(Draft)
ALCATOR C-MOD
FY07-08 WORK PROPOSAL

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1. Introduction

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 a burning plasma tokamak experiment and for attractive fusion reactors.

Organization of the program is through a combination of topical science areas supporting integrated thrusts. The topics relate to the generic plasma science, while the thrusts focus this science on integrated scenarios, particularly in support of ITER design and operation. There are currently four topical science areas: transport; plasma boundary; wave-plasma interactions; and macrostability. Integrated scenarios encompass the ITER baseline inductive H-Modes, and Advanced Tokamak (AT) operation including partially inductive hybrid modes and fully non-inductive weak and reverse shear operation with active profile control. AT operation takes advantage of the unique long-pulse capability of the facility (relative to skin and L/R times), at $B \leq 5$ Tesla, combined with new current drive and density control tools, to investigate the approach to steady-state in fully non-inductive regimes at the no-wall beta limit; this is particularly relevant to the prospects for quasi-steady operation on ITER. The connections among the topical science areas and the integrated scenarios are illustrated in Figure 1.1

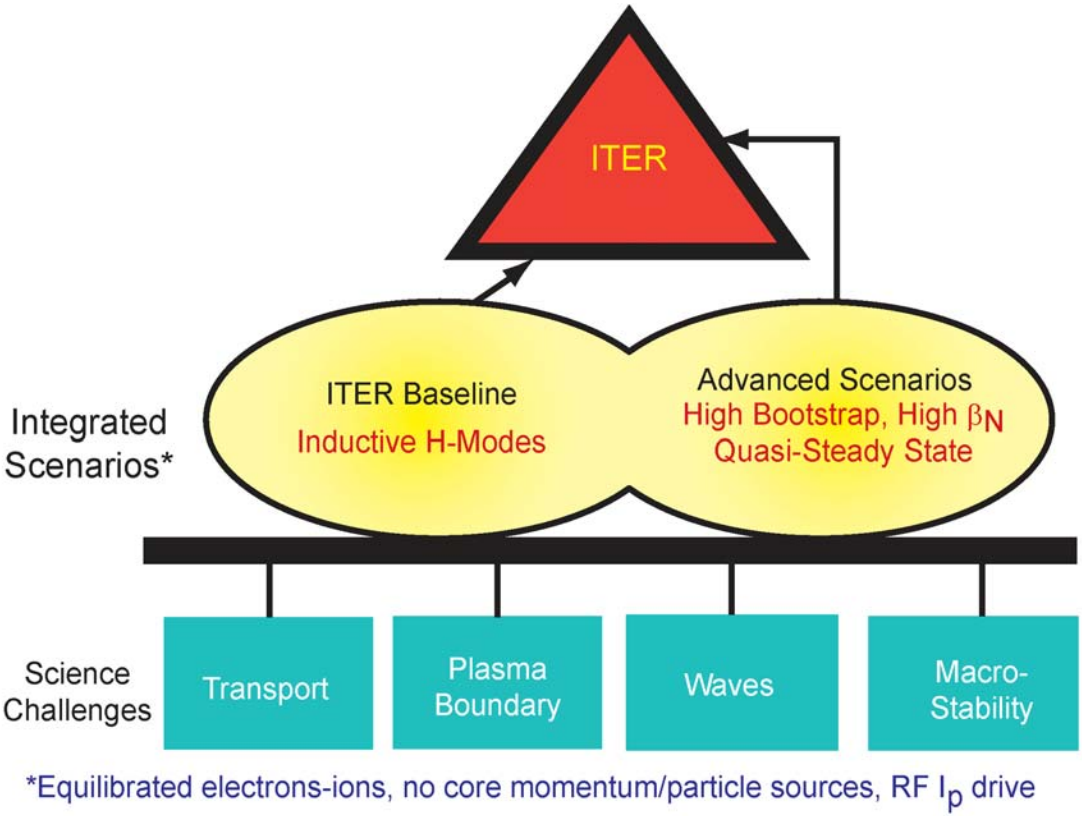


Figure 1.1 Integrated scenarios and topical science areas.

