1. Introduction

Alcator C-Mod is the high-field, high-density divertor tokamak in the world fusion program. The overall theme of the Alcator program is

Compact high-performance divertor tokamak research to establish the plasma physics and plasma engineering necessary for an ignited tokamak experiment and for attractive fusion reactors

The Alcator C-Mod research program is organized along two parallel, complementary paths: **Advanced Tokamak** quasi-steady state configurations with high bootstrap current fraction and RF current and flow profile control; **High field** high pressure regimes providing the physics basis for the compact, high field approach to ignition.

Unique aspects of the Alcator C-Mod facility provide the logical foundations for the scientific areas of emphasis in our research endeavors to answer key outstanding questions in the development of practical fusion energy:

- **Long pulse capability**—C-Mod has the unique ability among highly-shaped, diverted tokamaks, to run high pressure plasmas with pulse length equal to the L/R relaxation time, at $B_T > 4$ Tesla. This provides an outstanding opportunity to investigate the extent to which enhanced confinement and stability of Advanced Tokamak configurations can be maintained in steady-state, using active profile control.

- **High magnetic field**—With capability to operate at very high absolute plasma densities and pressures, and with magnetic field up to 9 Tesla, C-Mod offers a unique test-bed for exploring the physics and engineering which is prototypical of possible compact ignition experiments.

- **Exclusively RF driven**—C-Mod does not use beams for heating, fueling or momentum drive. As a result, the heating is decoupled from particle sources and there are no external momentum sources to drive plasma rotation. It is likely that the same constraints will obtain in a fusion power plant; the studies of transport, MHD and AT physics in C-Mod are highly relevant to reactor regimes.

- **Unique dimensional parameters**—C-Mod is dimensionlessly comparable to larger tokamaks, but dimensionally unique, which allows us to provide key points on scaling curves for confinement, H-Mode threshold, pressure limits, etc. At the same time, coordinated experiments with other facilities allow for important tests of the influence of non-similar processes, including radiation and neutral dynamics.

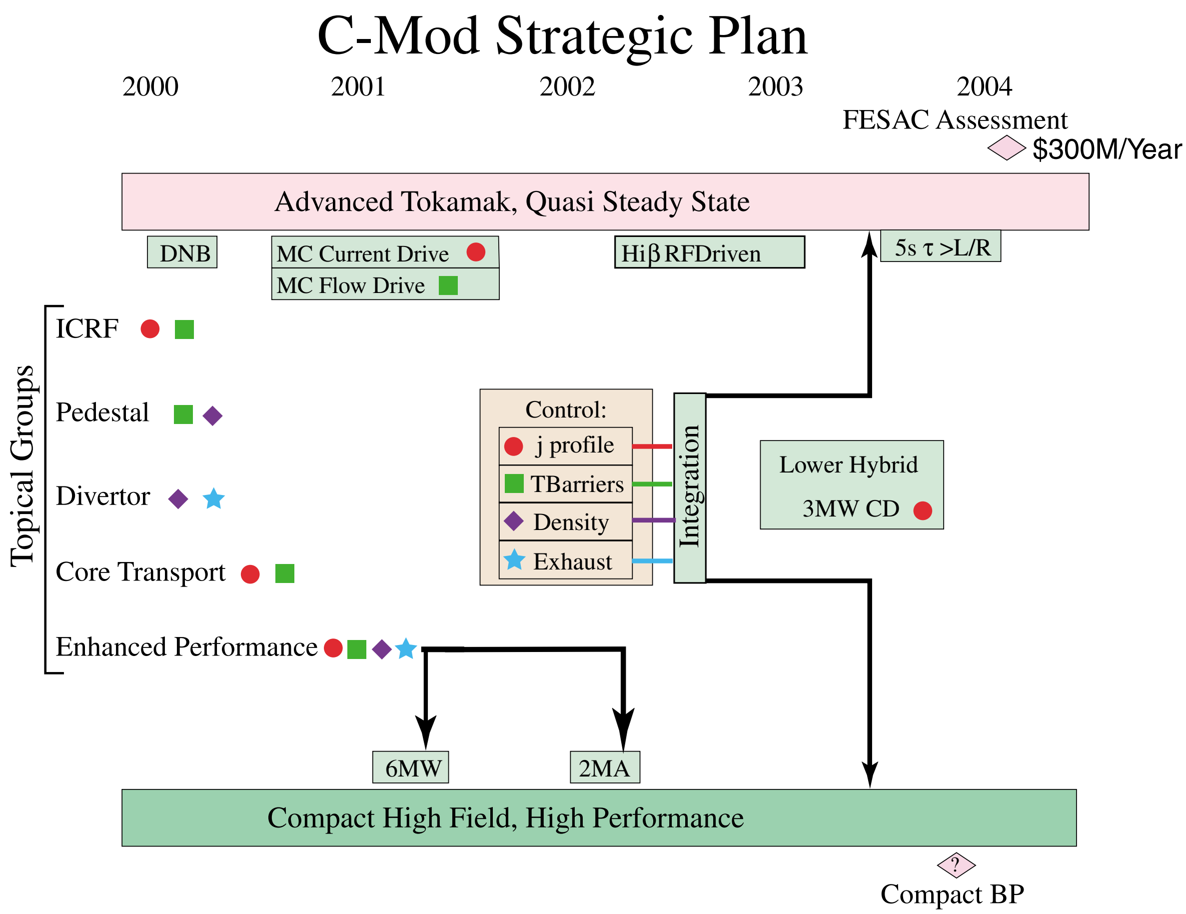
- **Very high power density scrape-off layer plasma**—With parallel power flows approaching 1 GW/m^2, C-Mod accesses unique divertor regimes which are prototypical of reactor conditions. The issues of edge transport and power handling which are explored go beyond those specific to the tokamak, being relevant to essentially all magnetic confinement configurations.

- **High Z metal walls**—The molybdenum plasma facing components on C-Mod are unique among the world's major facilities. The use of high Z PFC's is also reactor prototypical, and leads to unique recycling properties, and density and impurity control challenges.

Education is an integral part of the Alcator project mission, and the project has a large contingent of graduate students working toward their PhD degrees. They are drawn from multiple departments at MIT, as well as from collaborating Universities.

The connections among the topical science areas and overall programmatic thrusts are illustrated in Figure 1.

Fig 1.



Links to the FESAC MFE Goals

The FESAC has endorsed four high level goals for the Magnetic Fusion program in the US: 1) Advance fundamental understanding of plasma, and enhance predictive capabilities, through comparison of well-diagnosed experiments, theory and simulation; 2) Resolve outstanding scientific issues and establish reduced-cost paths to more attractive fusion energy systems, by investigating a broad range of innovative magnetic confinement configurations; 3) Advance understanding and innovation in high-performance plasmas, optimizing for projected power-plant requirements, and participate in a burning plasma experiment; 4) Develop enabling technologies to advance fusion science, pursue innovative technologies and materials to improve the vision for fusion energy, and apply systems analysis tools to optimize fusion development.

The Alcator program contributes to all four of the goals, with our strongest efforts concentrated on goals 1 and 3. For goal 1, Figure 2 gives a graphical representation of the mapping between specific C-Mod program components and the 5-year objectives identified by FESAC for this science goal. Note that our program targets specific scientific contributions, and many of our initiatives address overlapping topics. Regarding FESAC goal 3, the two main thrusts of the C-Mod program, quasi-steady state Advanced Tokamak and Compact High-Field investigations are focussed on addressing the 5-year objectives related to Steady State, High Performance and Burning Plasma, as illustrated in Figure 3. Concerning goal 4, the C-Mod program focuses attention in selected areas: ICRF and Lower Hybrid technologies, and high Z metal walls/divertors with reactor level heat flux. The Advanced Tokamak is an innovative concept that is a critical part of the broad range emphasized in goal 2.

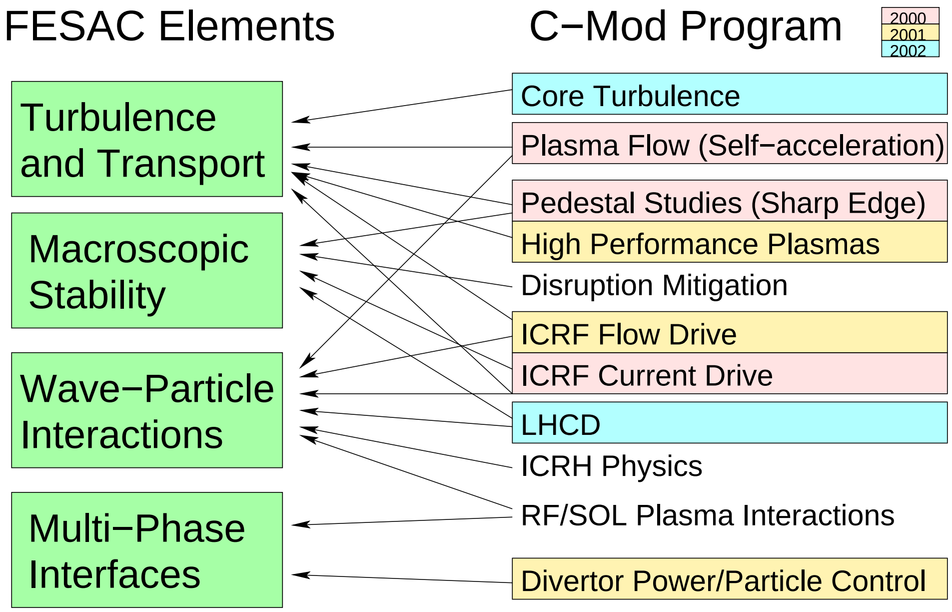


Figure 2. Mapping between our program and FESAC Goal 1 objectives

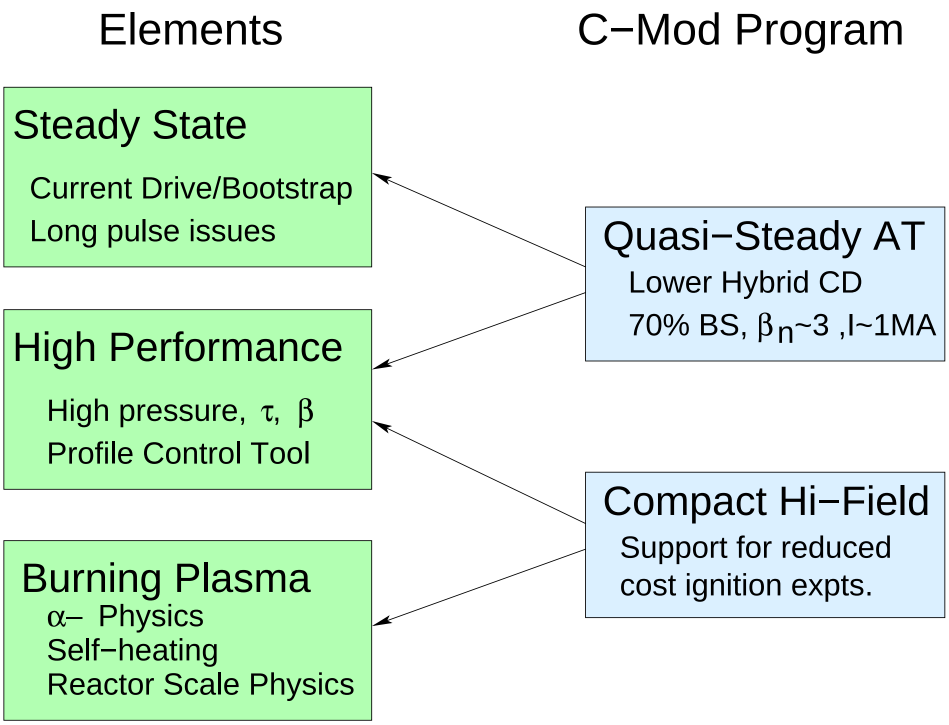


Figure 3. Mapping between our program and FESAC Goal 3 objectives

Budget and Schedule

The Alcator C-Mod program consists of work performed by the MIT Alcator group who originally proposed, designed, built and commissioned the tokamak, plus significant collaborations from different institutions. The present Work Proposal covers in detail the MIT responsibilities in the program, but assumes an integrated effort involving all the major contributors, PPPL and the University of Texas at Austin, plus a large number of smaller national and international collaborations at levels outlined in the overall budget summary.

The baseline (A) budget for the C-Mod project has been determined by guidance from the DoE Office of Fusion Energy Sciences, assuming flat funding in FY2002 and FY2003 relative to the present funding in FY2001, having total project funding of $17.8M, including $14.4M at MIT, and major collaborations totalling $2.4M at the Princeton Plasma Physics Laboratory and $0.91M at the Fusion Research Center, University of Texas, Austin. We also propose a higher, B budget, totalling $20.2M in FY2002, with a 6% increment in FY2003. The reductions from FY2000 to FY2001 have already caused serious dislocations to the research program, including a reduction of run time from 18 to 12 weeks, and the inability to replace personnel who have retired or departed for other employment. In addition to normal personnel cost of living increases, the project faces substantial increases in the costs of liquid nitrogen and electricity. As a result, the consequences of the flat budget in FY2002 include: further reduction of run time, projected to only 10 weeks in FY2002; layoffs of scientific, engineering and technical staff; deferral of needed upgrades including diagnostics, active cryopumping and data acquisition. Assuming continued flat funding in

FY2003, the required periodic inspection of our alternator and flywheel exerts even more pressure on the budget, making it difficult to operate the experiment productively that year. Other consequences include: further staff reductions, reduction in the number of graduate students, and deferral of all upgrades except for completion of the Lower Hybrid Current Drive M.I.E. initiative. It will also be impossible to begin building the second lower hybrid launcher, which is required to bring our current drive capability for Advanced Tokamak studies to full power. The incremental (B) budgets, with corresponding increments at PPPL and the University of Texas F.R.C., will allow for a healthier research program, increased run time, a timely completion of the most important upgrades, and a start on the second Lower Hybrid launcher, along with very modest staff additions to ensure more effective use of the campaigns. The target dates associated with achieving our plain English goals, as well as with the more detailed milestones set out in Chapter 5, have been established assuming that the project is funded at the B-budget levels.

Proposed tokamak run-time, assuming the A budget levels, is

Fiscal Year	01	02	03
Operating Weeks	12	10	8
Operating Hours	400	330	250

Under the incremental, B, budgets, run time would increase to 14 weeks in FY02 and to 18 weeks in FY03. A summary of the planned facility schedule, assuming the guidance, A, budget levels, is shown in Figure 4.

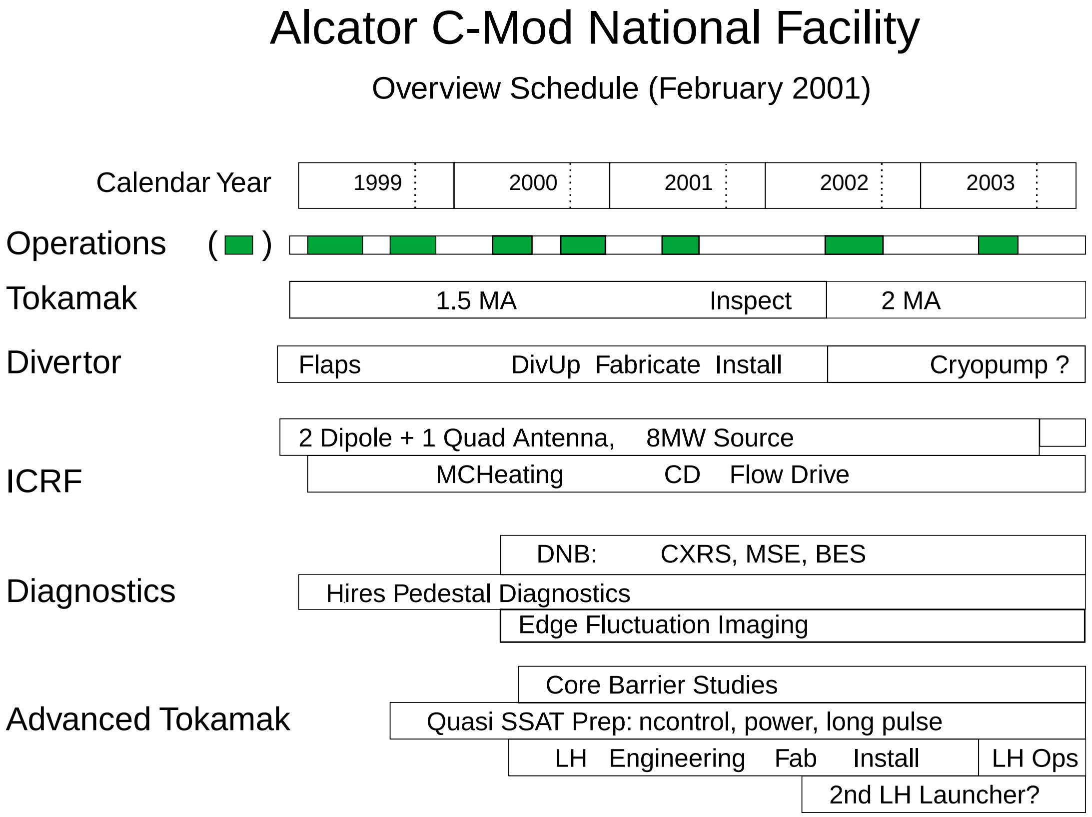
Research Goals in Plain English

In order to communicate the excitement of plasma fusion science to a wider audience, each year we develop research goals, expressed in non-technical language, which reflect some highlights of our program plans. These goals do not represent the full scope of our program; the detailed technical research plans are described by topical science area in the succeeding sections of this proposal.

Active formation and Control of Internal Transport Barriers [Aug 01]

Plasmas with Internal Transport Barriers show promise as the goal for future advanced tokamak steady state reactor operation. Plasmas with simultaneous edge and core energy and particle transport barriers have been formed in Alcator C-Mod with auxilliary radio wave heating. This is accomplished without the need for neutral beam particle and momentum sources, which have been required to achieve these core barriers on other tokamak experiments, and which are likely to be unavailable in tokamak reactors as they are presently envisioned. The prescription for achieving the core barrier is to lower the magnetic field, which causes the radio waves (at 80 MHz) to be concentrated in the inner portion of the plasma, a significant distance away from the hottest reacting core of the

Fig 4.



plasma. Further heating power at 70 MHz will be concentrated at the core of the plasma to increase the temperature of the plasma center, inside the barrier region, to hold the density profile steady and to arrest impurity accumulation. The fundamental role of plasma rotation in the internal barrier formation will also be investigated.

Plasma Probing with Energetic Neutral Particles: Critical Measurements [Aug 01]

By injecting energetic neutrals into the plasma, a wealth of new information will be gathered on (a) the profiles of ion temperature and ion flow which are important for plasma confinement, (b) the plasma current profile which determines the macroscopic stability of the plasma, particularly at high pressure, and (c) plasma density fluctuations, which are responsible for the turbulent transport that ultimately determines the quality of confinement.

Plasma Flow Control with Radio Waves [Sep 01]

A crucial part of control of transport is the control of the flow that helps to stabilize the responsible turbulence. Theoretical studies suggest that radio waves of the type used for heating Alcator C-Mod can control the plasma flow. We will complete the first experiments to verify, using our new diagnostics and the high power RF, what degree of control is possible, and how this can be used to optimize the plasma confinement.

Driving Electric Current with Radio Waves [Sep 02]

For steady-state operation, which is attractive for a reactor, it is necessary to drive current in the plasma with waves, not just with DC electric fields. A new method of driving the current involves launching waves in such a way that they are converted by interaction with ion resonances inside the plasma from long wave-length to short wave-length. They then drive the electrons of the plasma, creating a current. The first round of C-Mod experiments on this scheme will be completed, establishing its efficiency and suitability for the future.

Higher Performance Plasmas [Sep 02]

Significant modifications and upgrades to the inner divertor structures in Alcator C-Mod have been designed and will be implemented in Fiscal Year 2001. The hardware will be strengthened to withstand the forces resulting from the halo currents expected during full current disruptions, and the present highly shaped inner wall structure will be replaced with a flatter, more open design, which will allow for the investigation of a broader range of plasma shapes, especially those with increased plasma triangularity. These changes will open up new regimes of plasma operation on Alcator C-Mod, including the ability to study plasma behavior and performance at the maximum currents and plasma pressures available, with maximum flexibility to detail the effects of the full range of shaped plasmas which can be confined and heated in the device.

Power and Particle Handling for Advanced Tokamak Plasmas [02]

Techniques for safely radiating away the extremely large parallel heat flow encountered in magnetic confinement plasma exhaust have been demonstrated at relatively high density. Quasi-steady state Advanced Tokamak plasmas may require lower density and involve techniques that are constrained by the needs of optimizing confinement. We will establish the limits of the divertor techniques and their performance in regimes appropriate for these plasmas.

Visualization of Turbulence [02]

The plasmas in our experiments exhibit strong turbulence, chaotic behavior which can spoil confinement by transporting energy and particles. The understanding of turbulence, which defies straightforward analysis, is perhaps the greatest challenge left in classical physics. Our goal is to attempt to study this process by imaging fluctuations in the edge and core of our plasmas. The measurements will be compared to results of simulations run on powerful supercomputers and should provide insight into the origins and effects of turbulence in fusion plasmas.

Commissioning of the Microwave Current Drive System [03]

Theory and past experiments show that microwaves launched as so-called Lower Hybrid waves can be used to drive toroidal plasma currents with high efficiency, and that these currents can be localized radially. Importantly, hollow current profiles can be formed which lead to improved stability, higher plasma pressures, and nearly steady state "Advanced Tokamak" operation. To pursue this research on Alcator requires the installation of a microwave transmitter system and an appropriate launcher. We plan to complete this engineering and commence Advanced Tokamak experiments before the end of FY 2003.

