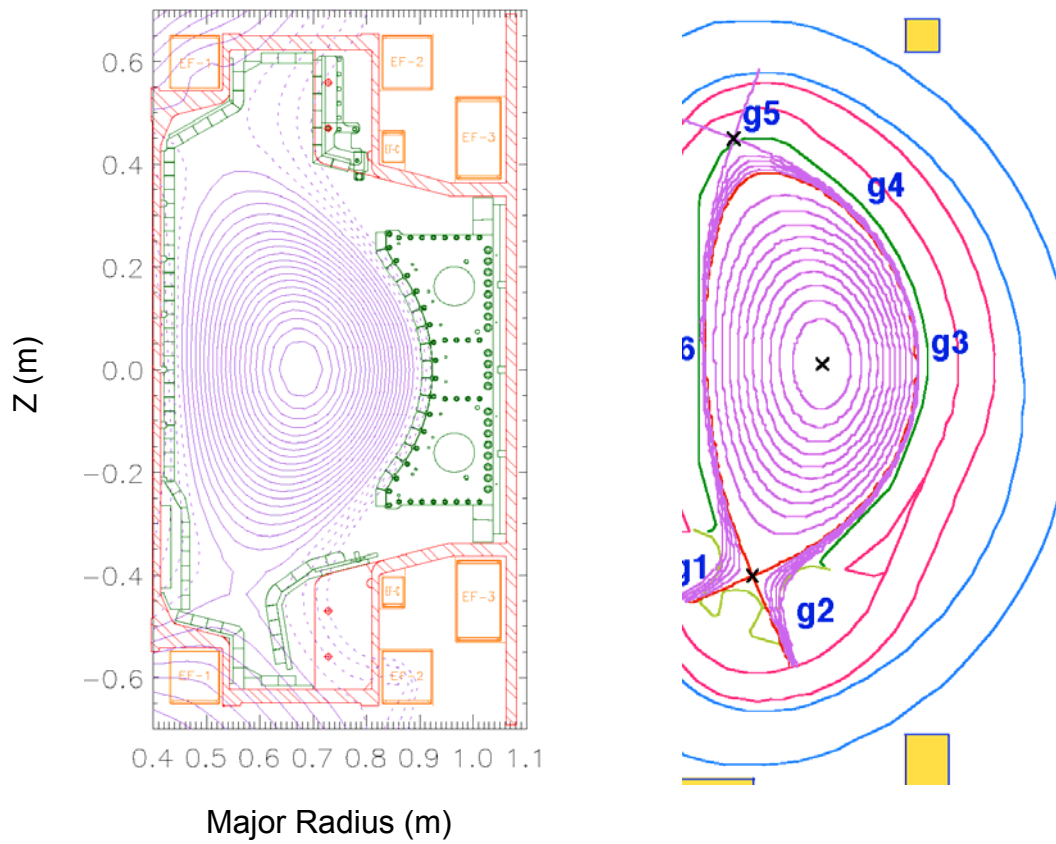|	
Dimensions; Plasma Volume; Surface area	R=.68m, a=.22m; 1 m^3; 7 m^2
Maximum Toroidal Field	8.1 tesla
Maximum Plasma Current	2.01 MA
Plasma Shaping	Elongation < 1.9, Triangularity < 0.85
Max Toroidal Field pulse length (at 5 Tesla)	5 seconds, $>>\tau_{CR}$
Ion Cyclotron RF source power	8 MW, 50 to 80 MHZ
Lower Hybrid RF source power	3 MW, 4.6 GHz
Plasma Pressure	Up to 1.8x10^5 Pa (volume average)

Figure 1 *C-Mod equilibrium (left) can match precisely the shape of the ITER design (right, shown scaled by 1/9 in linear dimension). The parameters for this actual C-Mod discharge were B_T = 5.3 T, I_P = 1.6 MA. The corresponding parameters for ITER are B_T = 5.3 T, I_P = 15 MA.*

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