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 (to 10^{21} m^{-3}) and pressures (approaching 10 atmosphere), and with magnetic field spanning the ITER field (5.3 Tesla) and beyond (to 8 Tesla), C-Mod offers a unique test-bed for exploring the physics and engineering which is prototypical of ITER.

- **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 exist in a fusion power plant; the studies of transport, macro-stability and AT physics in C-Mod are thus 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. Many of these experiments are coordinated through the International Tokamak Physics Activity (ITPA).

- **Very high power density scrape-off layer plasma** — With parallel power flows approaching 1 GW/m^2 (as expected in ITER), C-Mod accesses unique divertor regimes which are prototypical of burning plasma 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 plasma facing components** — 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 wall conditioning, density and impurity control challenges. Because of the tritium retention issues, ITER must consider high Z plasma facing components as one option, and studies of hydrogenic retention in C-Mod, both with molybdenum and tungsten, will contribute significantly to this decision.

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 four departments at MIT, as well as from collaborating Universities. Currently 29 graduate students are doing their research on Alcator C-Mod.

High Priority ITER R&D

C-Mod is positioned to investigate many of the key outstanding issues that need resolution to support successful operation of ITER. Research has begun on most of these, and all will be studied in the FY06-FY08 period:

- Steady state operation
  - Hybrid scenarios
    - Priority for AT thrust
  - Develop real time j profile control using heating and CD actuators; assess predictability, in particular for off-axis CD
    - Main thrust of LHCD program; also MCCD, FWCD
    - State of the art modeling tools being developed and applied
- Boundary Science

  - Improve understanding of SOL plasma interaction with main chamber
  - Improve understanding of tritium retention and the processes that determine it
    - Understanding D levels on tiles (including sides) for B and Mo
    - Understanding removal of H at low tile temperature
  - Develop improved prescription of inter-ELM transport and SOL perpendicular transport and boundary conditions for input to modeling
  - Dimensionless cross-machine comparisons for SOL physics
- Pedestal and Edge
  - Construct physics-based and empirical scaling of pedestal parameters
    - Priority of transport task group; coordinated experiments through ITPA
  - Improve predictive capability for ELM size and frequency and assess accessibility to regimes with small or no ELMs
    - Emphasis on small ELMs at higher β, and EDA
      - Effects of collisionality studied through joint experiments
  - Improve predictive capability of pedestal structure through profile modeling
    - Supplying data to pedestal profile database
- Transport Physics
  - Address reactor relevant conditions: including electron heating: equilibrated electrons-ions; impurities; density; edge-core interaction; low momentum input; edge fueling
    - >90% of C-Mod operation is in these regimes
  - Commonality of transport physics in hybrid and steady-state scenarios with reactor relevant conditions
  - Comparisons of turbulence measurements with simulations
    - Encourage tests of simulation predictions via comparisons to measurements of turbulence characteristics, code-code comparisons and comparisons to transport scalings
    - Upgraded turbulence diagnostics
    - Increasingly strong interactions with theory and modeling
- Wave-Plasma Interactions
  - ICRF
  - Lower Hybrid RF
    - Coupling studies
    - Higher power (>0.5 MW)
      - Begin current drive investigations
- Macrostability
  - Disruption mitigation and disruption database
    - Unique investigations at absolute plasma pressures comparable to those on ITER
    - Close coupling to 3-d MHD simulations
  - Intermediate toroidal mode number Alfvén Eigenmodes
    - Active antennas, passive magnetics, PCI
    - Fast particle drive (ICRF minority tails)
    - Strong theory and modeling effort