Current Profile Control with Microwaves [04]

In FY2003 we will begin experiments aimed at developing efficient steady-state tokamak operation by launching microwaves into Alcator C-Mod plasmas. Up to 3 MW of RF source power will be available at a frequency of 4.6 GHz. This frequency is characteristic of the geometric mean frequency of ion and electron gyrofrequencies, the frequency at which these particles gyrate around the lines of magnetic field. It is known as the lower hybrid frequency and the launched waves are known as lower hybrid waves. The location of current driven by lower hybrid waves depends on their wavelength as measured parallel to the magnetic field. We intend to vary this wavelength and to measure the location and amplitude of the driven current. By adding plasma heating due to ion cyclotron waves, the plasma pressure will be raised, and by varying the location of the RF-driven current, we can begin to investigate the stability limit of the plasma, i.e. the maximum pressure the plasma can sustain without developing gross modes of instability, as well as other properties of the plasma such as its ability to confine energy.

Goal Accomplished in FY2000

Goal *Understanding the Sharp Edge of the Plasma*

Tokamak plasmas spontaneously generate "transport barriers" which substantially reduce the loss of energy and result in high confinement. When such a barrier forms, the steep pressure gradients that result can give rise to a variety of instabilities. While some instabilities are deleterious, others are benign or even beneficial, as in the EDA regime discovered in C-Mod, which combines favorable high energy confinement with sufficient particle transport to maintain plasma purity. We will continue to exploit the recently-installed high-spatial-resolution diagnostics to study the underlying physics which governs the formation and evolution of these narrow regions of good confinement.

Report:

Detailed measurements of the quasi-coherent fluctuations seen in the edge density, potential, and magnetic field and a series of theoretical and computational studies have supported the hypothesis that the EDA H-mode is a manifestation of the resistive ballooning mode. Detailed analytic work by Hastie, on the Carreras-Diamond branch of this mode has found a threshold with $q^2\nu/\sqrt{m_i} > 250$ for strong growth. This expression agrees with the measured dependences on q, $\nu*$, and mass as well as the mode location. Electromagnetic drift-ballooning simulations performed by Rogers and Drake (U.Md.) had previously showd the possibility of a strong coherent mode developing non-linearly. A similar coherent mode was seen in non-linear resistive MHD simulations of Garcia when the pressure gradient was just above the instability threshold. Most recently, calculations with the gyro-kinetic stability code gs2 performed at MIT in collaboration with Dorland of U. Md. used measured plasma profiles and the calculated MHD equilibrium. Results showed dependences similar to those found in the experiment and supported the basic hypothesis. The work is being extended to include the effects of real geometry including the separatrix and x-point by Xu and Nevins using the BOUT code from LLNL.

2. Alcator C-Mod Research Areas

2.1 Transport

Connection to Integrated Program Planning Activity

Transport research on C-Mod is well aligned with the fusion sciences mission and goals as outlined in the IPPA report of November, 2000, namely 1.1 "Advance scientific understanding of turbulent transport forming the basis for a reliable predictive capability in externally controlled systems." Most of the transport related scientific issues outlined in appendix III are addressed directly by the C-Mod program.

> *What is the theoretical and experimental basis for local control of tokamak transport including the ability to predict the position, strength and dynamics of transport barriers?* Experiments on ICRF-induced ITBs and EDA H-mode address the issue of control of local transport, particularly the character and dynamics of transport barriers.
>
> *What are the effects of finite-beta and confinement geometry on transport?* Measurements of the very strong magnetic component to the quasi-coherent oscillation seen in EDA H-modes along with a significant magnetic component observed in the L-mode plasmas indicate an important role for finite β and β' in edge transport. With available increases in heating power, we are also beginning to see some β related effects in the core plasma. As the RF power is raised, these effects will grow in importance. The modification of the inner divertor will enable the study of a wider range of plasma shapes. Recent experiments have shown a strong triangularity dependence on the EDA/ELMfree boundary.
>
> *What are the mechanisms responsible for anomalous electron thermal transport?* We will begin by documenting regimes in which the electron channel dominates transport, looking for evidence of critical gradients and comparing these with theoretical calculations. Experimental profiles will be used to calculate the expected spectrum of short wavelength fluctuations and optimize a CO_2 scattering experiment currently under design.
>
> *How does the power threshold for internal transport barriers scale with gyroradius in the absence of externally driven rotation?* With its ability to create transport barriers without the use of neutral beams, C-Mod is in a unique position to explore this issue.
>
> *What is the fundamental origin of the observed density limit on tokamak operation?* Recent experiments on C-Mod, described below, suggest that the density limit is due to changes in edge fluctuations and anomalous perpendicular transport which occur as the density is raised. Further studies will attempt to uncover the physical mechanism responsible for this change in transport.
>
> *Can RF waves maintain and control desirable confinement in long-pulse or reactor-scale plasmas?* The C-Mod program is approaching this problem on several fronts. We are investigating the role of heating and flow drive in creating and sustaining the ITBs generated by off-axis ICRF. We are planning to study the effects of current profile modification with mode-converted current drive and the effects of poloidal flow drive with mode-converted IBW. Later, we will be using the lower hybrid system to create strongly modified q profiles and study their role in providing access and control

of enhanced confinement regimes.

How does neutral hydrogen recycling affect stability and transport? C-Mod has been using it's unique set of edge diagnostics which include high-resolution Ly_α arrays to measure the local neutral density and ionization rate in the plasma. With our theory collaborators, we are studying the role of neutral density in modifying energy, momentum and particle transport.

What is the influence of the plasma edge on the plasma core and on the global properties of confined plasma? Past work has clearly shown evidence of critical temperature gradient lengths. The manifestation of this behavior is self-similarity of the profiles, where an increase in the edge temperature leads to a proportional (rather than additive) increase in the temperature profile everywhere. Current work emphasizes the quantitative comparison between the measured gradients and those predicted from theories of ITG turbulence.

What determines the amplitude and width of edge pedestals in plasma pressure and temperature? Studies in this area have been enhanced by the recent addition of a high-resolution edge Thomson scattering system and have included scaling studies with plasma current, input power, and plasma shape. Studies of the EDA H-mode emphasize the role of a quasi-coherent oscillation in determining pedestal transport and profiles.

Recent Accomplishments

Recent work in edge transport has focussed on the micro and macro stability of the pedestal. In particular, we have investigated mechanisms which may be responsible for the EDA (Enhanced D_α) regime. EDA discharges have only slightly lower energy confinement than comparable ELM-free ones, but show markedly reduced impurity and particle confinement. Thus EDA discharges do not accumulate impurities and typically have a lower fraction of radiated power. The edge gradients in EDA are relaxed by a continuous process rather than an intermittent one, as is the case for standard ELMy discharges, and thus do not present the first wall with large periodic heat loads. The EDA is clearly associated with a quasi-coherent (QC) mode which is always present in such discharges. Particle diffusivity, $D_{eff} \equiv \Gamma/\nabla n$, measured at the separatrix was found to increase in proportion to the amplitude of the QC mode which is observed by phase contrast imaging (PCI), reflectometry and probes. Earlier experiments with a fast scanning Langmuir probe, which measures both density and potential fluctuations, showed that the transport driven by the fluctuating quantities was sufficient to account for the macroscopic changes which were observed. These results support the observation that the EDA regime encompasses a continuum, from those with low levels of the QC mode and transport indistinguishable from ELMfree, to others with very strong levels of the fluctuation and significantly higher particle transport.

Previous work has shown that the MHD safety factor, q_{95} and triangularity, δ, are critical parameters for establishing the boundary between the EDA and ELMfree H-mode regimes. EDA typically occurs for $\mathrm{q}_{95} > 3.5$-4 and with $\delta > 0.3$. Experiments with an upper null divertor configuration suggest that the triangularity dependence is an intrinsic plasma shape effect, and is not due to a change in plasma-divertor interaction. The density

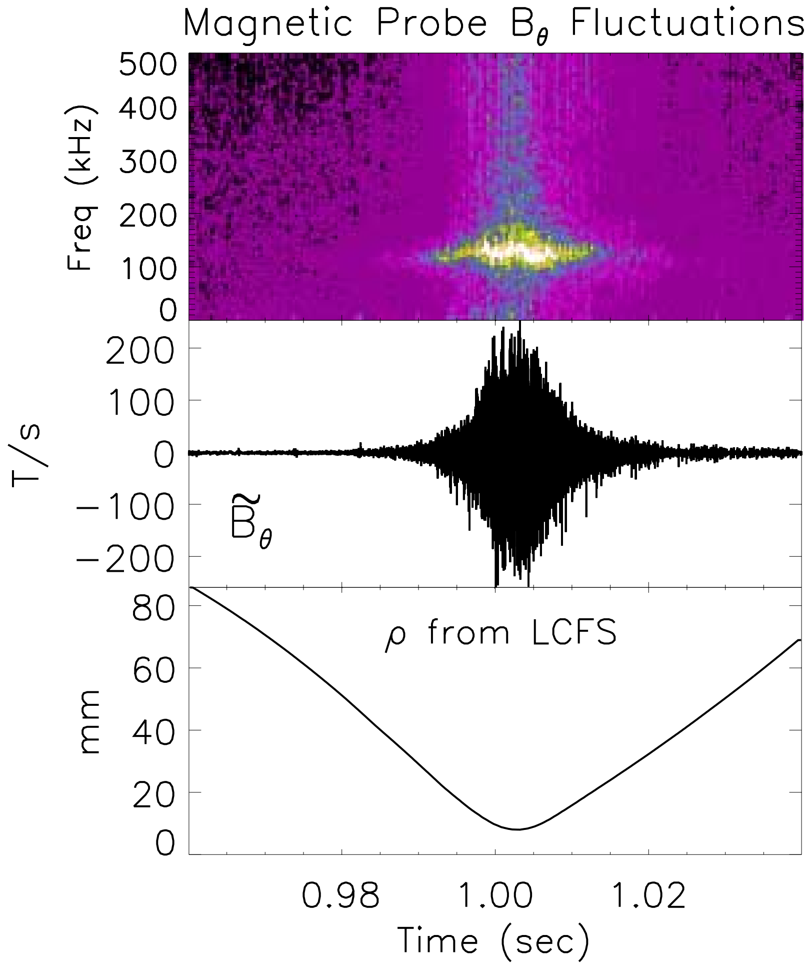


Figure 5. During EDA H-modes, large magnetic signals are obtained when a pick-up coil is scanned within a few cm of the plasma edge.

of the L-mode target turns out to be an important parameter as well, with QC mode amplitude increasing at higher densities. EDAs are not seen for target densities below $1 \times 10^{20}/m^3$, suggesting a role for collisionality in the mechanism which drives the mode. The mode has been localized to the lower portion of the radial density profile, close to the separatrix. The radial width of the mode is uncertain, but there are data suggesting that it could be as narrow as 1-2 mm. To further investigate the character of the QC mode, a scanning probe was fitted with a magnetic coil. When the probe is inserted into the edge of an EDA plasma, large coherent magnetic fluctuations are observed at the same frequency as the density and potential fluctuations measured by other diagnostics. The estimates of the perturbed current density range from 0.1-0.35 × the local equilibrium current density, with uncertainties coming from the unknown radial mode structure. Thus the QC mode is a strongly electromagnetic phenomenon. The standard coils on the vacuum vessel wall are too far from the pedestal to detect the QC mode, due to its short poloidal wavelength, which is in the range of 2-4 cm. The mode wavenumber, k, has been found to be proportional to the magnetic field, so that $k\rho_s$ is approximately constant. It is not known whether the magnetic field has a direct effect on the mode, or an indirect effect through the plasma profiles.

The existence of the mode in regions of strong pressure gradient and high resistivity suggests that the linear instability which drives the quasi-coherent mode could be resistive ballooning. This mode is more unstable at higher q, which is consistent with our observations. This tentative identification suggested an experiment to test the scaling of the EDA boundary with isotope mass. The findings were consistent with the theory for resistive ballooning: hydrogen plasmas were in EDA at values of q where deuterium plasmas were found to be ELMfree. Detailed analytic work on the Carreras-Diamond branch of resistive ballooning by Hastie of Culham laboratory has shown a threshold with $q^2\nu/\sqrt{m_i} > 250$ for strong growth of the mode. This expression agrees with the measured dependences on q, $\nu*$, and mass, as well as with the mode location. Previous work by Rogers and Drake (U.Md.) showed the development of a strong coherent mode which was identified as a collisional ballooning mode. In collaboration with Dorland of U. Md., we have begun calculations with the gyro-kinetic stability code GS2 using measured plasma profiles and the calculated MHD equilibrium. The code includes the full electron dynamics and electromagnetic terms. We are also working with Xu and Nevins from LLNL and Myer from Lodestar on simulations of the C-Mod edge with the BOUT code. This code has the real geometry, including the separatrix and x-point, and early results suggest that these may be important.

The new edge Thomson scattering diagnostic obtains routine measurements of both density and temperature profiles in the pedestal region, with 1 mm radial resolution. This has allowed unambiguous measurements of the pressure profile and comparisons of the widths and detailed structure of various electron profiles. All H-mode profiles show a very steep region of width 2-4 mm in the outer region, with typical gradients $\nabla T_e \sim 50keV/m$ and $\nabla n_e \sim 5 \times 10^{22} m^{-4}$. A new discovery is that there is often a region of moderately increased gradient, still above the core gradients, but less steep than those in the outer region; this is particularly apparent in the T_e profiles.

While it is now widely recognized that the conditions at the top of the pedestal set the boundary condition for core transport and can indeed largely determine the core thermal gradient (see below), there is not yet a good theoretical or empirical understanding of what sets the width or height of the edge transport barrier. This is critical for the design of next step machines, whether of compact high field, conventional lower field or advanced tokamak designs. The need for better predictions motivates studies of the scaling of pedestal parameters. Somewhat surprisingly, pedestal profiles in EDA and ELM-free discharges with similar global parameters were not markedly different. We also found, in contrast to results on DIII-D and JT60-U, but consistent with those on ASDEX-Upgrade, that pressure profiles do not get narrower at higher plasma current. This does not support a scaling with ρ_{pol}. The peak electron pressure gradient, however, does scale as I_p^2, or alternatively $I_p^{1.2}\nu*_{ped}^{-0.4}$. The width of the density pedestal increases with triangularity.

We are continuing studies of effects of neutrals on edge transport. Exploiting a unique set of diagnostics including a pair of Ly$_\alpha$ arrays and a high resolution edge Thomson scattering system, we are able to calculate the neutral density in the plasma edge as well as the local ionization source. The Ly$_\alpha$ arrays view tangentially at the inner and outer midplanes. These show a strong poloidal asymmetry with roughly a factor of six higher emission from the inside edge. The results of these measurements have been compared directly to calculations from a new neutral transport code written at MIT.

Density limit studies have focussed on the role of turbulent transport. A fast-scanning electrostatic probe was used to measure profiles and fluctuations in the plasma edge as the density was raised to the density limit. The edge plasma shows two distinct regions, a near scrape-off layer with steep temperature and density gradients and high frequency fluctuations, and a far scrape-off layer with flat profiles and large long-correlation fluctuations. As the density is raised, the transition point between these two regions moves inward toward the separatrix, eventually intruding onto plasma on closed field lines. Not long after this, the plasma experiences a density limit disruption. These experiments suggest that the density limit is the result of the domination of anomalous perpendicular transport over classical parallel transport at very high collisionality.

Previous experiments on C-Mod have shown that the normalized temperature gradient, R/L$_T$, is close to constant, varying only from 5 to 8 over a wide range of input powers and confinement regimes, a result anticipated by theoretical considerations. C-Mod is expected to be strongly tied to the critical gradient, as the power which can be carried by gyrobohm turbulence is large compared to available heating power. This result is confirmed by transient transport experiments using the recently commissioned high resolution heterodyne ECE system. A strong increase in the incremental diffusivity, χ_{HP}, was found as the input power was raised, even though R/L$_T$ was nearly constant. However, calculations of the theoretically expected temperature gradient length, R/L$_{Tcrit}$, (using ITG models, for example IFS-PPPL) found values which were significantly smaller than those in the experiment. Working with Dorland of U. Md, we have explored the origins of this discrepancy using the non-linear version of the gyrokinetic stability code gs2. A non-linear upshift in the normalized critical gradient is found, apparently due to the generation of zonal flows which are believed to be the principal saturation mechanism for ITG turbulence. This upshift, which may reconcile experiment and theory, tends to disappear at lower collisionality, suggesting an important non-collisional damping mechanism for the zonal flows. This effect is due to the proper non-adiabatic treatment of electron dynamics. Treating electrons adiabatically leads only to collisional damping of the zonal flows.

Four types of internal transport barriers (ITB) have been observed on C-Mod. 1.) P-modes or PEP-modes, generated by pellet fueling, create a self-sustained peaked-density plasma. The bifurcation is believed to be driven by the stabilization of the ITG modes that occurs when $R/L_n \simeq R/L_T$, that is when η_i drops to values near one. Reversed magnetic shear, which is measured in these discharges, may also play a role in sustaining the regime. 2.) Short-lived ITBs are generated following nearly every H/L back-transition in C-Mod. These may derive from the same mechanism as the P-modes, where, in this case, the steep density gradient is obtained from the decay of density in the outer regions of the plasma following the H/L transition. These barriers are typically destroyed by sawtooth activity after about one τ_E. Stabilizing flows driven by ∇p may also play a role in creating or sustaining the barrier. 3.) Ohmic H-modes often show a spontaneous density peaking which can last for the duration of the current flat-top, up to 10 $\times\tau_E$ so far. 4.)

With strong off-axis high field side ICRF heating, H-modes with peaked density profiles are produced and sustained for up to 10 $\times \tau_E$. Toroidal rotation, which is normally in the co-current direction, is suppressed and then reversed. The source and role of sheared flows in this regime are under investigation. It is not clear what physics determines the barrier location and extent, as the foot of the barrier does not coincide with the position of the RF resonance or the q=1 surface. These are sawtoothing discharges, so it is believed that the q profile is monotonic, with q_{min} at the axis. All of the ITB modes listed above show peaked density profiles, with sharply reduced particle diffusivity in the core. Analysis with the TRANSP code has shown that the energy diffusivity is also strongly reduced in the same region, as illustread in figure 6.

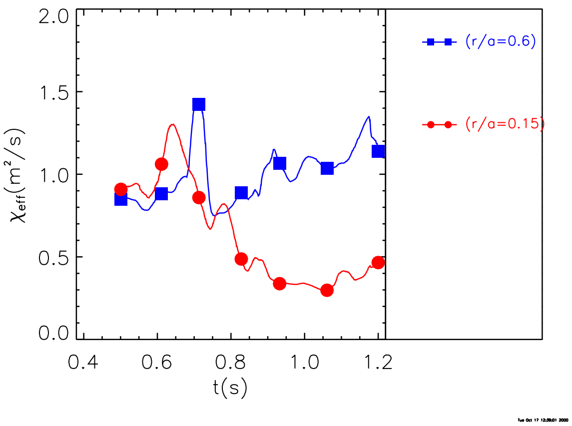


FIG. 6(a) Thermal diffusivity vs time comparing radii inside and outside the transport barrier. A clear transition is seen at 0.8 sec

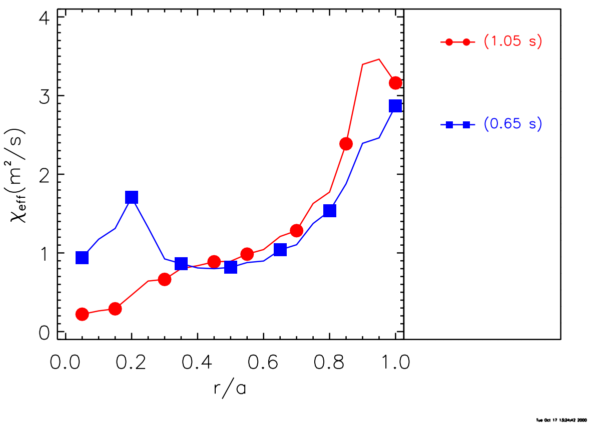


FIG. 6(b) Thermal diffusivity vs radius comparing times before and after the transition. The region of improved energy confinement is localized to about 0.3 r/a

Research Plans

While EDA H-mode has characteristics which would be favorable in a fusion reactor, the prospects for extrapolation depend on a better fundamental understanding. The key to that understanding lies in the quasi-coherent oscillations which drive enhanced particle transport in EDA. New diagnostics and new analysis tools should allow us to test various hypotheses for the underlying physics of the EDA H-mode. Higher frequency reflectometer channels will be added allowing more complete coverage of the H-mode pedestal and may help in localization of the QC mode. We are building a probe head with two pickup coils to measure the wavelength of the mode directly. We will try to image the QC mode with a fast edge camera. The divertor upgrade will allow higher triangularity plasmas, enabling further studies of the shape dependence for the EDA/ELMfree boundry. We will use the electro-static probes and the heterodyne ECE system to look for a $\tilde{T}_e$ component to the QC mode. We will be collecting more fluctuation data and improving analysis methods in order to make quantitative comparisons between the particle flux driven by the QC mode and that calculated from edge particle balance. The latter will require additional profile and

Ly$_\alpha$ data. In collaboration with the JET group, we have performed a set of preliminary experiments looking for EDA in that device. Further experiments are planned during the current JET operating period. We have also begun discussions with DIII-D, ASDEX upgrade and TCV to search jointly for EDA and the QC mode on those experiments. A mini-proposal, written jointly with DIII-D, has been scheduled for their current run period.

We will continue our strong collaborations with theory and modeling groups working on the EDA problem. The analytic work on resistive ballooning will be extended to include the role of plasma shaping, shear flows and temperature equipartition. As this work suggests that the mode should be resonant on rational q surfaces, we will carry out experiments to look for this effect. We will complete parameter scans with the linear version of gs2 and begin non-linear runs. These should allow us to assess the role of shear flow stabilization and the implications for particle and energy transport. The work with the BOUT code will continue, focussing on the role of the separatrix and x-point. We will look for EDA in limiter H-modes in order to benchmark these calculations. We will investigate the macro-stability of the pedestal, looking at pressure driven modes with the BALOO code and current driven modes with ELITE - a collaboration with Wilson of Culham Laboratory. It is worth noting that, for $\beta_N > 1.2$, small and inconsequential ELMs are seen on top of EDA in C-Mod. Finally, in collaboration with Carreras of ORNL, we will test an alternate theory for the EDA/QC physics based on the self-organized transport of systems with transport barriers.

In future campaigns, systematic studies of pedestal widths and height will be continued and extended over wider parameter ranges, in particular to higher input power, which may change the edge stability regime. Core and edge profiles will be combined to allow analysis of the moderately improved transport region, which sets the boundary for core confinement, as well as of the steepest region, which is critical for edge stability. Results will be compared with those from other machines, in order to establish the critical size scaling. This will be done through inter-machine experiments with DIII-D and ASDEX-Upgrade as well as contributions to, and use of, the international pedestal database. We will also attempt to compare with theories and numerical models of the transport barrier width as these become more fully developed.

Studies of the L/H transition will emphasize local measurements correlated to the threshold. While there is roughly a factor of two in power between the L/H and H/L transitions, previous work showed a clear threshold in T$_e$ with no hysteresis, suggesting that the local temperature or its gradient is a key parameter. We will repeat these experiments with our improved edge profile diagnostics, looking for other critical variables. Comparisons with theory will emphasize the role of shear-flow suppression of drift ballooning turbulence. The role of neutrals will be investigated. Using our measurements of neutral density, we will study the role of neutral collisions in flow damping and energy transport. Exploiting these measurements should also shed light on the relative importance of ionization source, as compared to transport, in the enhanced fueling observed after an L/H transition. Experiments at low density and with helium plasmas should allow us to assess the importance of the ionization source profile in determining the H-mode pedestal width. We will study the implications of the source asymmetry and compile a complete data set for the 2D modeling effort carried out in collaboration with ORNL. Finally, we will investigate the dynamics of the L/H and H/L transition, comparing the evolution of edge profiles to various bifurcation models in collaboration with Carreras of ORNL.

In the plasma core, we will continue our analysis and comparison between measured and calculated critical gradient lengths. A series of high k-resolution GS2 runs will be performed to verify preliminary results, particularly the role of non-adiabatic electrons and the dependence of the non-linear upshift on collisionality. Using the MSE diagnostic made available by the DNB, we will look for the effect of magnetic shear on the upshift. Predictions are that strong shear leads to a larger upshift, however the estimates of the q profile from EFIT and TRANSP are not accurate enough to assess this effect without additional measurements. The DNB will also allow improved measurements of ion temperature and rotation, enabling us to extend our analysis farther out in the profile. We will continue the transient transport experiments, particularly with modulated ICRF running in the mode-converted IBW regime. This allows for direct electron heating, obviating the slower transient response due to ion-electron coupling in minority heating experiments. The aim is to verify the "marginal stability" relationship between heat flux and temperature gradients. We will also run more cold pulse experiments and compare the results to our model calculations.

In advance of the lower hybrid current drive experiments, support for the AT program will emphasize the creation, characterization and control of internal transport barriers. C-Mod, with its high density and strong ICRF as the only heating scheme, presents a unique testbed for these studies, with equilibrated electrons and ions, no direct momentum input and no continuous central fueling source, a regime prototypical of a tokamak reactor. Most ITB studies in other devices have been carried out in the opposite regime.

We will attempt to answer a number of questions raised by the results with ITBs in C-Mod. First, we will try to determine if a clear transition to the improved confinement regime can be identified in all cases. We will look for evidence of hysteresis and use core fluctuation diagnostics to search for bursting behavior - both evidence of a bifurcated regime. We will assess the role of power, fueling and flow drives in causing the ITB and look for thresholds in these drive terms. We will examine the causes and effects of rotation in initiating and sustaining core barriers. Strong central heating of the off-axis ICRF induced barriers should allow assessment of the relative improvement in particle vs. energy confinement. For all these, the improvements in core diagnostics for profiles and fluctuations will be crucial. In addition to the DNB related diagnostics mentioned above, we will be adding more channels and a second laser to the Thomson scattering system, to increase the spatial and temporal resolution of that instrument. We will use the GS2 code to follow the evolution of the growth rates of the relevant instabilities and to assess the relative importance of the various transport channels. In an attempt to sustain and optimize the ITB modes, we are planning to use mode converted ICRF current drive to modify the q profile sufficiently to suppress sawteeth. We will try to understand the localization of the barrier and determine if it can be controlled with the current set of tools. (Eventually the LHCD system will be used produce current profiles optimized for ITB regimes.) We will also try to produce enhanced confinement regimes by directly driving poloidal flow with mode converted IBW waves.

The C-Mod ITB regimes are particularly well suited for studies of particle transport. Because of the ICRF heating source, there is no core particle source to complicate analysis. The regimes are easy to obtain, but not universal, allowing direct comparisons between plasmas with and without barriers. Transport studies will include analysis of profile evolution to obtain transport coefficients, impurity transport studies with laser blow-off sources and modulated gas puff experiments. BES and PCI will be used to measure changes in fluctuations correlated with the production of the barrier. Non-linear simulations, with the full electron dynamics, will be used to compare the results to theory.

The BES and ECE radiometer systems will enable investigations of core turbulence, measuring density and temperature fluctuations respectively. The first step will be characterization of the turbulence in the various C-Mod operating regimes. Steady-state fluctuation parameters will be compared to predictions of turbulence-based theories, including correlation lengths and times, and turbulence decorrelations with ExB flow shear. These measurements will complement comparisons of transport with turbulence-based theories including the modulated heating experiments mentioned above. Spatial resolution should be adequate to diagnose changes in turbulence in the vicinity of internal transport barriers. We will test the dimensionless scaling of fluctuation parameters (fluctuation level, frequency and wave-number spectra, correlation lengths) in intra- and inter-machine scans (with DIII-D). We will carry out statistical analysis of quiescent fluctuation data to test SOC models. Data from the ECE radiometer may be particularly well suited to these studies. We will extend our studies of transport scaling with ν^* to lower collisionalities. These provide an indirect test of the effects of zonal flows, which are believed to be the main non-linear damping mechanism for turbulence. Since the flows are principally damped by ion-ion collisions, transport should decrease at lower collisionalities.

Electron transport will be studied by heat pulse experiments carried out at low density using both sawteeth and modulated RF as sources. The principal aim will be to obtain χ_{HP} vs. power and to measure the scaling of the critical gradient length. We will document electron transport dominated regimes, and use these as input for the GS2 and GENE codes, to predict the role of ETG and other short wavelength modes. Finally, these data will be used for the design of a high-k fluctuation diagnostic based on collective CO_2 scattering.

The CXR diagnostics will allow us to make measurements of the toroidal and poloidal rotation profiles across the entire minor radius and supplement the core measurements made by X-ray spectroscopy. An important goal is to understand the origin of the strong rotation which has been observed in the absence of direct momentum input. The complete profile measurements will allow us to determine if there is a specific RF drive which operates in addition to an as yet undetermined transport mechanism. With theory collaborators, we will try to evaluate the strength of the turbulent Reynold's stress and its contribution to the observed self-generated flow. Completion of the V_ϕ, V_θ, n_I, and T_i profiles made possible by the CXR will also allow us to compute the E_r and $\omega_{E\times B}$ shearing rate profiles. These can then be compared to calculations of instability growth rates calculated by the GS2 code. Finally, both old and new diagnostics will support measurements of plasma flow driven by mode-converted IBW waves. These experiments will attempt to create internal transport barriers by directly driving sheared flows as mentioned above.

2.2 Edge and Divertor Physics

Connection to Integrated Program Planning Activity

Edge and divertor research on Alcator C-Mod is well aligned with the FESAC goals contained in the IPPA report. We report recent results and future research later on in this section. Here we will delineate the particular areas of emphasis in terms of the IPPA goals.

A number of diagnostics are being utilized to provide characteristics of the SOL and across the separatrix. The information, provided in the form of profiles and turbulence measurements, are all important for determining the accuracy of transport models (*predictive capability* - IPPA 1.1.1) and more specifically to the region of the separatrix (*integrated models of core and edge physics* -IPPA 1.1.3; *coupling between edge and core plasmas* - IPPA 1.4.2).

The analysis of perpendicular transport in the SOL is aimed at a general understanding of *plasma edge physics* (IPPA 1.4.1). The focus is not only on characterizing the perpendicular transport, but in determining the underlying phsyics (*pursue fundamental research in plasmas* - IPPA 1.5.1). This has lead to a new undersatnding of the relative role of divertor and main chamber walls. In particular, it has been found that there is much more interaction with the walls (*plasma walll interaction* - IPPA 1.4.3) than previously thought. There are significant levels of particle and heat flows to the wall. The latter may be very important in understanding the density limit which reflects back on the goals - *coupling between the edge and core* - IPPA 1.4.2, and *disruption control/amelioration* - IPPA 3.2.4.

The strong cross-field transport of particles and heat appears to lead to wall impurity sources. The mechanisms for such impurity sources (e.g. sheath rectification and sputtering) and resultant transport to the core can be characterized as contributing to *plasma edge physics* - IPPA 1.4, and *general plasma science* - IPPA 1.5.1. We have varied the antenna limiter material from molybdenum to boron in an effort to evaluate the relative merits of these two materials. This is part of a general emphasis on improving the core plasma through improving materials used for the first-wall (*materials* - IPPA 4.4).

All of the above work is useful for development of so-called ‘advanced’ plasmas as well as ‘burning plasma’ scenarios. In addition we have spent time addressing the control of plasma density in an all-metal machine (*profile control; low-density divertor operation* - IPPA 3.1.1).

We have made a number of important advances in understanding divertor physics and more specifically detachment physics (*plasma-wall interactions* - IPPA 1.4.3), and this work will continue.

Recent Accomplishments

The study of cross-field transport across the separatrix and in the SOL has again been a central focus of the C-Mod program this year [1-3]. The breadth of these studies has expanded to include effects on heat transport as well as the relationship of measured particle transport to measured turbulence and the density limit. Such wide-ranging effects of cross-field transport emphasize the importance of continued studies in this area for optimization of tokamak performance with respect to fueling, wall impurity source minimization, divertor design and density limits.

The analysis of the cross-field particle transport in C-Mod leads to the following conclusions: 1) radial particle fluxes (integrated over the entire SOL) are of similar order or dominate the flux into the divertor; 2) these radial particle fluxes lead to high recycling on the vessel walls and high resultant main chamber neutral pressures; 3) cross-field transport in the C-Mod SOL is at or above a critical level, above which particle balance requires the main-chamber plasma to recycle more on the main-chamber wall surfaces than in the divertor; 4) the effective particle diffusion coefficient near the separatrix scales with the inverse of the electron-ion mean free path (λ_{ei}) normalized to the magnetic connection length (L) in the SOL, $D_{eff}\alpha\ (\lambda_{ei}/L)^{-1.7}$.

The scaling and magnitude of cross-field particle fluxes have implications for cross-field heat convection and the density limit. Figure 7 shows estimates of total power into the SOL, power conducted along field lines to the divertor, and cross-field heat convection at the main-chamber limiter and separatrix locations as functions of the line average density normalized to the Greenwald density. The figure clearly shows the increasing importance of cross-field heat convection as the density (or collisionality) of the discharge is raised; at high plasma densities, cross-field convection to the main-chamber walls becomes comparable or exceeds the power conducted along field lines to the divertor. At even higher densities, closer to the density limit, the level of convective heat transport across the separatrix and onto the main chamber wall is larger still, becoming comparable to the total power crossing the separatrix, thus raising the possibility of a connection to the density limit through thermal destabilization of the core plasma.

Measurements of turbulence and fluctuations in the SOL have been made with a number of diagnostics over the past year. It has been found that the character and strength of turbulence varies across the SOL and as a function of density and correlates with the observed radial transport variations. We have identified a 'near' SOL where the effective diffusivity, D_{eff}, and observed fluctuations are low compared to those farther out in the SOL ('far SOL'). This difference between far and near SOL regions has also been observed in the character of 2-D images of visible light emission with 2 μs exposure. These images show intermittent, spatially separated 'blobs' of emission in the far SOL. Blobs ($\approx$ 1 cm in poloidal and radial extent) are often seen extending across the far SOL, into the shadow of the main-chamber limiters. The bursts seen in probe and fast-diode data [4] combined with the blobs seen in the 2-D imaging data are consistent with the idea that large density and perhaps temperature perturbations propagate across the far SOL, convecting particles and energy to the main-chamber limiters.

We have performed experiments investigating the magnitude and the effect of gas leakage from the divertor to the main chamber [5,6]. These experiments have used our novel divertor bypass to control the effective "closedness" of the divertor baffle structure. The results suggest a relatively high level of leakage flux, both with bypass open and closed, of the same order as the level of recycling flux off of limiters and walls in the main chamber. The potentially large leakage flux is due in part to the high divertor gas pressures obtained in C-Mod. Results with the bypass suggest that the magnitude of the leakage

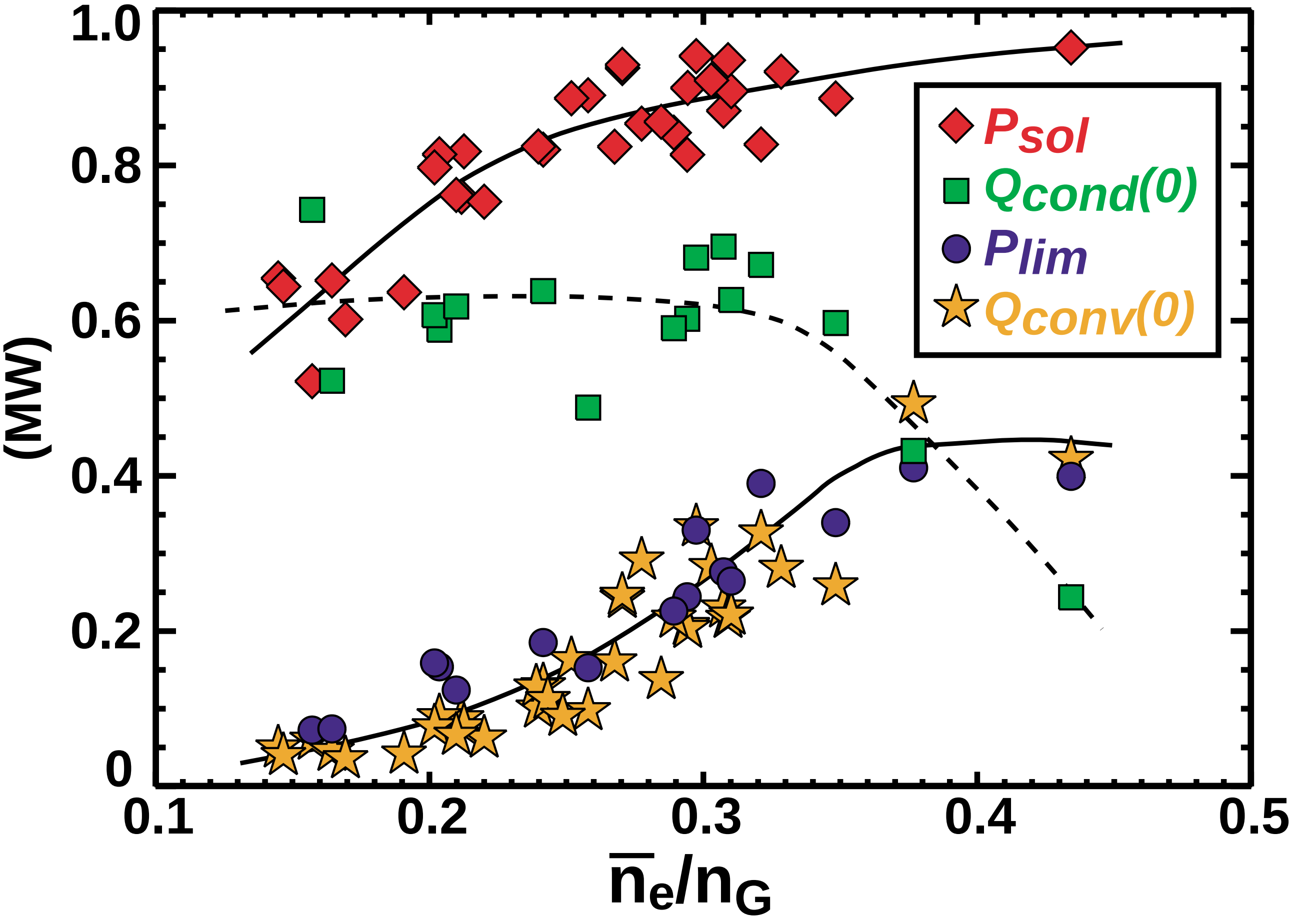


Figure 7. Comparison of power flows into the scrape-off layer: [diamond] power flowing into the SOL, based on input power minus core radiation; [square] power conducted along field lines into the divertor, based on integration of $T_e^{7/2}$; [circle] power convected to the main chamber limiter, based on limiter probes; [star] power convected across the separatrix based on ionization profile measurements. The horizontal axis is lined-averaged density normalized to the Greenwald density.

flux is not influenced by the leakage conductance, but is instead determined, or limited, by the neutral transport out of the divertor plasma into the private flux region.

Experiments have been performed on the effects of baffling on divertor impurity compression for recycling impurity neutrals and ions [7]. The baffling has a strong effect on the divertor impurity neutral compression, where a reduction by a factor of ~ 2 can be produced by opening the divertor bypass. This contrasts with the case of impurity ion compression, where no effect of the bypass is observed. There thus appears to be a separation between recycling impurity ions in the divertor and the corresponding neutrals immediately outside the divertor plasma. We are performing DIVIMP modelling in an attempt to clarify the reason(s) for this apparent separation.

We continue to study the role of recombination and ionization in the divertor. The primary approach for these studies in the past year uses CCD cameras with a toroidal view of the divertor and filtered for D_α and D_γ. We have found that there is significant plasma and recombination in the private flux region prior to detachment [8]. Previous work had shown that the amount of plasma in the private flux region scaled with our estimates of the $\nabla\phi \times B$ poloidal flux across the separatrix from the outer divertor common flux region. We are presently trying to develop a simple model of these flows, and the losses along field lines to the divertor and from recombination.

Further analysis of Mo source rates and Mo levels in the core plasma yielded several results [9]. We found that the divertor and inner wall sources are relatively large but well-screened from the core plasma. During ICRF heating, the antennas became sources of Mo as well. These sources, primarily from the antenna protection tiles, are much smaller than those from the outer divertor or inner limiter, but the screening is much poorer, by a factor of 10-100. All Mo sources appear to be density dependent, dropping with increasing density.

Research Plans

We are planning to continue the work outlined above with regard to perpendicular transport (particle and energy). In the remainder of the FY01 run period we have planned experiments aimed at probing the characteristics of the density limit. The transport in the H-mode pedestal region will be further explored using probes in ohmic EDA H-mode plasmas. Fluctuation diagnostics play an important role in these studies. In particular we are planning to obtain a new 2-D, time-gated, imaging camera to be able to acquire multiple frames at a rate of 10^6 frames/second. The present camera is limited to 1 frame (1 μsec exposure) every 16.7 msec.

The SOL particle transport analysis that has been utilized to obtain effective diffusion coefficients relies on both Ly_α measurements and assumptions of poloidal and toroidal symmetry in the main chamber SOL. The S/N of the Ly_α diode arrays is being upgraded to allow for better transport measurements as well as better modeling of the role of neutrals in edge particle and energy transport. A set of Penning gauges are being upgraded to improve their performance. These will be employed to measure poloidal and toroidal variations in the neutral pressure. The role of neutrals escaping from the outer divertor to the midplane will be further examined through a set of experiments aimed at better quantifying the escape rate.

Following the end of FY01 run period, the emphasis will turn to analysis of the large body of data from this run period that needs to be analyzed. These data will be used to study the dependence of radial transport on Ip, Bt (forward and reversed fields) and density. This database of information will also be used to see if the size of the observed 'near' and 'far' SOL regions of short and long gradient scale lengths correlates with bulk plasma parameters and/or fluctuation characterisitics.

We are planning to continue a new collaboration with the DIII-D experimental group to explore the similarities and differences between C-Mod and DIII-D with regard to radial transport in the SOL, wall recycling, and impurity sources at the vessel walls. The primary collaborators in this work are Dennis Whyte and Peter Stangeby from DIII-D and Bruce Lipschultz from MIT. Bruce Lipschultz traveled to LaJolla to present the C-Mod results.

While there he worked with the staff to generate a DIII-D miniprosal. The miniproposal has been approved for a study of wall-gap variation on radial transport. We have agreed on a similar set of measurements and scaling of wall gap at Alcator C-Mod. We will share data after the experiments are run.

In general we will endeavor to expand the connection between transport observations and the turbulence measurements to "advance the fundamental understanding of the plasma through well-diagnosed experiments, theory and simulation". Probe fluctuation measurements have been made during this run period that will be analyzed, both internally for fluctuation-induced transport and correlation with other measurements, but also in association with external collaborators using, for example R/S statistics. Work has started on a magnetic field fluctuation probe to understand the influence of magnetic turbulence.

We are planning to pursue a deeper understanding of divertor physics along a number of paths. We will implement diagnostics to extend previous work on imaging of recombination light from the divertor to explain the clear evidence of plasma in the private flux zone. A new divertor probe is being designed to probe the entire divertor region (2 dimensions in the poloidal plane). This will allow better analysis of the ionization/recombination processes in the divertor, and better benchmarking of codes, and should elucidate the role of the x-point in divertor and core physics.

We will continue several collaborations in the area of modelling that impact on a number of edge and divertor issues. Xavier Bonnin of the Greifswald Max Planck Institut has installed B2-Eirenne on a Linux PC located at the PSFC. He is going to spend one month at the PSFC in February-March of 2001 to implement the codes and use them to investigate the recombination image analysis and pressure anomalies mentioned above. The U. Toronto group is working on an implementation of the Divimp-Eirenne combination (using the Onion Skin model in Divimp for the plasma background) to try to better explain the divertor and midplane pressures and the role of the divertor bypass. In addition the U. Toronto group is working with MIT, DIII-D, and KFA Institute in Julich (Detlev Reiter) to develop a simplified model for testing and understanding plasma neutral interactions. More specifically, they will be looking at issues of whether neutrals generated in the region of the divertor can circulate around the outer SOL and thus lead to sources of ionization all around the plasma. Another topic that will be studied with the simplified plasma-neutral model is the known interaction of plasma with the upper section of the C-Mod divertor. Plasma recycling above the divertor bypass would lead to increased neutral pressures there, thus possibly changing the effective conductance of gas through the divertor bypass. This study will assist us in understanding the divertor bypass results and in better estimating the real throughput of the bypass valve.

We are now modelling the large database of impurity screening measurements. The clearest defining characteristic of those measurements is a decrease in screening and divertor compression of the impurity species gas with decrease in mass. In addition, the Ar compression and screening are found to be affected by the state of the divertor bypass (open/closed). We will continue to model these data with the DIVIMP impurity transport code in collaboration with the U. of Toronto. Previous work on Mo impurity sources in the divertor will be expanded to include the characterization of B and F sources.

Major upgrades

The major upgrade to the divertor is the reduction of the inner divertor “nose”. This is discussed in detail in the Facilities Section (Sect. 2.7). Its primary effects will be to allow for higher plasma-current operation, a larger operational space for plasma shaping (higher triangularity), and greater control of divertor recycling. The modifications of the inner divertor will allow the upgrade of Langmuir probes located in the inner divertor. The description of a number of new diagnostics is included in the general physics description above.

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2.3 MHD

The MHD research program on Alcator C-Mod is oriented around several of the key objectives in Goals 1 and 3 of the FESAC recommendations laid out in the Integrated Program Plan. In the H-mode edge pedestal region, our studies of the quasi-coherent (QC) mode relate directly to the issue of turbulence and transport (Goal 1.1), while our work on pedestal stability and our studies of high-β modes in the core plasma are encompassed by Goal 1.2 (macroscopic stability), as is the active MHD spectroscopy planned for the future. Disruption research on C-Mod, including mitigation techniques, ties directly to FESAC Goal 3.2 (high β stability and disruption mitigation).

Recent Accomplishments

In the edge pedestal region, characteristic of H-mode, our understanding of the quasi-coherent mode, which is now recognized as key to the EDA, has advanced markedly during the past year. A new $\tilde{B}$ pickup coil was installed in a fast scanning probe in order to make transient measurements very close to the last closed flux surface. A clear magnetic signature of the QC mode was discovered, implying that it must be an MHD phenomenon. Furthermore, from the sharp radial falloff in the $\tilde{B}$ magnitude measured during the probe scans, mode numbers of order ~ 30 can be deduced. More details about the QC mode are contained in the transport section of this work proposal.

The MHD stability of the H-mode edge pedestal to ideal ballooning modes has been revisited, now that high spatial resolution profiles of n_e and T_e are available in the pedestal region. The ideal ballooning code BALOO takes equilibria generated by kinetic EFIT runs using measured plasma pressure profiles obtained from edge Thomson scattering measurements (assuming $T_i = T_e$ in the pedestal). Bootstrap current is included in the EFIT calculation. Because of the high densities in the C-Mod pedestal, the reduction of J_{BS} ($\nu^* \simeq 2 - 15$) is significant and must be included in the calculation. It is found that in both ELM-free and EDA H-mode pedestals, the bootstrap current provides access to the second stability regime, even accounting for the ν^* reduction. Without the inclusion of J_{BS}, the pedestal would be ballooning unstable.

In the plasma core, the first signs of possible β-limiting MHD have been observed in C-Mod discharges with $P_{\mathrm{RF}} \geq 4$ MW and low collisionality ($\nu^* = \nu_i/\epsilon\omega_{*e} < 0.5$), as shown in Figure 8. The observed 2/1 mode has some characteristics of a neoclassical tearing mode (NTM), but it is not unambiguously triggered by a sawtooth crash (which would generate the necessary seed island), so its classification remains unsure at the moment. There are a number of discharges with higher β_N, higher β_p, and $\nu_* \approx 1$, which don't have any noticeable MHD activity.

In the area of disruption research, our first experimental test of the neutral point concept was successfully carried out. TSC modelling done previously in collaboration with JAERI predicted that C-Mod would have a neutral point, and that it should be at $z \simeq +1$ cm. The goal of this year's run was primarily to determine if C-Mod, with its elongation of 1.6, did indeed have a neutral point. Plasmas were run at different heights in the C-Mod vessel, and disruptions were reproducibly induced by injection of impurity pellets. The results are shown in Figure 9. The neutral point was found to be at $z = +2.7$ cm, which is slightly higher than predicted, but more importantly, it verifies that a neutral point does indeed exist for C-Mod's elongated plasmas. So far, plasmas have exhibited up to 8-9 ms of post-thermal-quench stability.

The rangefinder diagnostic was upgraded to have six measurement points on the inboard vessel wall, although the present laser system can only utilize two of these at any one time. Data using the new lower pair of retro-reflectors, which straddle a weld, were taken during the past year's operation, and are currently being compared to earlier measurements from higher on the vessel wall, and to finite element stress/strain calculations, to verify that the planned strengthening of the inner wall will meet the desired goals.

Research Plans

As ICRF operation at power levels above 5 MW is extended to longer pulse durations, β_N should increase from its current maximum of ~ 1.7 to values ≥ 2.0. As discussed in the preceding section, we already have a few examples of a global MHD mode arising at $\beta_N = 1.5$, under certain circumstances, which appears to limit β and even causes it to rollover. Although it is not yet clear if this is a neoclassical tearing mode, other tokamaks have exhibited $m/n = 3/2$ NTM's at similar β_N's and low collisionality. With IPPA/FESAC Goal 1.2 specifically in mind, we plan to map out the parameter space in which such MHD activity occurs on C-Mod, in particular, ν^*, β_N, and β_p, and compare this to observations on other tokamaks. If these modes are found to be NTM's, it may be possible to actively avoid them by using ICRF stabilization of sawteeth to eliminate the seed islands.

As demonstrated on JET, and explicitly stated in Appendix III.2 of the IPPA, active MHD spectroscopy can be a useful tool for understanding macroscopic stability and for developing predictive capabilities. Such a system is currently being built for Alcator C-Mod. It consists of a pair of antennas above and below the midplane on the outboard vessel wall at each of two toroidal locations, separated by $\sim 90°$ (i.e. four antennas in toto). The first two antennas (see photograph, Figure 10) should be completed in time for installation before the remaining FY01 campaign, with the subsequent pair to be installed in FY02, and with full operation planned for FY02-03 and beyond. Power supplies for this experiment will operate in both a low frequency band (1-20 kHz) to study global MHD modes, and a high frequency band ($\sim 100 - 900$ kHz) to study Alfvén eigenmodes. The principal objective is to search for stable modes and measure their damping rates by sweeping the antenna $\tilde{B}$ fields to identify resonant frequencies and to measure the frequency widths of the resonances (which directly give the mode damping rates). In this way, it should be possible to determine in real time whether a particular mode is getting close to marginal stability, and ultimately to take evasive action if necessary, such as reducing heating power or modifying pressure and/or current profiles before a problem occurs. Initially at least, the power available will be rather modest (~ 1 kW), so active feedback stabilization may not be feasible. In principle, it may be possible to excite the QC mode ($O(100$ kHz$)$) and thus control access to the EDA H-mode, but the relatively large size of the antennas will not couple efficiently to the small spatial size characteristic of this mode.

As described earlier, the MHD contribution to turbulence studies on C-Mod is primarily focussed on the quasi-coherent mode in the edge pedestal. Our experimental work in this area is being upgraded by the installation of a new scanning probe head containing a poloidally displaced pair of $\tilde{B}$ coils designed to measure k_θ directly, as well as to do turbulence correlations ($d = 7$ mm separation). We will continue our collaboration with the PSFC theory group on the resistive ballooning work, which is one of the leading candidates for explaining the QC mode. In addition, we expect to enhance our pedestal modelling efforts by utilizing the GA ELITE code, which handles intermediate-n as well as infinite-n stability.

In FY02, the disruption diagnostics in Alcator C-Mod will be greatly enhanced as part of the installation of the new girdle/inner divertor project. In the design of the girdle, accommodation for arrays of Rogowski coils has been incorporated right from the beginning. Figure 11 shows a mockup of a portion of the new girdle/divertor with two Rogowski coils in place. This instrumentation will allow us to measure all of the halo current entering the wall from the inner divertor, with $n = 10$ toroidal resolution. In addition, for the first time, measurements of toroidal eddy currents will also be possible, again with $n = 10$ toroidal resolution. This will allow us to compare disruption forces produced by halo and eddy currents respectively. The toroidal symmetry of disruption eddy currents can also be studied, as can currents generated during startup and other non-disruption transients.

The rangefinder laser interferometry measurements of disruption-induced flexing of the inboard vessel wall will be very useful in verifying the strengthening effected by the new girdle, and the girdle design accommodates the present in-vessel components of this diagnostic. Although there are six measurement points installed on the vessel wall, the present system can only give the relative displacement between two of these at any one time, and since the interferometer currently has no absolute reference arm, we cannot get absolute displacement. In FY02 incremental B budget, we would do a major upgrade of the rangefinder so that all 6 of the measurement points, plus a reference arm, could be used simultaneously to give an absolute displacement profile of the inboard vessel wall/girdle during disruptions. The upgraded rangefinder capability, combined with the new halo/eddy current instrumentation, would be an enormously useful tool for studying and understanding disruption forces and vessel stresses. This upgrade is deferred under the guidance budgets.

Disruption mitigation studies will continue, with further work based on the initial success with the neutral point technique. The extended period of post-thermal quench vertical stability (8-9 ms so far) is long enough for some of the PF power supplies to effect a response. Future work will explore the possibility of extending this period of vertical stability by setting up the plasma control system (PCS) to jump to an alternate programming set at the time of the disruption. Several different PCS programming scenarios have been proposed to try to keep the plasma at the midplane longer, for example, quickly decreasing the elongation. The ultimate goal would be to keep the disrupting plasma positioned on the midplane throughout the entire current quench, thereby greatly reducing $J \times B$ forces in the divertor region. For long timescales, it may even be possible to recover from a thermal quench without terminating the discharge if the current quench can be arrested or reversed.

In the out years, the planned addition of LHCD and advanced tokamak operation opens up a host of issues related to reversed shear, profile control, and very high β_N, such as resistive wall modes, rotation stabilization, double tearing modes, and bootstrap alignment.

Upgrades

The active MHD spectroscopy, which has already entered the fabrication stage, will require some additional amplifiers, detection circuitry, and digitizers in FY02 to be completely operational. Given an incremental budget, the rangefinder upgrade would require replacing the present HeNe laser with a more powerful one, along with many new beam splitters and associated optics. There is also the option of using a pair of solid state frequency-stabilized diode lasers, which could be directly coupled to fibreoptics, using circulators instead of splitters. Finally, if the new PCI-based data acquisition works well for the magnetics, then in an incremental budget, we would like to have a similar system replace the core and edge x-ray array CAMAC digitizers.

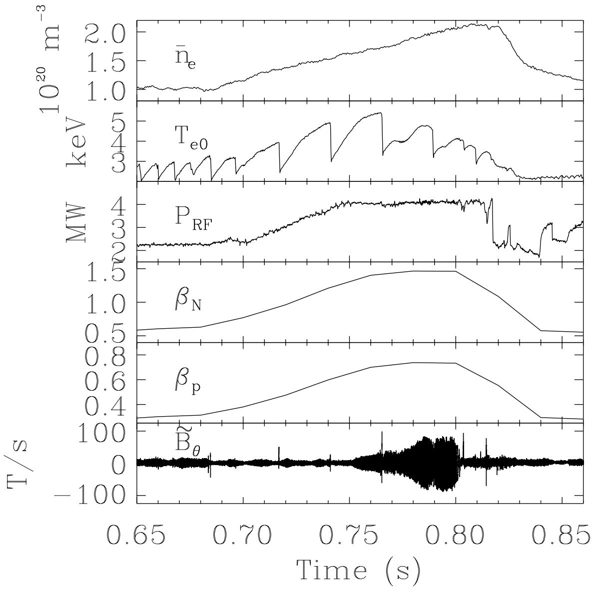

Figure 8 – In this high β, low ν_* discharge, the rollover in β is correlated with the growth of an $m = 2/n = 1$ mode, and not with a drop in P_{RF} nor an H/L back transition. (Shot 1001013016)

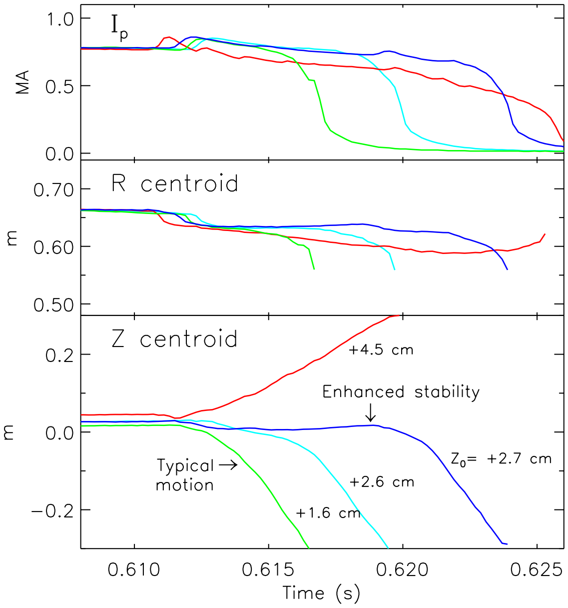

Figure 9 – Results from the first neutral point experiment show that C-Mod plasmas positioned at $z = +2.7$ cm remain vertically stable for 8-9 ms after a disruption thermal quench.

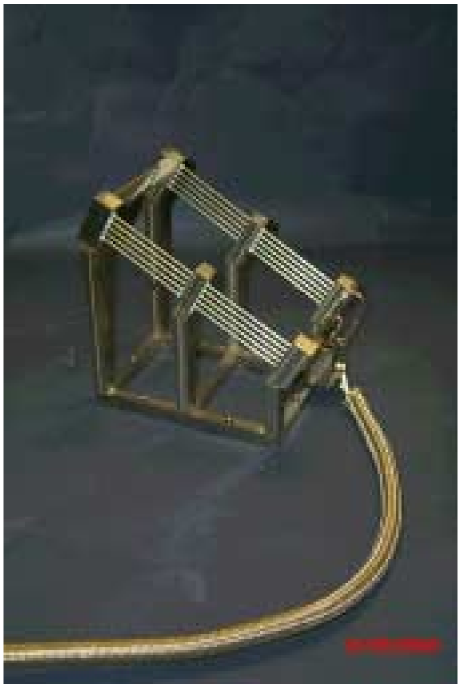

Figure 10 – Partially constructed active MHD antenna (1 of 4) for C-Mod

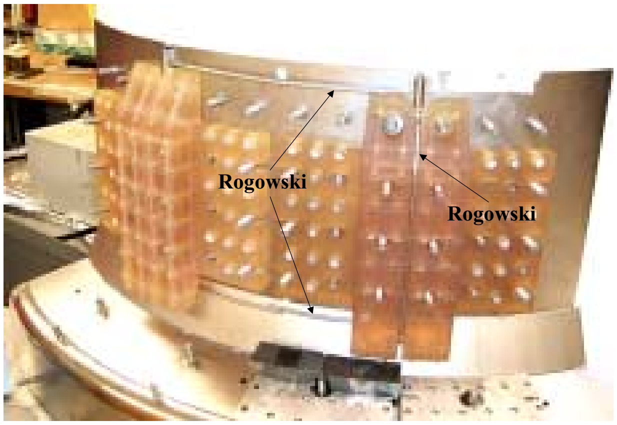

Figure 11 – Mockup of inner wall girdle/divertor upgrade showing halo and eddy current Rogowski coils (partially hidden)

2.4 RF Heating and Current Drive

A major program goal for the Alcator C-Mod RF group is to provide 6 MW of reliable heating power for a variety of plasma conditions. The typical high power density, high density, C-Mod plasmas, and resulting high edge neutral pressures, combined with limited access, demand new approaches to ICRF antenna design. C-Mod also provides a unique opportunity to explore ICRF heating, current drive, flow drive, and mode conversion physics. These investigations are facilitated by: a flexible ICRF system; access to sophisticated ICRF simulation codes; and the availability of advanced diagnostics for RF wave measurements.

With respect to recently developed IPPA goals, the C-Mod RF group is well situated to make significant contributions. Supporting IPPA goal 1.3, wave particle interactions, we are concurrently developing numerical codes to simulate and predict wave propagation, absorption, and mode conversion in 2-D and 3-D, and developing sophisticated RF diagnostics to measure wave propagation, power absorption, and ion distribution function in the presence of RF. We have had recent success in high poloidal mode simulations and measuring the density fluctuations associated with both the fast waves and the ion Berstein waves. From the IPPA Appendix III, an important question to address is: can RF waves maintain and control desirable confinement in long-pulse or reactor-scale plasmas? Profile control using ion cyclotron and mode conversion heating, current drive, and flow drive are under investigation (IPPA 3.3.1). We have had recent success in establishing an internal transport barrier with off-axis heating alone, without an external central particle source or external torque. Experiments are planned to investigate current and flow drive using mode converted ion Bernstein waves (IPPA 3.3.1.1 and 3.3.1.3).

a. Recent Accomplishments

An internal transport barrier (ITB) mode has been observed during intense, off-axis ICRF heating in Alcator C-Mod. This barrier mode is routinely produced in enhanced D_α (EDA) H-mode, 4.5 T, 0.8 MA discharges (H cyclotron resonance position is r/a = 0.44 to the high field side). From TRANSP analysis, the χ_{eff} is significantly reduced inside r/a 0.4 where significant density peaking is observed. Concomitant with the barrier formation, the toroidal plasma rotation near the axis reversed from co-current (+50 km/sec) to counter current direction (-20 km/sec). Increased core impurity radiation, presumably due to improved particle confinement, accompanied the strong density peaking, arrested the improved energy confinement and typically led to a barrier collapse. In an effort to maintain the barrier, experiments using a third antenna, operated at 70 MHz, allowed central ICRF heating of the ITB mode. Initial experiments with up to 1.2 MW confirmed the energy transport barrier and allowed significant increases in the ion and electron temperatures. Mode dynamics were affected by the amount of central heating.

We have successfully operated a modified J-port antenna at 2.5 MW into H-mode discharges and up to 3 MW into L-mode discharges. To obtain this power level, several modifications were required: improved protection tile grounding; additional protective shields for Faraday screen ceramic isolators; and transmission line modifications to optimize the antenna spectrum for heating. The combined (D, E, and J antennas) ICRF power delivered to the plasma reached 5 MW for short pulses. From data and post run analysis, the J-port antenna requires additional modifications to deliver 3 MW of heating power reliably to the plasma. During operation, a strong RF-plasma interaction, observed primarily on corner tiles connecting to side limiters, limited the power in H-mode plasmas to 2.5 MW. Above 2.5 MW this interaction resulted in excessive impurity injections that caused the plasma to disrupt. Unlike the injections observed during the initial J-port operation (FY99), the power limit varied with antenna phasing and plasma current scans, indicating that this interaction followed magnetic field lines and suggesting that substantial voltage is induced due to asymmetric coupling of RF flux on this field line. In addition, significant arc damage was found on the antenna stripline feeds. This damage was localized to sections where the magnetic field connected the stripline to ground. We found no evidence of arcing that crossed the B-field. Furthermore, post-run analysis found cases where the reflection coefficient suddenly increased without a corresponding change in plasma loading that did not exceed the reflection coefficient threshold for arc detection. These arcs would persist for as long as 240 μsec, possibly depositing as much as 100-1000 J into an arc.

The first direct observation of ICRF ($\sim$80 MHz) waves (with $k_\perp$ from 0.5 cm^{-1} to 10 cm^{-1}) was made using an optical heterodyne technique on the Phase Contrast Imaging (PCI) system. The CO_2 laser power is modulated at the RF frequency +/- δf. The electron density modulations at the RF frequency are detected at δf, chosen to be between 200 and 400 kHz. Mode-converted ion Bernstein waves (IBW) in plasmas composed of H, D and ^{3}He at 6 T have been observed at both high (800 kA) and low (400 kA) current. In addition, the fast magnetosonic wave, used to convey the power to the minority resonance or mode conversion layer, has been observed in high density ($\sim 5 \times 10^{20}$ m^{-3}) D(H) plasmas with off-axis ICRH at 4.5 T. Measured wave numbers are in good agreement with the local dispersion relations for both waves. Since the PCI observations are vertical chord averages, the full-wave ICRF code TORIC has been used to simulate the wave fields in these plasmas, thus aiding the interpretation.

For the first time in C-Mod H-mode plasmas, the minority ion distribution function was directly measured while the ICRF was operating, using neutral particle analyzer measurements. Comparison of D, E, and J antenna operation indicated that the parametric decay instability was absent for the J antenna and was strongest for the D antenna. This effect may reflect the decreased local electric field associated with J-port and mapping of the ion orbits.

We have completed the first successful 2-D simulations of mode conversion using a poloidal mode code (TORIC) in collaboration with Marco Brambilla of Asdex Upgrade. The TORIC code employs a poloidal mode representation of the wave electric field in a toroidal geometry. The code solves explicitly for the mode-converted ion Bernstein wave (IBW) and calculates from first principles the electron Landau damping of the IBW and fast waves. In order to carry out large poloidal mode simulations of mode converted IBW electron heating experiments, the TORIC code was implemented on the SV1 CRAY cluster at NERSC. It was found that, with a sufficient number of poloidal modes (N_m), the predicted power deposition profiles agree well with experimental results. The value of N_m is given by a theoretically derived convergence criterion for the poloidal mode expansion; where typically $N_\mathrm{m} \simeq 150 - 250$ is required.

In order to compute the power partition between background electrons and ions that occurs as the minority ion tail slows down via electron and ion drag, the TORIC code has been coupled to the bounce-averaged Fokker Planck module FPPRF (Hammet) in collaboration with Cynthia Phillips of PPPL. This allows detailed analysis of the minority tail formation and fast ion dynamics. During the past year, the coupled TORIC - FPPRF module in TRANSP was thoroughly tested and benchmarked in collaboration with PPPL. The improved TRANSP ICRF package was used extensively to analyze the ITB mode that was obtained with intense, off-axis ICRF heating (discussed above). This same package was also used to analyze discharges in C-Mod where enhanced neutron core barrier modes were observed.

b. Research Plans

The ICRF power and pulse duration suggest further challenges need to be resolved that originate from the high power densities, limited access, and high neutral pressures found in typical C-Mod operation. From analysis of the antenna and data, we have decided to modify the antenna further. To reduce the RF-plasma interaction, the top and bottom protection tiles will be moved to the same radius as the side protection tiles, and a septum will be installed to interrupt long field lines. All protective plasma-facing components, except the Faraday screen, will be covered by BN tiles to minimize the RF-plasma edge interaction. To improve the antenna voltage characteristics, the stripline feeders will be rotated to take advantage of so-called magnetic insulation, and the stripline spacing will be increased to reduce the maximum electric fields. To limit the arc energy, additional arc detection will be based upon the voltage phase measured in the transmission line. The voltage phase was observed to shift promptly by up to 180 degrees during arcs. This new system will be similar to the phase balance monitors currently used on the D and E-antennas.

The immediate run campaign objective is to evaluate the performance of the modified J-port antenna using D(H) minority heating. These data will allow a direct comparison to D- and E-port antenna heating effectiveness and impurity production under similar conditions. The goal is to provide 3 MW of reliable heating power ffrom J-port, which will require voltages up to 30 kV. The additional heating power will be used for central heating to further study the ITB mode observed with intense, off-axis ICRF heating. We will explore the effects of timing and additional central heating on the ITB formation and maintenance. If the antenna proves unsuccessful, a new 4-strap design based upon end-fed, center-grounded straps with coaxial feeds will be considered.

In the longer term, the increased ICRF power should allow greater exploration of the H-mode threshold (P_{th}) at high fields and at high and low densities. It may also allow investigation of H-modes regimes with injected power of 2-3 times P_{th}, and access to EDA H-modes at higher plasma current (lower q). With increased injected power, regimes where pressure driven instabilities become unstable and regimes where type-I ELMs are present may become accessible. Furthermore, the higher power densities will test divertor issues including detachment and power handling. Another issue is to understand and characterize the local RF-plasma edge interactions.

Another important emphasis, perhaps relevant to triggering and controlling transport barrier formation, is RF driven flows. A promising scenario for poloidal flow drive is using mode converted ion Bernstein waves (IBW). With present diagnostics, the poloidal rotation, RF power deposition, and RF density fluctuation profiles can be simultaneously measured. These data will allow us to assess the amount of poloidal flow, its profile, and its relation to RF wave propagation and absorption. Depending on its success, RF flow shear could be used to trigger or maintain internal transport barriers induced by strong off-axis ICRF, pellet injection, or H-L back transitions.

Localized RF current drive via mode converted IBW is another candidate for ITB sustainment. Experiments investigating the current drive efficiency will begin once antenna operation in heating phase has been demonstrated. It is expected that the local RF-plasma edge interaction will be stronger for current drive antenna phasing. Developing a means to minimize these interactions and their impact will be an important issue to address. The driven current will be deduced from analysis of the surface loop voltage. For the off-axis case, the RF current density is a significant fraction of the ohmic current density. For centralized current drive, the predicted driven current exceeds the local ohmic current density. This central current may allow discharges with high-(ℓ_i), which has improved confinement without a power threshold, to be investigated. However, overall current profile control is expected to be limited. The installation of the 4.6 GHz LHRF (lower hybrid range of frequencies) system in FY03, as describe in the AT section of this proposal, will provide a more efficient current profile control capability.

To explore the underlying RF wave physics, a detailed research plan comparing simulation results and experiments will be pursued. The primary areas of emphasis are antenna coupling, mode conversion, and power partition in minority heating experiments. We have begun collaborating with Phil Ryan of ORNL on comparing loading data with the 1-D, slab model GLOSI and RANT3D. We plan to explore new computational platforms on which to implement the TORIC solver (such as MPP devices) in order to carry out large scale poloidal mode simulations on a routine basis. This will be done in collaboration with Cynthia Phillips (PPPL) and Marco Brambilla (Garching). Recently, the TORIC ICRF solver was coupled to an adjoint calculation of the ICRF current drive efficiency. This model will be used to carry out detailed simulations of planned mode conversion current drive (MCCD) experiments on C-Mod. In particular the capability to accurately resolve mode converted IBW in 2-D using TORIC, should provide realistic assessments of current profile control experiments using MCCD.

2.5 Advanced Tokamak Research

Advanced tokamak operating modes with high confinement, β_N and bootstrap current fraction can lead to more attractive fusion reactors. The key tools for accessing these regimes are pressure and current density profile control. Two scenarios for steady-state high performance operation have been identified: optimized shear (OS), where the shear may be reversed or nearly zero over a part of the minor radius, and high ℓ_i . To obtain the necessary current profiles, efficient off-axis current drive is required to sustain OS regimes, while efficient central current drive is needed in the high ℓ_i case. OS scenarios tend to have broader profiles which are more favorable for maximizing the bootstrap fraction, while high ℓ_i regimes tend to be more attractive for producing higher β_N, at least without wall stabilization.

Alcator C-Mod provides an excellent vehicle to explore the physics of advanced tokamak operation in a unique parameter regime namely without central fueling, without external torques, in regimes where ions and electrons are strongly coupled, and with pulse lengths somewhat longer than the L/R time. In addition, C-Mod has unique RF capability with up to 8 MW of ICRF heating presently installed and 3 MW of RF power in the vicinity of the lower hybrid frequency scheduled for deployment in FY 2003. While extension of OS AT modes to essentially steady-state conditions must await the addition of the LH current drive power, these modes, as well as high ℓ_i modes, can be at least transiently investigated using the presently available tools. Accordingly, a significant fraction of the operating time during the FY 2002 and FY 2003 campaigns will be devoted to their study.

a. Recent Accomplishments

1. Experiments

An important element of the advanced tokamak program in Alcator C-Mod is the identification and production of internal transport barrier (ITB) modes which could provide attractive target plasmas for current and pressure profile control studies, and ultimately provide access to regimes of high normalized beta - $\beta_{\rm N}$. Recently an ITB mode was produced in enhanced D_α H-mode discharges in C-Mod via the application of 1 - 2.5 MW of off-axis ICRF heating power at 4.5 T, 0.8 MA, and 80 MHz. The H-minority cyclotron resonance was at $r/a \sim 0.44$ to the high field side in these plasmas. The ITB is characterized by strong peaking of the electron density profile with $n_{\rm e}(0)$ increasing to $0.8 - 1.0 \times 10^{21}$ m^{-3} while the plasma remains in H-mode, with $n_{\rm sep} \simeq 1.5 - 2.0 \times 10^{20}$ m^{-3}. Formation of the density barrier is typically accompanied by a decrease in the toroidal plasma rotation velocity ($v_{\rm tor}$) from co-current to zero. In some cases $v_{\rm tor}$ is found to reverse direction. Transport analysis of these discharges using the TRANSP code indicates the formation of a transport barrier in $\chi_{\rm eff}(r)$ at $r/a \lesssim 0.5$. Impurity radiation, presumably due to improved particle confinement, accompanied the strong density peaking, and would typically lead to a barrier collapse after approximately $10 \times \tau_{\rm E}$. In an effort to maintain and heat within the barrier, experiments utilizing a third antenna, operated at 70 MHz, allowed central ICRF heating of the ITB mode. Initial experiments with up to 1.2 MW confirmed the energy transport barrier and allowed significant amplification of the ion and electron temperatures. In some cases, the central heating also arrests the density and impurity peaking, leading to the potential for steady-state. In these cases, the central co-rotation again speeds up, but the ITB is maintained.

b. Research Plans

1. Experiments

Transport Barrier with Off-axis ICRH

The ITB mode produced with off-axis ICRF heating at 4.5 T may be a promising target plasma for advanced tokamak studies planned for Alcator C-Mod. The toroidal magnetic field at which the mode is produced is compatible with both long pulse operation ($t_{\rm pulse} \sim \tau_{\rm L/R}$) and high $\beta_{\rm N}$. Also, the B-field is consistent with wave accessibility requirements of the lower hybrid (LH) current profile control system planned for Alcator C-Mod. With the availability of addditional central heating power (up to 3 MW) from the J-port antenna we plan to further study this transport barrier mode. We will explore the effect of timing and additional central heating on the ITB formation and maintenance. The role of impurity radiation in the evolution of the barrier will be studied in some detail and techniques for mitigating the impurity build-up will be explored further .

Optimized Shear Discharges

Also in preparation for future LH current profile control experiments, we plan to try to induce OS regimes by a combination of current ramp and heating techniques. In one approach, which has been successfully employed on Tore-Supra and JET, a high-q discharge with $q(0) > 1$ is first established and then followed by a rapid current ramp that lowers the edge q to more nominal values. Heating can also be applied to slow down the diffusion of the current toward the core of the plasma, thus maintaining $q(0) > 1$. In the Tore-Supra experiments, a transport barrier forms, which is related to the formation of a region of shear reversal.

High ℓ_i Regimes

We are planning experiments using IBW mode conversion current drive (MCCD) to explore enhanced confinement regimes with strongly peaked current profiles, i.e., high-ℓ_i regimes. Modeling of MCCD indicates that the driven on-axis current density can exceed that associated with the inductive drive. With co-MCCD, strongly peaked current profiles can be produced; however the degree to which confinement improves will depend on whether, and to what extent, sawteeth are suppressed.

By reversing the direction of the MCCD, i.e., counter current drive, it may also be possible to raise $q(0) > 1$ and form an OS mode. Mode converted IBW are also theoretically predicted to produce flow drive. We intend to explore the conditions appropriate for producing such flows and search for evidence of their existence. If successful, we plan to use this tool to produce ITB's as well, and to extend their lifetimes.

A detailed modelling effort in support of the pressure and current profile control experiments on C-Mod will be continued. Stationary (in time) studies will be done using the ACCOME - KINX model with temperature barriers included. Experimental density profiles characteristic of the 4.5 T density barrier mode (internal and edge density barriers) will be used. Time dependent studies of advanced tokamak modes will be continued using the TRANSP code operated in a predictive mode. In this way, sophisticated numerical packages for ICRF heating (FPPRF) and lower hybrid current drive (LSC) can be utilized. The effectiveness of LH current profile control will be explored in an evolving C-Mod plasma under various assumptions about ITB formation and transport. We will also investigate time dependent feedback schemes on the heating and current drive power for controlling ITB formation and limiting the pressure within the internal barrier.

Lower Hybrid project

The Lower Hybrid Project is scheduled to be completed in mid-FY 2003 and physics experiments will then begin. The programmatic thrust is to achieve efficient steady-state tokamak operation at moderately high $\beta_{\rm N} \sim 3$, with good confinement, $H_{\rm H} > 1$, and high bootstrap current fraction, $f_{\rm BS} \sim 0.75$. Simultaneous achievement of these parameters at moderate $q_{95} \sim 4.5$ will help to establish the reactor potential of advanced tokamak regimes, at medium β and power density. Further advancement of the reactor prospects of AT regimes to higher power density, in line with ARIES-AT studies, requires higher $\beta_{\rm N}$ than can be achieved without stabilization of resistive wall modes (RWM's). Feasibility studies of stabilization approaches will begin with the active MHD spectroscopy diagnostic being installed for operation during the present run campaign (See Section 2.3, MHD). Implementation of active stabilization of RWM's in C-Mod would depend on the outcome of these studies, drawing also on the results of the investigations already taking place on DIII-D.

Initial investigations of lower hybrid current drive will focus on optimizing the coupling of the lower hybrid waves to the plasma, and on evaluating possible interactions between the ICRF antennas and the LH launcher. Full coupling of the LH power (3 MW at the source) to the plasma corresponds to a rather high power density, comparable to that achieved in PBX-M for relatively short pulse (0.5 s vs. up to 5 s planned in the C-Mod experiments). Fortunately substantial current drive can be achieved even with 2 MW delivered to the plasma, and this should be considered a more realistic goal for the first phase of these experiments. The double-barrier mode, recently discovered on Alcator C-Mod with off-axis ICRF heating, presents an attractive target for initial investigations. Ideally, the density should be somewhat lower than in the present experiments in order to achieve higher driven currents. Achieving barrier formation at somewhat lower density, and arresting the monotonic temporal rise of the central density, will be a continued subject of near-term investigations.

Full utilization of the lower hybrid RF capability requires installation of the 4 klystrons remaining from the Alcator C experiment, and fabrication and installation of an additional coupler. Work should begin on this second phase in FY 2003, in order to meet an installation date in FY 2004. Early results from experiments with the first coupler will provide useful feedback for possible modifications to be incorporated into the second coupler design. This upgrade requires incremental funding, both at MIT and at PPPL.

As shown in Table I, there is a strong overlap of the goals of the Alcator C-MOD lower hybrid current drive experiments and the FESAC goals as articulated and expanded on in the IPPA document. Once underway in FY 2003, these experiments are expected to contribute heavily to achievement of the FESAC-IPPA objectives.

Diagnostics and Machine Capability

Experience with AT regimes in other tokamaks has shown that current profile measurements are important for understanding the physics of barrier formation and confinement improvement, in particular the effects of magnetic shear and q_{min}. As discussed in the Facilities Section (2.7), a high priority is being placed on the development of the diagnostic neutral beam and MSE measurements in C-Mod. An additional diagnostic under development is an imaging X-Ray spectrometer. Such an instrument has proven invaluable to understanding the location of the current driven by LH waves and correlating the results with theoretical predictions. C-MOD plasmas can be viewed either tangentially, as in PBX-M (although without spectral capability) or transversely, as in the tomographic approach developed by Y. Peysson on Tore Supra. We are currently engaged in discussions with Dr. Peysson concerning the possibility of fabricating an instrument similar to that used on Tore Supra through a US-CEA collaboration. Design will begin this fiscal year and fabrication could begin in FY 2002. The fabrication should be completed in FY 2003, in time for use in the initial LH experiments. The funds for this important diagnostic are not available in the guidance A budgets, but are provided by the incremental budgets.

A third diagnostic that is expected to be particularly useful for AT experiments is based on the active MHD spectroscopy technique developed at JET. The principle is to use a coil mounted near the plasma to excite MHD modes. When these modes approach the stability boundary, their damping becomes small and is simply related to the bandwidth of the mode resonance. In principle, knowledge of the proximity to instability can be used to "steer" the plasma away from the instability boundary. This diagnostic is presently being developed and will be installed during the extended machine opening beginning in FY 2001. The design of the exciting coil is sufficiently robust that it could serve as a protoype for coils that could be used in "active wall" stabilization experiments.

Finally, AT experiments will also place heavy demands on machine performance, in particular in areas of pulse length, heat management and density control. Various initiatives to address these issues are underway and described in other Sections of this proposal.

Table I: Areas of contribution of the lower hybrid experiments to the IPPA goals

IPPA Top Level Goals	General areas of Contribution	In-depth areas of Contributions
3.1 MFE Goal 1: Advance understanding of plasma, the fourth state of matter, through well-diagnosed experiments, theory and simulation	**3.1.1 Turbulence and Transport**	**3.1.1.2 Understanding transport barriers** - explore roles of magnetic and velocity shear
	3.1.2 Macroscopic Stability	**3.1.2.1 Understanding Observed Macroscopic Stability Limits** - use current profile control to understand limits to operation near the β_N limit
	3.1.3 Wave - Particle Interactions	**3.1.3.1 Plasma Heating and Current Drive** - Check theory of current drive by lower hybrid waves, e.g. $f(\mathbf{v})$. Investigate off-axis current drive and sustainment for long (L/R) time scales
3.3 MFE Goal 3: Advance understanding and innovation in high-performance plasmas, optimizing for projected power-plant requirements; and participate in a burning plasma experiment	**3.3.1 Profile Control**	**3.3.1.1 Plasma Current Profile** -Develop consistent models of self (bootstrap) and LH driven current. Seek to demonstrate states of higher plasma density.
		3.3.1.2 Plasma Pressure Control -Explore use of variable frequency ICRF heating in conjunction with current profile to improve plasma confinemnet and stability.
	3.3.2 High β Stability and Disruption Mitigation	**3.3.2.3 Active Profile Control to Avoid Unstable Boundaries**- Use combination of external and internal current drive, and variable heating profiles to explore and extend stability boundaries.

2.6 Physics Basis for Next Step Burning Plasma Options

Three of the main goals of the fusion energy sciences program, both within the U.S. and abroad, are: to advance the understanding of plasmas and enhance predictive capabilities (IPPA 3.1 MFE Goal 1); to advance the understanding and innovation of high performance plasmas to an alpha heating dominated burning plasma regime (IPPA 3.3 MFE Goal 3); and to develop enabling technologies to advance fusion science (IPPA 3.4 MFE Goal 4). The high field, compact tokamak is presently one of the leading Next Step Options (NSO) for a burning plasma experiment and Alcator C-Mod is an ideal smaller scale experiment where many of the burning plasma physics and technology issues can be tested.

In the near-term, increased ICRF power to greater than 5 MW will allow C-Mod to reach high β and low ρ_* regimes with higher ν_* than other tokamaks have achieved. Scalings based on these dimensionless parameters improve understanding of turbulence and transport and aid in designing a future NSO (IPPA 3.1.1, 3.3.3). The increased power will allow C-Mod to study higher β macroscopic stability limits (IPPA 3.1.2, 3.3.2). Since C-Mod operates exclusively with RF auxiliary heating, the particle and heating sources are decoupled, with no external momentum drive, and with $T_i = T_e$. All of these are prototypical of the burning plasma regime, which makes C-Mod results essential for a better understanding of the plasma physics expected in any of the NSO's. Understanding the wave-particle interactions due to RF heating and current drive together with profile effects and the effects on the high energy ion tails produced are further IPPA goals addressed on C-Mod (IPPA 3.1.3, 3.3.1, 3.4.1). Upcoming divertor modifications and the toroidal field coil inspection will permit C-Mod to reach plasma currents up to 2 MA at the full toroidal field of 8 T with increased triangularity to more easily access the theoretically predicted enhanced confinement regimes with second stability. These advanced shaping capabilities and changes in divertor geometry help to integrate core and edge physics models and better understand boundary physics and technology issues (IPPA 3.1.4, 3.4.1). The first 4 - 5 s long inductive pulse experiments on C-Mod will be carried out in FY2001, which are the first steps toward long pulse scenarios in support of IPPA goals 3.1.1 and 3.1.4.

In the longer term, when 3 - 4 MW of lower hybrid heating will be available in addition to the ICRF power, long pulse advanced tokamak scenarios will be achievablem and issues of profile control (IPPA 3.3.1, 3.4.1) at high density and power and particle handling of the divertor (IPPA 3.1.4) will be addressed. These high performance experiments provide a test bed to assess fusion technologies as well as to supply data to benchmark transport codes and challenge present plasma science theories (IPPA 3.1.1, 3.1.5).

(a) Optimized Performance

The Enhanced D_α (EDA) H-mode pioneered on Alcator C-Mod has many of the characteristics required for a burning plasma experiment: high energy confinement, high density, and low impurity accumulation, all in a steady-state regime without the large intermittent heat loads found in Type I ELMy regimes. After the divertor modifications and toroidal field inspection, C-Mod will be able to operate at plasma currents up to 2 MA at 8 T and a key question is whether or not EDA H-mode performance can be extended to even higher field, current, and input power. Modest extrapolations of C-Mod H-mode performance that has already been achieved at lower current, field, and power indicate that central plasma pressures up to 10 atmospheres may be achievable at full parameters in C-Mod. Such record plasma pressures in a tokamak approach those required for next step burning plasma conditions. Exploring this high performance, high pressure regime will be important in order to test whether or not instabilities driven by the necessarily steep pressure gradients will degrade confinement.

A next step burning plasma experiment must optimize the current rise to make full use of the volt-s capability of the machine and to gain current density profile control. Experiments are planned on C-Mod to optimize the current rise with strong fuelling and ICRF heating to attempt to simulate proposed ignition scenarios in high field compact burning plasma experiments. In the longer term, lower hybrid heating (LHH) and current drive (LHCD) will also be available to further optimize the current rise to achieve peaked high density and temperature conditions in quasi-steady-state.

Designs of next step burning plasma experiments depend on scalings from present experiments to predict the parameters required for high performance. C-Mod plays an important role in confinement, pedestal height and width, and H-mode threshold scalings, because it operates in a density and toroidal field range that is inaccessible to other tokamaks. These data help reduce the uncertainties in extrapolating to larger next step burning plasma devices.

(b) Plasma Stability at High Performance

In a burning plasma experiment, large modulations of the central ion temperature due to particularly large sawteeth, make it difficult to maintain the plasma burn, and provide large seed islands that destabilize Neoclassical Tearing Modes (NTM's) which can substantially degrade confinement or even lead to disruptions. By operating at lower β_p and at moderate edge safety factor, a compact high field burning plasma experiment can reduce the impact of sawteeth on performance. With current drive phasing of the ICRF and in the longer term with LHCD, C-Mod will develop sawtooth stabilization scenarios that can be applied to future burning plasma experiments.

At low collisionality and high β, tokamaks are often limited in performance by NTM's. At higher collisionality, such as that normally found in EDA H-modes in C-Mod, the neoclassical effects are reduced by collisions so that NTM's may not be a problem. However, when C-Mod operates at high power and lower collisionality, large m=2, n=1 modes have been observed that grow in amplitude with increasing β. While these modes have not yet been clearly identified as NTM's, it is likely that with increasing power, C-Mod will be able to study regimes with such β limiting MHD activity to attempt to determine the operational range of collisionality and β where such modes limit performance. Then, by comparison with NTM data from other tokamaks, including DIII-D, ASDEX-Upgrade, and JET, scalings can be refined to predict how a next step burning plasma experiment can be operated to avoid these instabilities.

Under some conditions when the plasma becomes unstable to NTM's, ideal kink or resistive modes, or vertical instabilities, it can rapidly disrupt, causing a loss of all of the plasma energy to the walls. Disruptions are particularly dangerous for high field tokamaks because of the extreme $\mathbf{j} \times \mathbf{B}$ forces on the vessel and surrounding components as the plasma current is lost. The effects of disruption forces can be reduced if the plasma can be made to disrupt against the inner wall rather than up or down into the divertor as is usually the case. The neutral point investigations described in the MHD Section (2.3) are aimed squarely at this problem.

Another class of modes that are predicted to limit performance in a burning plasma experiment are the fast particle driven modes such as Toroidal Alfvén Eigenmodes (TAE's). These modes are driven unstable by a steep gradient in the fast ion β, which may be particularly severe in an α-heating dominated burning plasma. The effects of a fast α tail can be simulated in C-Mod through hydrogen minority ICRF heating at low H concentration and relatively low density to drive a fast H ion tail. Under such conditions, TAE modes have been driven unstable during the current rise phase of C-Mod discharges, in the frequency range from 150 to 450 kHz. Other high frequency modes in the frequency range $600 < f < 900$ kHz, which may also be AE's, are observed in some ICRF heated EDA H-mode plasmas when the H concentration is sufficiently low, even though a high energy fast ion tail is not expected at such high densities. The proximity to marginal stability of these kinds of high frequency modes will be studied with a set of four Active MHD Spectroscopy Antennas, which will be used to drive stable modes in the plasma in much the same way as has been done on JET. The first two of these antennas will be installed in FY2001 and the remainder will be installed in FY2002. The basic plasma science of the stability of AE's in high density, high field plasmas should then be applicable to a wide range of fast particle driven modes in plasmas.

(c) RF Physics and Technology for Next Steps

At the high densities required of burning plasma experiments, neutral beam heating requires very high energy beams to penetrate to the plasma core and large ports are required to pass the beam into the chamber. ICRF heating, on the other hand, can be tuned to a resonance in the core of the plasma and the high density actually helps to concentrate the heating in the core. High edge densities also provide good coupling of the ICRF waves into the plasma. For these reasons, ICRF is the preferred method of auxiliary heating for a burning plasma experiment. Alcator C-Mod has demonstrated the feasibility of high power ICRF heating in high field, high density plasmas. Current experiments are aimed at optimizing the design and operation of the four strap antenna at J-port. Future experiments will aim at high power He^3 minority heating at 8 T to achieve high plasma performance at high field and current.

Recent experiments with off-axis ICRF heating at lower toroidal field indicate that it is possible to control the spontaneous density profile peaking found in H-mode under these conditions through proper tuning of the ICRF. In particular, further two-frequency ICRF experiments are planned to attempt to maintain high confinement and control the impurities and density peaking with off-axis ICRF at 80 MHz and simultaneously heat the plasma core with on-axis ICRF at 70 MHz. If these techniques are successful, they may extrapolate to improve the fusion yield of a next step burning plasma experiment through controlled density peaking.

In the longer term, with the addition of 3 MW of LHCD, experiments will be performed to control the current density profile with off-axis current drive. By maintaining $q(0) > 1$, sawteeth can be eliminated and if $q(0)$ can be maintained above 2, NTM's with m=3, n=2 and m=2, n=1 can be avoided even up to high β. By combining core ICRF heating with off-axis LHCD, it may be possible with full power to achieve $\beta_N \geq 3$ with high bootstrap fraction ($\sim$ 70%) and high confinement (H$_H$ $\sim$ 1-2) in quasi-steady-state conditions. Demonstrating these conditions in C-Mod will provide an important benchmark for future next step burning plasma experiments to develop a more efficient 'Advanced Tokamak'.

(d) Edge Physics and Technology for the Next Step

One of the key challenges for next step devices is to operate at extremely high wall power loadings, while maintaining good core confinement and a good edge energy and particle transport barrier with the prevention of impurity accumulation. Power loading of the divertor becomes unacceptably high unless most of the power through the edge of the plasma is dissipated before reaching the divertor plates. This is accomplished in C-Mod through strong gas puffing in the divertor with deuterium and impurity gases such as nitrogen. Under such conditions, divertor detachment occurs when the divertor density exceeds about 10^{21} m^{-3} and the temperature drops below 2 eV, at which point the power loading is reduced to acceptable levels. Detachment is more difficult to obtain, however, at higher input power. Experiments have been carried out with parallel power flows in the scrape off exceeding $0.5\,GW/m^2$. With additional ICRF power, the parallel power densities should reach $\sim 1\,GW/m^2$, similar to that expected in NSO designs.

While graphite is a good first wall material for high power density, the co-deposition of tritium with carbon results in unacceptably high tritium inventories in a D-T reactor with a carbon based first wall. So, a next step burning plasma experiment will likely have a high Z metal first wall much like that of Alcator C-Mod, which uses molybdenum. The first 4 - 5 s long pulse experiments will be performed in FY2001 on C-Mod, which with the higher power expected, will provide valuable tests of first wall power loading and impurity control. The longer pulses will also give more information on the degree to which boronization or other wall conditioning techniques can apply to a next step burning plasma regime.

(e) Advanced Tokamak Integration for the Next Step

The 'Advanced Tokamak' regime with high bootstrap fraction ($\sim$ 70%), $\beta_N \geq 3$, and high confinement (H$_H$ $\sim$ 1-2) in steady-state could eventually lead to a more efficient fusion reactor. Such a regime requires integrating most of the above physics and technology into a next step burning plasma experiment. In C-Mod, this requires putting together high ICRF power and high power LHCD to reach high β_N for a long pulse in quasi-steady-state. The synergy between the ICRF and LHCD is also essential for current and density profile control to stabilize sawteeth and maintain q(0) $>$ 2 to avoid low order NTM's and achieve low or reversed shear with high confinement. These high power long pulses at relatively low densities will challenge the divertor power handling and impurity control. Integrating all of these diverse physics issues into such an 'Advanced Tokamak' regime will be a valuable achievement on the path toward a more efficient burning plasma experiment.

2.7 Facilities

Recent Accomplishments

During the last year, 1580 plasma discharges have been produced in support of C-Mod science goals. They have been produced with an 80% startup reliability and an engineering systems availability of greater than 97%. The engineering and physics groups have worked to make improvements to the ICRF transmitters and antennas, and DNB system and diagnostics. We have also completed the design of the new inner divertor and are nearing completion of its fabrication. In February of 2000 we were given approval for the Lower Hybrid MIE Project and have made a great deal of progress on this major new initiative over the last year.

In support of our heating, and flow and current drive goals, a major activity over the last year has involved improvements to the J-Port ICRF 4-strap antenna. Over the last year, this antenna has gone from operation at power levels of 1.3 MW with virtually no plasma heating, to greater than 2 MW of power with heating efficiencies as good as those of the D and E-Port antennas. Power levels above approximately 2.5 MW were limited by impurity influx from the front of the antenna. In addition, arcing along the stripline components also limited reliability. These problems are currently being addressed and are discussed in detail in the RF section (2.4) of this proposal.

In FY2003, experiments will begin aimed at developing efficient steady-state tokamak operation by launching lower hybrid waves into Alcator C-Mod plasmas. Up to 3 MW of RF source power will be available at a frequency of 4.6 GHz. Twelve of the 16 klystrons originally used on Alcator C will be applied in the first stage of the project. A klystron test stand has been built to test all of our klystrons to full power with power supply limited pulse widths of 10 ms. We have verified that 13 klystrons operate at full power the test stand, and the remaining three klystrons are repairable. We have also made progress reworking the klystron carts, each of which will house four klystrons, and the required bus work, filament transformers, chokes, and other electronics required to support these tubes. These carts will be installed in the C-Mod experimental cell in early FY2002. The high voltage power supply and modulator needed for long pulse operation of the klystrons has been ordered. This 50 kV, 206 amp supply, using rapid IGBT switching techniques, will be able to both regulate with precision the high voltage, and also completely shut down in 5 μs, providing fault protection for the klystrons. Selection of the circulator, another critical component, is also complete. One circulator is needed for each klystron.

Work has also begun on the control systems needed for this project. PLC software and hardware are under development; they will be used for slow control functions such as bringing up and monitoring supplies and water systems, latching and clearing faults, and providing personnel safety interlocks and monitors.

The fast response needed to control both the power and phase of the klystrons will be provided by LabView RT PC based systems. A development program for this system is underway, with the required hardware and software in house and in operation. Testing of the I/Q detector required to measure the phase and amplitude of the output power from each klystron is also underway. A control loop consisting of the I/Q detector, LabView PC, and a vector modulator will make up this system. Progress on klystron fault detectors and fault detector boards has reached the point of printed circuit board fabrication. A 24 bit fiber optic link has also been developed to communicate between control components.

PPPL has lead responsibility for design and construction of the lower hybrid launcher. Detailed drawings now exist for the launcher windows, tiles, coupler, and high power microwave components needed to feed the output from the 12 klystrons into the 96 waveguides at the coupler. Four modules, each with 24 waveguides, make up the launcher front end. Alumina windows brazed into the waveguide provide the vacuum interface. The launcher position can be moved between shots radially, from 0.5 cm outside the separatrix to well behind the limiter. Drawings of the coupler and launcher are shown in figures 12 and 13. A final design review is scheduled for April 2001, with delivery of the system expected in November 2002.

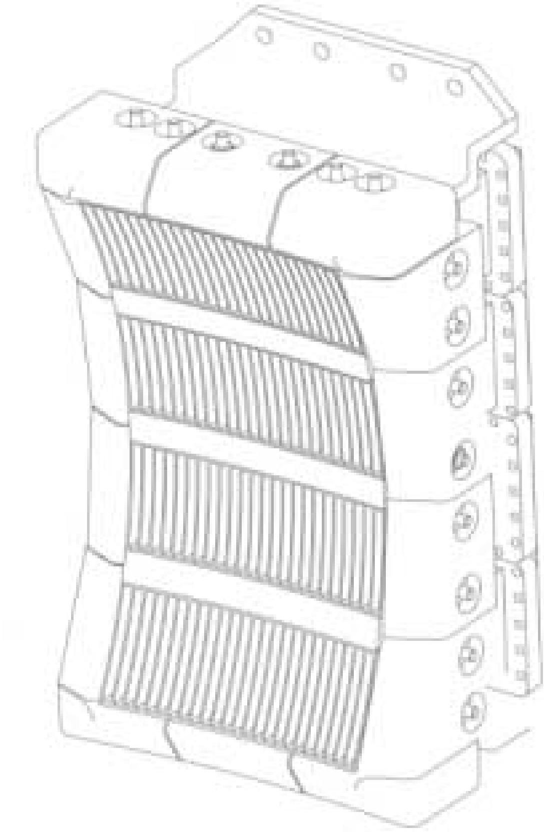

Fig 12. An array of four modules each with 24 waveguides makes up the lower hybrid coupler. Each waveguide is vacuum sealed with an $A1_2O_3$ ceramic window recessed 10 cm into the waveguide to protect it from plasma. Boron nitride protection tiles are also shown.

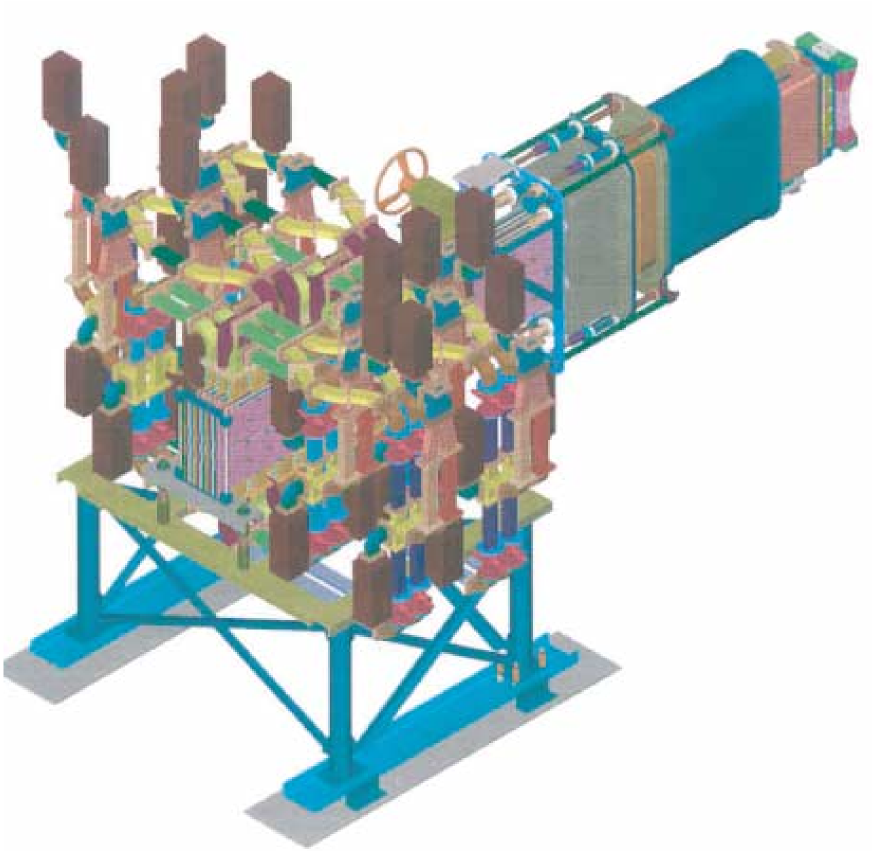

Fig 13. The launcher and hardware required to couple the power from 12 klystrons to the plasma. Bellows allows the launcher to be moved radially several centimeters.

In support of the Lower Hybrid Project, as part of the MIT infrastructure improvement program, MIT has formally agreed to provide approximately 1.2 million dollars in site improvements over the next year. MIT will provide the pumps, water lines, and valving needed to supply 1200 gallons/min of cooling water to the klystrons and associated equipment at up to 200 psi. MIT has made improvements to the lab space needed for lower hybrid system development and support. They will provide the new mezzanine in the experimental cell needed to support the klystrons, and will upgrade the air conditioning to support the added heat load. They will also provide the slab in our Hi-Yard needed to support the new high voltage supply and modulator and provide the 13.8 kV feed to the supply and high voltage cable to the cell.

In support of high performance plasmas, significant upgrades to the inner divertor structures in Alcator C-Mod have been designed and will be implemented in FY2001-

FY2002. This upgrade has gone from conceptual design to the procurement stage over the last year. Besides allowing high field, high current operation at the 8 T, 2 MA level, the new design provides a larger range of plasma shapes, including significantly increased triangularity. It will consist of twenty interlocking inconel 625 plates forming a girdle surrounding the inner wall. The number of studs on the inner wall will be tripled to provide the strength needed to hold the girdle in place during worst case disruption events. A drawing of the new divertor is shown in figure 14. Extensive structural and thermal modelling of the new components indicates a very strong structure that can easily handle the forces and heat loads expected during disruptions and long pulse AT operation. Tiles, fastener and keeper hardware, tile support plates, and other components are either under procurement or in house. The backing and C plates that form the heart of the divertor are being fabricated. All divertor components are expected to be in house by early April 2001.

Operation of the Diagnostic Neutral Beam is of vital importance to both our short and long term physics goals. Crucial measurements of plasma current, ion temperature, and rotation profiles, as well as fluctuation levels will be provided by the DNB and its associated diagnostics. The University of Texas with help from MIT scientists and engineers brought the DNB, together with several very important beam related diagnostics, into operation this year. The high voltage supply, mod/reg, filament supply, vacuum system, control system, CAMAC data acquisition equipment, and other support systems were commissioned and debugged by April 2000. The beam is now operating near full current and voltage. Improvements to the vacuum and gas feed systems have allowed the full energy component fraction to be improved and water levels to be greatly reduced. The route to further improvement of the beam quality primarily involves automatic operation of the beam to properly condition it. Signals have been obtained from the BES, MSE (PPPL/MIT), and CXRS diagnostics. However, because of the initially poor performance of the beam, together with boron coatings on the optics, and misalignment caused by disruptions, the signal levels have been too low for useful physics results to be obtained. We are addressing some of these problems during our current short up-to-air period. We are providing all support possible to this very important project to bring it into productive operation over the next year.

Near the end of the FY2001 campaign , discharge lengths long enough to be of interest in planning for our advanced tokamak experiments will be performed. Mechanical and thermal stresses on the coils, power supplies, bus, and alternator have been evaluated. An extensive set of new instrumentation has been added to our power systems to monitor component and bus temperatures during long-pulse operation. This first period of long-pulse operation will be used to determine what changes and upgrades are required to C-Mod systems before the Lower Hybrid experiments begin. In addition, experiments to study wall conditioning, plasma fueling, divertor heat loads, and density control can benefit from this early long-pulse operation.

Research Plans

The engineering group will work to provide the new RF, power, and diagnostic systems needed to pursue the major physics goals of Alcator C-Mod.

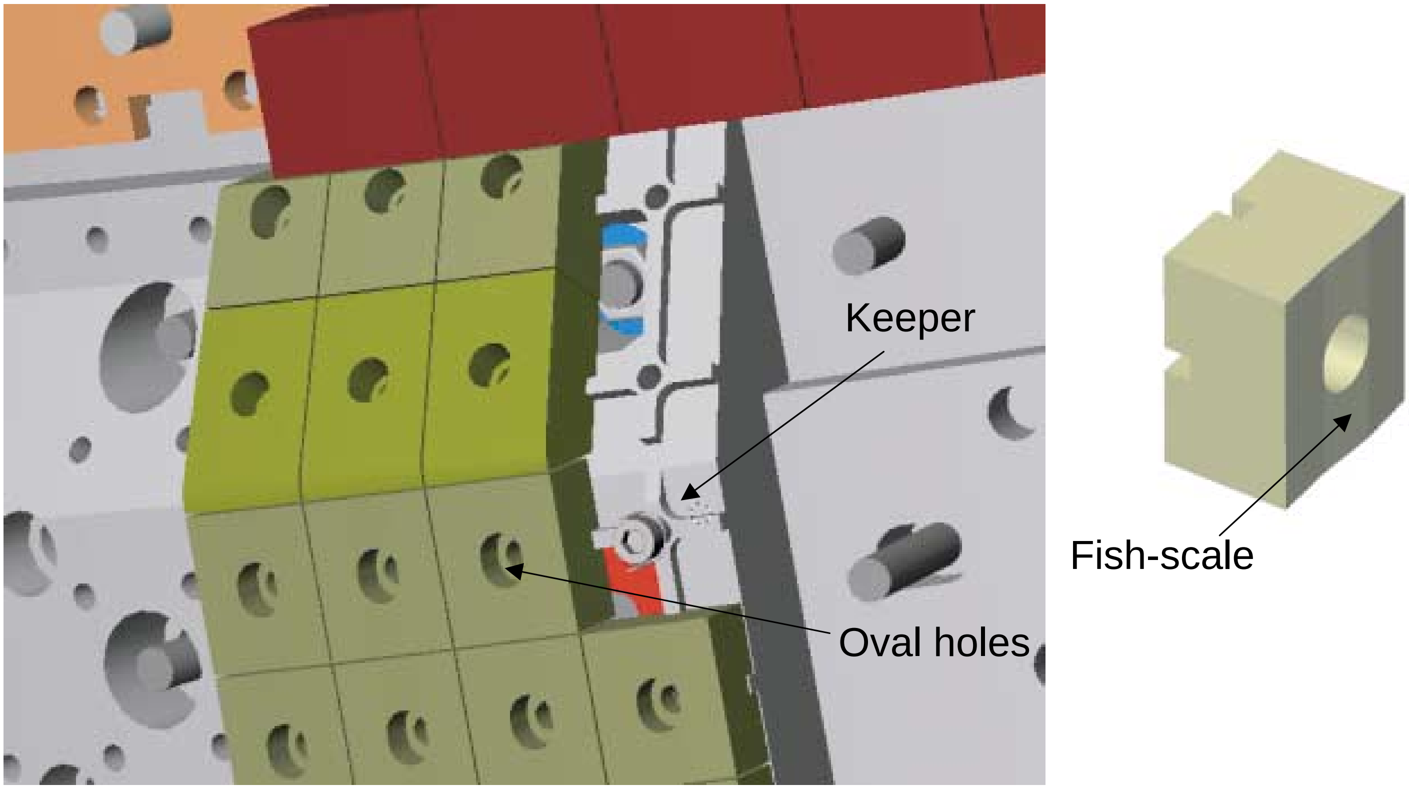

Fig 14. Inner divertor girdle and tile components are shown. Each tile is fixed in place with one inconel 718 bolt. Tiles are keyed into position to prevent rotation. Keeper retrains bolt rotation without a washer stack (oval hole). Fish-scale protects hole in tile.

Upgrades

All inner divertor upgrade components will be in house and under test by April 2001. All QA and prep work will be completed well in advance of our up-to-air period, scheduled to begin in the summer of 2001. Installation will require about 8 months of invessel work.

A machine inspection will take place from late FY2001 through the middle of FY2002. During this time the machine will be disassembled to the point of removing the retaining cylinder, so that many machine components including those of the TF magnet, OH coax, and curved bus will be available for inspection. A successful inspection of the machine will allow high field, high current operation to begin in the FY2002 campaign. An upgrade to the C-Mod machine is also planned during this time. If a coaxial feed system is found to be needed for the J-Port ICRF antenna, new 6” ID holes through the cylinder and vessel would be machined to provide access. In addition, modifications to the cylinder around the horizontal ports will allow more tangential diagnostic access. This change will partly make up for the loss of diagnostic access incurred with installation of the lower hybrid launcher. Structural analysis of the vessel and cylinder indicate these changes are acceptable. In addition, we will upgrade the liquid nitrogen cryostat so that horizontal flanges, which are now captured by the cryostat, can be removed with the cryostat in place. In addition, new instrumentation ports will also be installed on the cryostat to allow for more efficient cable passage. These changes will also provide improvements in our ability to properly seal and insulate the cryostat.

The MIT sponsored site modifications in support of the Lower Hybrid project will take place over the next 12 months. Improvements in lab space, installation of the new water system, construction of the klystron support mezzanine, improvements to the cell air conditioning system, and installation of new 13.8 kV service and high voltage cabling, will all take place during our next up-to-air period. We will continue to develop the slow and fast control systems and data acquisition equipment and software needed for the lower hybrid project. Installation of the high voltage power supply/modulator is expected by October 2001. Assuming the current guidance A budgets, all work at MIT will be completed on the original schedule, by October 2002. Delivery of the PPPL launcher is scheduled for November 2002, and will be followed by the high power coupler interface components by March of 2003.

We consider it essential to our program goals to be able to reliably supply 5 to 6 MW of ICRF power to the plasma. We are therefore planning for the possible fabrication of a new ICRF J-Port antenna in FY2002. This upgrade is dependent on whether or not >3 MW of RF power can be reliably supplied by the J-Port antenna with the improvements currently being implemented.

A fast ferrite tuning system is being considered for FY2002 and FY2003. This system will eliminate the shot to shot tuning process that now makes inefficient use of run time and the scientific staff. This system would be able to automatically maintain a good match between antennas and transmitters with shot to shot variations in plasma conditions and changes in confinement mode. In FY2002 we will install a prototype system, which at one time was planned for operation on the ASDEX-U experiment. We will develop and test the fast control system electronics using this prototype. In FY2003 we could, with substantial incremental funding, install four of these devices, so that all transmitters would be operating with fast tuning systems.

Several smaller upgrade projects are planned over the next 12 to 18 months for the ICRF systems. The control system is currently a network of interconnected modules using in some cases 20 year old technology. Each transmitter will soon have a modern control system consisting of one rack of identical, easy to debug equipment. This change should greatly improve the ability to maintain the reliability of this critical system. The low power RF drive components for each system will be moved from the instrumentation racks, where they are currently installed, to shielded enclosures near the inputs to the transmitters. RF pickup should be reduced in this new configuration. Improvements will be made to fault detection, and new fast data acquisition channels will be added. New acoustic arc location detectors are being developed, and electronic arc localization techniques are also being planned in collaboration with Rick Goulding at ORNL. We are also considering modifications to the # 3 and # 4 transmitter final output cavities, so that a lower plate output impedance can be obtained. This change allows the tubes to operate at a lower plate voltage, for a given output power, and should therefore extend the tube lifetime. Higher output power levels could also be reached, if needed.

As a significant upgrade to our in house analysis capability, CST electromagnetic design software will be used to model ICRF transmission lines, phaseshifters, and antennas, and lower hybrid coupler water loads, couplers, and waveguide components. This software provides fast detailed analysis of the RF hardware, but does not model antenna or coupler interactions with the plasma. However, it will provide quick feedback with which to make engineering decisions regarding changes to these complex systems. More detailed modelling of the ICRF antennas, including plasma effects, will be done in collaboration with ORNL.

Upgrades to the Diagnostic Neutral Beam will include changes to the beam duct to reduce the neutral pressure along the beam path. Significant attenuation of the beam may be taking place in the duct during some high density plasma operation because of this neutral gas. Pressure gauges together with H_α detectors will be installed on the duct to determine the importance of the neutrals to beam operation. Automatic conditioning of the beam will be brought into operation to improve the full energy fraction of the beam. A new magnetic bucket design may also be developed to improve the full energy component. Wherever feasible, shutters will be added to the beam diagnostics to protect them from glow discharge cleaning and boronization. Several improvements will be made to the MSE system to strengthen it mechanically and to improve its optical performance.

3. Alcator C-Mod Collaborations

- active as of FY 2001 -

Collaborative efforts, with both domestic and international participation, form a vital and integral part of the Alcator C-Mod program. Contributions of collaborators cover the entire spectrum of activities, with particular emphasis on theory, modeling, plasma heating and current drive, and diagnostic development. Ongoing collaborations are summarized in the following bulleted lists. Continuation of these collaborations is contingent upon each collaborator's separate funding situation.

3.1 Major Collaborations

PPPL

– ICRF physics and modelling, antenna upgrades, operations
– Comparison with physics based transport models
– Motional Stark Effect diagnostic implementation, measurements and upgrades
– ECE diagnostic upgrades and measurements
– Reflectometry upgrades
– Fluctuation studies
– Lower Hybrid Current Drive Project
– Advanced Tokamak Physics
– Divertor modeling

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– Diagnostic Neutral Beam (DNB) operation and maintenance
– DNB diagnostics (CXRS, BES)
– ECE diagnostics, temperature profiles and fluctuations
– Reciprocating turbulence probe
– Fluctuation physics and diagnostics

3.2 Other Collaborations

- Auburn University – ECE diagnostics, temperature fluctuations (C. Watts)
- Australia National University – Plasma flow and biasing, MDSplus (B. Blackwell, M. Schats)
- Carlos III University, Madrid – Interferometry (P. Acedo, H. Lamela)
- C.E.A. Cadarache – X-Ray imaging diagnostics, Lower Hybrid modeling (Y. Peyson)
- Chalmers University, Sweden – Resistive MHD modeling (Bondesson, Fulop)
- C.R.P.P. Lausanne – MDSplus
- Culham Lab – MHD, rotation theory (J. Hastie, P. Helander)
- DIII-D – Coordinated SOL/divertor studies, ICRF physics, dimensionless similarity studies, confinement physics, density limit studies, pedestals, MDSplus (D. Whyte, P.

Stangeby, R. Pinsker, J. Deboo, C. Petty, R. Groebner, R. Moyer, J. Schachter)

- Ecole Royale Militaire, Brussels – ICRF modeling (Evrard, Ongena)
- Griefswald – Divertor modeling (X. Bonnin)
- International Database Groups – Databases (Cordey, Conner, Kamada)
- IGI Padua – X-Ray tomography, MDSplus (P. Franz, P. Martin)
- JET – Similarity studies, modeling, edge physics, Coordinated EDA studies, Coordinated SOL transport studies, MDSplus (G. Cordey, J. Christensen, Huysmans, G. Matthews)
- JT60-U – Lower Hybrid current drive modeling (S. Ide), Edge probe studies (N. Asakura), Disruptions studies (Yoshino, Nakamura, Neyatani)
- Keldysh Institute – Atomic physics and radiation transport (Novikov, Barob'ev)
- KFA Jülich – Rotation modeling, Plasma-neutral interactions (A. Rogister, D. Reiter)
- KSTAR (Korea) – MDSplus
- Kyoto U – Impurity spectroscopy (K. Kondo)
- Lehigh U – Transport modeling (G. Bateman, A. Kritz)
- LLNL – high-Z atomic physics and radiation transfer (K. Fournier, A. Wan, H. Scott), Transport and divertor studies (R. Cohen, W. Nevins, Xu)
- Lockheed Martin Idaho Technologies, Co. – Dust analysis (W. Carmack, D. Petti, K. McCarthy)
- Lodestar – Transport and divertor modeling (Myer)
- Los Alamos National Lab – Visible/IR imaging diagnostics, disruption studies (G. Wurden, R. Maqueda).
- Max-Planck Institut, Garching – ICRF heating, ICRF modeling, High Z first wall studies, Dimensionless similarity studies, Coordinated SOL transport studies (R. Neu, A. Kellman, D. Hartmann, J.-M. Noterdaeme, M. Brambilla, W. Suttrop, Mertens)
- MIT-PSFC Theory and Modeling Group – Transport, divertor, MHD, RF (P. Catto, J. Ramos, J. Hastie, F. Porcelli, P. Bonoli)
- MIT-Physics – Transport theory and modeling (Coppi, Sugiyama)
- NET –B2/EIRENNE edge modeling (A. Loarte)
- New York University – Theory of induced rotation (C.S. Chang)
- NIFS/LHD – Impurity studies, Diagnostics, Atomic physics, MDSplus (N. Noda, Y. Yamauchi, B. Peterson, H. Funaba, T. Kato)
- Notre Dame U – Atomic physics modeling (U. Safranova)
- ORNL – Neutrals and H-mode threshold theory and modeling, Self-organized criticality (B. Carreras, L. Owens); ICRF technology (R. Goulding, P. Ryan)
- Sandia National Labs. - Albuquerque – Surface analysis (W. Wampler)
- TEXTOR – Atomic physics modeling for He beam edge diagnostic (Schweer)
- U of Alaska – Internal transport barrier dynamics (D. Newman)

- U.C.S.D. – Divertor/edge theory, modeling, Coordinated SOL transport studies (S. Krasheninnikov, G. Antar, D. Whyte)
- U of Maryland – High resolution spectroscopic diagnostics, plasma flows, H/D ratios (H. Griem, R. Elton)
- U of Maryland – Transport, H-mode thresholds, and density limits (J. Drake, B. Rodgers, W. Dorland)
- U of Idaho – ECE diagnostics, temperature fluctuations (R. Gandy, Y. In)
- U of Texas, IFS – TAE theory (H. Berk), Linear gyrokinetic analysis (M. Kotschenreuther), Transport (D. Ross)
- U of Toronto – Edge Modeling (P. Stangeby, S. Lisgo, Elder)
- U of Washington – MDSplus
- U of Wisconsin – BES hardware and analysis techniques, MDSplus (R. Fonck, G. McKee)

3.3 Proposed JET Collaborations

In addition to the ongoing MDSplus collaboration, described in section 3.5, and supported under our base funding, we propose several new or extended activities with JET. These additional activities require incremental funding, as detailed in the B budget submissions.

3.3.1 Enhanced D-alpha H-Modes in JET

Purpose

The purpose of this proposal is to better understand the nature of the Enhanced D-alpha (EDA) H-mode observed on C-Mod. Is this mode applicable to larger scale and lower collisionality experiments? Any differences between C-Mod and JET will provide information as to why this mode does, or does not, occur, as well as furthering the understanding of H-modes in general. This proposal has been developed in collaboration with the S1 Task Force and the UKAEA-Association Euratom.

Scientific Objectives

The EDA H-mode regime in Alcator C-Mod provides good thermal confinement simultaneously with reduced particle confinement (compared to ELM-free H-modes). The absence of ELMs is highly advantageous from the point of view of divertor heat loads. The continuous degradation of particle confinement provides control of core impurity and density levels. (There is some evidence for similar phenomena on DIII-D, JT-60U, and on JET prior to their first divertor installation).

A similar regime to EDA H-mode, the Low Particle Confinement H-mode (LPCH), was discovered in JET in 1989 when the plasma volume was larger and higher triangularity was possible because no divertor was present. This occurred with ICRF heating, high triangularity, and high safety factor. All of these characteristics correspond to those seen in C-Mod. One difference, however, was that they occured when gas puffing was switched off or just after nickel impurity injection. We attempted to reproduce this regime though the high triangularity of the LPCH shaped plasmas could not be obtained because of the divertor. We achieved long steady H-modes with very rapid ($\sim$ 1 kHz) and small Type I ELMs in the first series of experiments. In the second attempt early this year, we achieved long ELM-free H-modes, but did not have sufficient run time to raise the density sufficiently to find the ELM-free to Type I ELM boundary. Impurity injection experiments were also not yet accomplished in these runs. While there were some discharges with 40 - 60 kHz low m, n edge modes, they did not appear to be the JET equivalent of a quasi-coherent mode. Since JET plans a long shutdown for the remainder of FY2001, the next opportunity for possible experiments will not be until well into FY2002.

Proposed Collaboration

If further runtime is approved on JET, we would like to continue to participate in task force S1 experiments on this topic. The operational experience from C-Mod is considered essential by the JET staff for this project. The planning process includes several steps - task proposal, approval, solicitation of tenders, and finally the specifics of the days' operation. We plan to be a part of most of these steps. The Alcator staff (I. Hutchinson and J. Snipes) will participate remotely or onsite in these experiments. After the run, there will be some preliminary analysis of data followed by further analysis jointly by MIT and S1 staffs.

Benefits to the U.S. Program

If the EDA mode can be understood well enough to transfer it to other tokamaks this will be very advantageous. The US program will gain in having an operating regime with a number of important advantages over ELM-free and type II ELMy discharges - good thermal confinement simultaneously with reduced particle confinement, no ELMs affecting antenna coupling and divertor heat loads.

3.3.2 Cross-field transport in the SOL and its relationship to main chamber recycling and the density limit

Purpose

The purpose of this collaboration with JET is to further the understanding of cross-field transport in the SOL. Closely related as part of this goal is to understand how cross-field transport affects wall-recycling, the density limit, and usefulness of a divertor.

Scientific Objectives

A number of tokamaks are finding that cross-field transport in the SOL can be much higher than expected, possibly leading to radial fluxes competing with flow into the divertor. The result is high main-chamber recycling. Such effects have very important implications for the tokamak as a reactor, for control of particle and heat flows, for control of main wall impurity sources (which are more poorly screened than that originating from the divertor), and also potentially for controlling core energy transport. In addition results from Alctor C-Mod show that cross-field heat transport cna become so strong near the density limit that it could lead to core cooling and be an important factor leading to core thermal collapse (density limit).

In the past year the collaboration has involved several efforts: 1) B. Lipschultz visited JET for a short period (2 days) and worked with G. McCracken to select a dataset for comparison with a number of tokamaks; 2) S. Pitcher visited JET for 6 weeks and more deeply studied the JET database for trends in SOL transport, neutral levels in the divertor and SOL, and any evidence for significant neutral flux escaping from the divertor; 3) B. Lipschultz has worked with the C-Mod staff to develop a gauge controller that could be installed on JET to improve on that measurement there. The comparison of JET to other tokamaks was presented at the Sorrento IAEA conference. These data indicated that all tokamaks evidence similar scalings for the neutral fluxes in the main chamber. Such fluxes are an indication that there is significant wall recycling on all divertor tokamaks. The in-depth study of the JET database by Spencer Pitcher indicated, on the other hand, that the neutral pressures could be explained completely by divertor leakage. So, at this time, we do not have a complete understanding of the JET SOL transport. Parallel to the above work, MIT and DIII-D (Whyte/Stangeby) have compared SOL data and executed a miniproposal to gather data on a common basis.

Proposed Collaboration

The proposal of collaborative work on JET falls into several areas. First, we plan to expand the collaborative work of gathering similar datasets from C-Mod to DIII-D to JET. The experience in working with DIII-D will be applied to determining the best scaling parameters to JET to identify the correct operating conditions for comparison of the SOL characterisitics. A copy of the MIT/D3D miniproposal has already been sent to JET Task force leasders and discussion of whether this will be something they want to plan for their program has started. The miniproposal. if successful, would provide data on two levels - first on the macroscopic level od the steepness or flatness of plasma profiels in the SOL. Secondly, we hope such work will provide some insight into the underlying physics. The latter would be achieved through analysis of the profile data to obtain cross-field diffusion coefficients. We hope to obtain turbulence measurements for correlation with the macroscopic profiles as well. If such turbulence measurements are available then we will be able to estimate the role of cross-field heat transport at the higher densities and at the density limit for comparison with C-Mod and DIII-D.

One of the deficiencies of comparing JET edge data to other experiments is in the area of pressure measurements. The JET Task Force E leaders have expressed an interest in MIT providing an inexpensive, simple gas pressure diagnostic for JET. We have spent time in the last year on the packaging of this diagnostic that will allow implementation on JET. We plan to continue this development. We have started discussions with the JETY team regarding its implementation on JET. If all works well we would work to install this pressure diagnostic on JET during the long shutdown they are planning for the installation of the new EP divertor.

Benefits to the US Program

An important part of the process to evaluate the usefulness of the tokamak configuration is the efficacy of the divertor in removing the plasma-wall interaction to the divertor from the main chamber. These new results from C-Mod and elsewhere indicate that our understanding of perpendicular transport is flawed. Understanding how such main chamber recycling characteristics scale from C-Mod to JET will be crucial in assessing the importance of this issue for the future of tokamaks. The universality (or lack thereof) of such recycling will be better understood if we have information about the physics underlying such macroscopic processes. We believe that determining the local diffusion coefficientand turbulence measurements will give us that needed information. An added benefit of determining the diffusion coefficient will be that scaling studies for this parameter can be done as well. This has important implications for the heat flux profiles in the divertor. We plan to work with members of the US theory/modelling community to test our understanding of the processes that lead to the measured diffusion coefficients.

3.4 Proposed FTU Collaboration

3.4.1 Advanced Tokamak Physics with Lower Hybrid Current Drive in FTU

Purpose

The purpose of this collaboration with FTU is to exchange knowledge and expertise in the analysis and understanding of the physics of Lower Hybrid Current Drive (LHCD) as applied to 'Advanced Tokamak' operation to obtain internal transport barriers (ITB's) and attempt to demonstrate the feasibility of steady-state tokamak operation.

Scientific Objectives

FTU, like C-Mod, runs with toroidal fields between 4 - 8 T at densities exceeding 1 $\times$ 10^{20} m^{-3}. In FY2002 and 2003, FTU intends to operate at full power with up to 5 MW source LH power, 1.6 MW Electron Cyclotron Heating (ECH) power, and 1.5 MW Ion Bernstein Wave (IBW) power. A major thrust of the FTU program is to attempt to obtain and control Internal Transport Barriers (ITB's) to demonstrate the feasibility of steady-state 'Advanced Tokamak' operation. This will involve LHCD in the plasma startup together with multiple pellet injection as well as the combination of the ECH and IBW heating schemes. The analysis of the improved confinement regimes that are associated with ITB's and the accompanying MHD activity will be the main thrust of this collaboration. The different heating schemes on the two machines will provide interesting comparisons to be made in the synergistic effects of different heating schemes combined with LHCD. Practical expertise on how to obtain and maintain ITB's with LHCD will be obtained that may then be applied to C-Mod when the LH system becomes operational.

Proposed Collaboration

The activities to be performed during visits to FTU include operating the tokamak with LHCD and analyzing confinement properties and MHD activity associated with the high confinement modes of operation. The visits will be for one physicist (J. Snipes) to work at Frascati for a period of 2 months in each fiscal year during FTU operation with LHCD when C-Mod is not operating. The present plans for operation of FTU are from 9/01 - 12/01 when C-Mod will be shutdown, so the collaboration would begin in that time period. The program beyond this has yet to be determined, though extensive experimental operations are expected. The exact schedule will depend also on C-Mod operation plans.

Benefits to the U.S. Program

This collaboration will benefit the U.S. fusion program by obtaining expertise in LHCD from the FTU team as well as share in physics results between these two similar tokamaks albeit with significant differences. Since C-Mod will soon be adding LHCD to its program, the expertise obtained will be quite valuable to the future C-Mod program. The similarities between the tokamaks and their operational regimes may allow experimental techniques for obtaining and maintaining ITB's learned on FTU to be directly applied to C-Mod. The differences in the heating schemes will also provide some contrast to determine how important are the synergistic effects of combined heating with LHCD.

3.5 MDSplus

Recent Accomplishments

Over the last year MDSplus has continued to gain acceptance in the worldwide fusion community. There have been new installations in the US, Europe, and Japan. There have been developments in support for new data acquisition hardware, software installation scenarios, and general code maintenance.

During the last year significant effort was put into support for existing MDSplus installations including DIII-D, NSTX, LDX, Heliac (Australia), TCV (Switzerland), RFX (Italy), UCLA, University of Washington, Hefei (China). These sites run MDSplus under a variety of operating systems, necessitating both bug fixes and new tuning scenarios. New MDSplus installations were done for CHS (NIFS), HBTEP (Columbia), MST (University of Wisconsin-Madison), and JET (England). At CHS the installation involved completely new CAMAC communication infrastructure using a VME to CAMAC communications grid. The installation at JET was to provide remote data access to the JET archived data under JET RP Task E05.05.01.

Several areas of new development were also pursued. A prototype linux based PCI data acquisition server was implemented, and procurement of a full implementation in CPCI was begun. Support for SCSI CAMAC serial highway under linux was begun. An install shield installation kit for WIN32, as well as rpm kits for TRUE64, linux, and HPUX were developed. A method for supporting new data acquisition hardware with no C language programming was implemented.

Over the next two years the effort will be divided between support for existing and new installations, system enhancements, and maintenance of the software. A major thrust for the system enhancements will be in the area of computer security, for both MDSplus and the associated relational database applications.

As MDSplus is being used at more sites, on heterogeneous computing platforms the need for support, maintenance and documentation increases. We plan produce both user and system level documents. Over the next two years we plan to put significant effort into these areas. We hope that some of the other installations will contribute, particularly to the documentation effort. To this end, General Atomics has contracted a technical writer to begin documenting some of their data access tools. We are planning an MDSplus user meeting in conjunction with the IAEA topical meeting on data acquisition in July 2001. Locally at MIT we are planning to utilize a variety of computing platforms including linux - PC, linux-ALPHA, and Windows 2000. We will continue to work on the development and optimization of cross platform data client tools.

MDSplus is emerging as a standard for remote data access in the world fusion community. The need for adding strong authentication, as well as optional data encryption is becoming apparent. We plan to add certificate-based authorization, with public key encryption to both the MDSplus data server and our commercial database applications. We have proposed a collaboration with ANL and LBL to integrate the GLOBUS GSI toolkit for authentication and the AKENTI authorization tools with our servers.

The CAMAC based data acquisition hardware is aging. Over the next two years we plan to use CPCI based hardware for both new applications, and to replace equipment as it fails. The prototype software for these new devices will be hardened, and as new modules are acquired they will be supported. A CPCI based replacement for the existing CAMAC timers is being developed at RFX in Padova. These timers will be compatible with the existing hardware and provide clocks up to 10 MHz. We plan to fully support them in MDSplus. The linux driver for SCSI CAMAC serial highway will be completed.

Compute intensive between shot user data analysis tasks will be migrated to a new linux based compute farm. This will offload some of the demands on the control room desktop machines. A cross platform action dispatching mechanism will be implemented. These developments, along with the heterogeneous client applications provide a migration path away from the OpenVMS operating system.

4. Run Planning

Standard operation on C-Mod consists of single shift (8.5 hours) run-days, typically four days per week, with Mondays normally reserved for maintenance activities. Between 20 and 30 plasma shots are normally produced during a run day. In FY 2000, a total of eighteen physics operating weeks were budgeted, which corresponds to about 600 hours of run time. For FY 2001 and FY2002 we are projecting 12 and 10 weeks weeks of physics operation respectively, at the guidance A budget levels. Operational time is funding-limited, with the direct costs being dominated by liquid nitrogen, electricity, and labor.

Because of the limited amount of run time, relatively detailed planning must be carried out prior to each experimental campaign to establish priorities and maximize productivity. The C-Mod Experimental Program Committee (EPC), consisting of the Group Leaders, the Project Head, and representatives from the major collaborating institutions, is responsible for setting goals and overall strategy.

The run planning process for each C-Mod campaign begins with an open "Ideas Forum", at which C-Mod staff, collaborators, and members of the community present ideas for experiments. Ideas can be at any stage of development, with no limit to the number of presentations per individual. At the last such forum, held in December 1999, 85 presentations were made by forty-five speakers, representing eight institutions including M.I.T.; in addition, a roughly equal number of experimental proposals which had previously been put forward for the previous campaign, and which were still considered worthwhile, were summarized by the topical area coordinators.

The Forum, and subsequent planning activities, are organized among about five principal themes, which correspond to the major research topics of the C-Mod Experiment. In addition, ad hoc task forces are organized to address critical research needs of a multi-disicplinary nature. This year's organizing research themes were:

Divertor & Edge Studies
Pedestal Transport & Stability
Core Transport
ICRF Current & Flow Drive
High Performance & Operations

Typically the number of proposed ideas in a given area substantially exceeds the available run time. Working groups identify opportunities for consolidation, and prioritize the key experiments for each theme area. Contingency plans and alternative experiments in each area are also identified at this level. The resulting prioritized lists are then reported back to the EPC, together with suggested changes in the initial allocations. At this point, the feasibility of further consolidation can be assessed, along with considerations of overall balance. This process is iterated until a list of high-priority experiments is developed. A tentative schedule for at least the first part of the campaign is then prepared.

Miniproposals are then prepared for the high-priority experiments. These detailed miniproposals are the basis for final approval and scheduling of runs. The MP's describe the purpose of the experiments, approach, required resources, an outline of the planned shot sequence, and a statement of the goals and anticipated impact of the experiment. Miniproposals normally are written for experiments requiring from one to a few run-days.

Runs are scheduled for approved miniproposals, with the schedule being continuously updated throughout the campaign. Depending on the available time, some days are kept available as contingency to accommodate new experiments as well as to provide make-up opportunities for experiments which are not completed as planned.

The 2000 campaign began plasma operation in late March, as scheduled, with physics plasmas starting in May. As anticipated, because of the number of new installations this year, we needed to exercise the option of a manned-access vent in the summer, after about nine weeks of physics operation. The vent allowed us to undertake modification of the new four-strap ICRF antenna at J-port. Operations resumed in August, 2000, and continued through December, 2000. A total of 84 physics run days were accomplished during *Calendar Year* 2000.

Additional modifications to the J-port antenna are being carried out, and operation is scheduled to resume in May, 2001, for a short campaign which will complete the budgeted 12 weeks for FY01. The priorities for this upcoming run period include evaluation of the antenna modifications, additional work on critical physics tasks from the 2000 Operation, and evaluation of long-pulse operation in preparation for AT experiments with LHCD in 2003. In parallel, we will be evaluating the operation of the Diagnostic Neutral Beam and associated diagnostics.

Since the main goals of the FY 2001 Experimental Campaign are closely related to those of the 2000 Campaign, no Ideas Forum will be held prior to this operating period. An extended vent for the purpose of installing new in-vessel hardware, and inspection of the magnet systems, is scheduled to begin in Summer 2001 and continue through Spring, 2002. A new planning activity for the subsequent experimental campaign will begin with an Ideas Forum, dates to be announced, approximately three to four months prior to the beginning of the 2002 Operating Period. In view of the increasing emphasis in the C-Mod program on AT issues, we anticipate that an Advanced Tokamak topic will be added to the list of major research themes for the next Forum.

5. Explanation of Milestones

In this section we report the current status of the Level I milestones pertaining to the Alcator C-Mod Base Program. A summary of the current milestones is presented in the accompanying table.

Milestones Completed Since March, 2000

69. The diagnostic neutral beam will have been successfully operated into plasma, with beam energy of 40keV or higher.

    During the current campaign, the diagnostic neutral beam successfully injected a hydrogen beam into the C-Mod plasma. For operation into plasma, the typical beam energy was 46 kV with currents between 4.5 and 5.2 A. Immediately following the start of the temporary shutdown, offline operation began to allow improvements in performance. At present, the beam can be operated without problems at 50 kV (typically, between 49.6 and 49.9 kV). The current is near 5.5 A. The best full energy component achieved thus far is 27 % of the beam density. In summary, the beam was installed on C-Mod on 2/16/00. Operation in the C-Mod cell began on 3/17/00. Beam operation into C-Mod began on 4/28/2000. Initial operation was in D_2 *to check gas control. Operation in* H_2 *began 6/13/00.*

Deleted/Replaced Milestones

The following milestone has been deleted.

65. Conduct ρ^* dimensionless scaling experiments in concert with other diverted tokamaks: JET, DIII-D, ASDEX-U, JT-60, etc. Specifically, operate at reactor-like values of ν^*, β_n, q, shape, aspect ratio, etc. and evaluate transport for achievable values of ρ^*. The nominal values of β_N=2, with q^*=3 and 4, could be obtained at 5 Tesla using up to 8 MW of source power at 80MHz, or at lower field using up to 4MW of tunable frequency ICRF, or both, depending on the scaling of confinement.

    The motivation for this work was derived from the "Urgent task 3.2" developed by the ITER EDA activity, which is no longer operative; furthermore, the U.S. is no longer participating in ITER Research and Development. The specific parameters called for in the Milestone are no longer directly relevant to any proposed Next Step Device. While the physics of transport scaling with ρ^* *is still of interest, and will be addressed as a natural part of the C-Mod program, the specific experiments called for in Milestone 65 no longer merit special status, either from a scientific or programmatic point of view.*

Outstanding Milestones

62. Operate with plasma current of 2 MA: Operation with $I_p \geq 2$ MA for a flat-top time $\geq$ 0.5 seconds will have been achieved in our standard single null divertor configuration.

Detailed inner wall strain measurements, being made during the FY99 run campaign, will yield important input for the design process.

64. Assess D-^{3}He mode conversion heating and current drive: Evaluate the effectiveness of heating and current drive in the D-^{3}He mode conversion regime compared to the minority heating regimes.

    Mode conversion heating experiments were begun in FY96.

67. With a total of 8 MW ICRF source power, full performance in H-Mode plasmas, with currents in the range above 1.2 MA, will be evaluated.
68. The four strap PPPL antenna will allow for experiments to evaluate mode conversion current drive with regard to efficiency and prospects for current profile modification. MCCD will be documented and the results reported in one or more publications.
70. CXRS systems are being added to look at toroidal and poloidal rotation, in conjunction with the Texas diagnostic neutral beam. The edge poloidal rotation system should yield high spatial resolution measurements, which will be particularly important for H-mode pedestal studies.
71. An MSE system, also operated in conjunction with the DNB, is being installed and the system will undergo evaluation.
73. Mode conversion electron heating experiments will be performed with our ICRF systems. During these experiments poloidal flow will be monitored using some combination of our charge exchange recombination and high resolution X-ray spectroscopy diagnostics. Flow drive efficiency will be quantified, and evidence for influence on internal transport barriers will be assessed.
74. Operate with increased triangularity, up to the range 0.8-0.9, as permitted by the modification of the inner divertor, and evaluate resulting performance. In H-mode discharges, the higher triangularity should provide improved MHD stability in the steep gradient pedestal, and is expected to modify the ELM behavior and the ELM-free/EDA boundary. High triangularity will also provide enhanced stability for AT optimized core shear equilibria and may impact internal transport barrier formation. The more open divertor configuration may also modify particle control properties.
75. Evaluate potential methods to control wall-gas inventory and target plasma density in anticipation of long pulse operation. Wall reservoirs for particles will be characterized and potential conditioning techniques and fueling options will be evaluated.

Proposed New Milestones

76. Plasmas with Internal Transport Barriers (ITBs) show promise as the goal for future advanced tokamak steady state reactor operation. Plasmas with dual edge and internal, energy and particle barriers have been formed in Alcator C-Mod with auxiliary radio wave heating, in the absence of the usual neutral beam particle and momentum sources (which will be unavailable in future reactors). The prescription for achieving these ITBs is to lower the magnetic field, which causes the radio waves (at 80 MHz) to be concentrated in the inner portion of the plasma, off-axis. Further heating power at 70 MHz will be concentrated at the core of the plasma to increase the temperature of

plasma center, inside of the ITB, to hold the density profile steady and to arrest impurity accumulation. The fundamental role of plasma rotation in the ITB formation will also be investigated.

77. Evaluate operation of the modified J-port 4 strap antenna. Operate the modified 4-strap antenna at 78 MHz with improved arc detection and additional diagnostics up to the maximum power that can reasonably be achieved. Using heating phase, evaluate the heating efficiency, power handling, reliability, and impurity generation of the 4-strap antenna.
78. All components of the inner wall modifications will be installed and ready for operation. These modifications will strengthen the inner divertor and wall, allowing for operation at higher plasma currents, while simultaneously increasing plasma shaping flexibility for our standard lower single-null divertor configurations.
79. The Lower Hybrid fabrication project, a collaboration between MIT and PPPL, will be completed. This project entails implementation of 3 MW of klystron source power at 4.6 GHz, with one waveguide array launcher, designed for current drive as a critical tool in the C-Mod long pulse Advanced Tokamak program.

Alcator C-Mod Program Milestones

	Milestones Completed since DEC 2000	Baseline	MAR 2000 Target	Date Completed	Comments
69.0	Diagnostic neutral beam operational into plasma	NOV 1999	APR 2000	APR 2000	

	Deleted Milestone	Baseline	Target Date Last Report	Comments
65.0	Conduct ρ^* scaling experiments at ITER-like dimensionless parameters	JUN 1997	NOV 2000	no longer a high programmatic priority

	Outstanding Milestones	Baseline	Target Date Last Report	Current Target	Comments
62.0	Operate with plasma current of 2 MA	JUN 1997	SEP 2001	SEP 2002	requires inner wall mods
67.0	Evaluate integrated H-mode performance with 6 MW ICRF, $I_p \geq 1.2$ MA	JUN 1999	AUG 2000	AUG 2001	
68.0	Evaluate mode conversion current drive	JUN 1999	OCT 2000	SEP 2002	
70.0	Measure edge rotation profile with CXRS	JAN 2000	JUN 2000	JUL 2001	
71.0	Measure current density profile with MSE	JAN 2000	OCT 2000	SEP 2002	
73.0	Evaluate mode conversion flow drive	SEP 2001	SEP 2001	SEP 2001	
74.0	Evaluate performance at high triangularity	SEP 2001	SEP 2001	SEP 2002	requires inner wall mods
75.0	Evaluate density control	APR 2002	APR 2002	AUG 2001	

	Proposed New Milestones	Baseline	Comments
76.0	Investigation of ITB control with multiple frequency ICRF	AUG 2001	
77.0	Evaluate operation of modified J-port 4-strap antenna	AUG 2001	
78.0	Complete inner wall modifications	APR 2002	
79.0	Completion of Lower Hybrid Fabrication Project	MAR 2003	

6. Alcator C-Mod Cost Summary & Budgets

MIT BUDGET SUMMARY
October 1, 2000 - September 30, 2003
Dollars in Thousands

	Research	Facilities Operations	Lower Hybrid	JET	MDS plus	Total
FY2001	4,736	8,291	1,347	55	150	14,579
FY2002 Ver. A	5,096	8,582	730	55	150	14,613
FY2002 Ver. B	5,497	9,993	730	130	150	16,500
FY2003 Ver. A	4,917	9,091	400	55	150	14,613
FY2003 Ver. B	5,639	11,176	400	135	150	17,500
Plan A Total	14,749	25,964	2,477	165	450	43,805
Plan B Total	15,872	29,460	2,477	320	450	48,579

Appendix A
Alcator Publications - 2000

Papers Published in Refereed Journals

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Eisner, E.C., et al., “Operation of the DNB on Alcator C-Mod.”

Elder, J.D., et al., “Onion-Skin Method and EIRENE Modeling of the Alcator C-Mod Edge Region.”

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Hughes, J.W., et al., “High Resolution Edge Thomson Scattering Measurements on the Alcator C-Mod Tokamak.”

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Lin, Y., et al., “Study of Enhanced D_α H-modes Using the Alcator C-Mod Reflectometer.”

Lisgo, D., “Onion-Skin Method and EIRENE Modeling of the Alcator C-Mod Region.”

Marmar, E.S., et al., “Experimental Phenomenology of the Enhanced D-Alpha H-Mode in Alcator C-Mod.”

Mazurenko, A., et al., “Fluctuations and Fast Wave Measurements by the Phase Contrast Imaging on Alcator C-Mod.”

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Porkolab, M., et al., “ICRF Driven Internal Thermal Barriers in Alcator C-Mod.”

Shugart, A.J., et al., “Computer Modeling of the C-Mod Phase Contrast Imaging System.”

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Wukitch, S., et al., “ICRF Current and Poloidal Flow Drive in Alcator C-Mod.”

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Boivin, R.L., et al., “Recent Results from the Alcator C-Mod Tokamak.”

Bonoli, P.T., et al., “Advanced Full-Wave Simulations of Mode Conversion Electron Heating and Current Drive in Alcator C-Mod.”

Granetz, R.S., et al., “Disruption Neutral Point Experiment on Alcator C-Mod.”

Greenwald, M., et al., “Studies of EDA H-mode and Its Relation to the Micro-stability of the Pedestal.”

Mikkelsen, D.R., et al., “Ion Temperature Gradient Scale Length in C-Mod: Testing Nonlinear Theory.”

Nelson-Melby, E., et al., “Observations of Mode-converted Ion Bernstein Waves in Alcator C-Mod with Phase Contrast Imaging Diagnostic.”

Snipes, J.A., “A Comparison of High Recycling H-mode Regimes on Alcator C-Mod and JET.”

Rice, J., et al, “Central Toroidal Rotation Reversal with ITB Formation in Alcator C-Mod Plasmas.”

Schilling, G., et al., “Overview of Results from the Upgraded ICRF System on Alcator C-Mod.”

Snipes, J.A., et al., “Peaked Density Profiles in H-mode in Alcator C-Mod.”

Zweben, S.J., et al., “Two Dimensional Imaging of Edge Turbulence in Alcator C-Mod.”

APS Invited Talks

Fiore, C.L., “Core Internal Transport Barriers in Alcator C-Mod Plasmas.”

Hubbard, A.E., “Pedestal Profiles and Fluctuations in C-Mod Enhanced D-alpha H-modes.”

LaBombard, B., “Particle Transport in the Scrape-Off Layer and its Relationship to Discharge Density Limit in Alcator C-Mod.”

Workshop Presentations

Transport Task Force Workshop
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Boivin, R., "Effects of Neutral Particles on Edge Dynamics in Alcator C-Mod Plasmas."

Fiore, C.L., "Spontaneous Internal Transport Barriers Following H to L Mode Transition on Alcator C-Mod."

Hubbard, A.E., "Response on the H-mode Pedestal to Input Power."

Mossessian, D., "Measurements of Temperature and Density Pedestal in Alcator C-Mod."

Snipes, J.A., "Fast Particle Driven Modes in Current Rise and EDA H-mode Plasmas in C-Mod."

Invited Talks

Boivin, R., "Effects of Neutral Particles on Edge Dynamics in Alcator C-Mod Plasmas," presented at the Transport Task Force Workshop, Burlington, VT, April 2000.

Fiore, C.L., "Core Internal Transport Barriers in Alcator C-Mod Plasmas," presented at APS, Quebec City, Oct. 2000.

Hubbard, A.E., "Pedestal Profiles and Fluctuations in C-Mod Enhanced D-alpha H-modes," presented at APS, Quebec City, Oct. 2000.

LaBombard, B., "Cross-Field Transport and Main-Chamber Recycling on Diverted Plasmas in Alcator C-Mod," presented at the 14th Int. Conf. on Plasma Surface Interactions in Control. Fusion, Rosenheim, Germany, May 2000.

LaBombard, B., "Particle Transport in the Scrape-Off Layer of Alcator C-Mod," presented at APS, Quebec City, Oct. 2000.

Pitcher, C.S., "The Effect of Baffling on Divertor Leakage in Alcator C-Mod," presented at the 14th Int. Conf. on Plasma Surface Interactions in Control. Fusion, Rosenheim, Germany, May 2000.

Wukitch, S.J., "Operational Experience with a 4-Strap Antenna in Alcator C-Mod," presented at the U.S.-Japan Technology Exchange, PPPL, Oct. 2000.

Other Presentations

Boivin, R.L., “High Resolution Measurements of Radiated Power in Alcator C-Mod,” presented at Max-Planck Institute, Germany, May 2000.

Greenwald, M., “Particle Transport Driven by Quasi-Coherent Fluctuations in EDA H-Modes on Alcator C-Mod,” presented at the European TTF Meeting, Varenna, Italy, Sept. 2000.

Hubbard, A.E., “H-Mode Pedestal Physics on Alcator C-Mod,” presented at a physics colloquium at Princeton Plasma Physics Lab., May 2000.

Hutchinson, I.H., “Toroidal Rotation without Momentum Input on Alcator C-Mod,” presented at a JET colloquium, June 2000.

Rice, J.A., Fournier, K.B., Safronova, U.I., et al., “The Rydberg Series of Heliumlike C1, Ar and S and their High n Satellites in Tokamak Plasmas,” poster presented at the Atomic Processes in Plasma Conference, Reno, NV, March 2000.

Schilling, G., et al., “Recent Results from the Extended ICRF System on Alcator C-Mod,” presented at the U.S.-Japan Workshop, PPPL, March 2000.

Snipes, J.A., “Enhanced Performance in Alcator C-Mod,” presented at the Workshop on Issues for FIRE, PPPL, May 2000,

Snipes, J.A., “Latest H-Mode Threshold Runs on C-Mod,” presented at the International Workshop on Heating and Transport in Tokamaks, Frascati, Italy, October 2000.

Snipes, J.A., “H-Mode Threshold and Confinement Using the International H-Mode Database,” presented at the Workshop on Burning Plasma Science, Austin, TX, Dec. 2000.

Other Publications

Snipes, J., “A Good START for Fusion,” Physics World **13** No. 4 (2000) 24.