  - Perform MHD stability analysis of H-mode edge transport barrier under type I and tolerable ELM conditions
    - Focusing on small ELM and EDA regimes
    - Access to discrete ELMs in 2004; continue to pursue with higher pressures, lower collisionality
  - Error field/locked modes
    - Extrapolation to ITER
  - Investigate/determine island onset threshold of NTMs … seed island control
    - Study at increased β
    - Sawtooth stabilization (ICRF, LH)
  - New disruption DB including conventional and advanced scenarios and heat loads on wall/targets
    - Contribute data from all scenarios at high absolute power and energy densities
- Confinement Database and Modeling
  - Effects of ν^* vs. n/n_G, β scaling, ρ^* scaling, analysis of ITER reference scenarios
  - Density peaking in the absence of core fueling
- Diagnostics
    - Dust measurement
    - Erosion/redeposition

Of the five science tasks assigned by the international ITER team in 2006 to the US for resolution, C-Mod is a leader or significant contributor to four:

- Disruption mitigation
  - Dennis Whyte (team leader); Bob Granetz (co-PI)
- Effects of radiation transfer on divertor plasma
  - Bruce Lipschultz (team leader); Steve Lisgo (co-PI); Jim Terry (C-Mod data)
- ICRF heating and current drive – Benchmarking of ICRF codes
  - Paul Bonoli (co-PI); Steve Wukitch (C-Mod data)
- Fast Particle Confinement
  - Joe Snipes (C-Mod data and Nova-K simulations)

Much of this research is coordinated through the International Tokamak Physics Activity (ITPA), especially for topics which require joint experiments on multiple facilities. Approved ITPA experiments, in which C-Mod is one of the key participants, can be found in the following table:

ITPA Designation	**Topic for Coordinated Joint Research**
CDB-4	Confinement scaling in ELMy H-modes: ν^* scans at fixed n/n_G
CDB-8	ρ^* scaling along an ITER relevant path at both high and low β
CDB-9	Density profiles at low collisionality
TP-4.1	Similarity experiments with off-axis ICRF-generated density peaking
TP-6.1	Scaling of spontaneous rotation with no external momentum input
TP-8.2	Investigation of rational q effects on ITB formation and expansion

PEP-7	Pedestal width analysis by dimensional edge identity experiments
PEP-10	Radial efflux at the mid-plane and ELM structure
PEP-16	C-Mod/NSTX/MAST small ELM regime comparisons
PEP-17	Small ELM regimes at low pedestal collisionality
DSOL-3	Scaling of radial SOL transport
DSOL-4	Comparison of disruption energy balance and heat flux profile
DSOL-5	Role of Lyman absorption in the divertor
DSOL-11	Disruption mitigation experiments
DSOL-13	Deuterium co-deposition in gaps of plasma facing components
DSOL-15	Inter-machine comparison of blob characteristics
MDC-1	Disruption mitigation by massive gas jet
MDC-5	Comparison of sawtooth control methods for NTM suppression
MDC-6	Low β error field experiments
MDC-10	Measurement of damping rate of intermediate *n* Alfvén eigenmodes
SSO-2.3	ρ^* dependence on confinement, transport and stability in hybrid scenarios

As our research emphasis shifts to concentrate on quasi-steady-state AT regimes, we expect to become more heavily involved with the Steady State Operations ITPA Group.

US Fusion Program Priorities

In April 2005, FESAC identified campaigns. C-Mod plays an integral role in addressing all of the magnetic fusion relevant topical questions:

- T1. How does magnetic field structure impact fusion plasma confinement?
- T2. What limits the maximum pressure that can be achieved in laboratory plasmas?
- T3. How can external control and plasma self-organization be used to improve fusion performance?
- T4. How does turbulence cause heat, particles, and momentum to escape from plasmas?
- T5. How are electromagnetic fields and mass flows generated in plasmas?
- T6. How do magnetic fields in plasmas reconnect and dissipate their energy?
- T7. How can high energy density plasmas be assembled and ignited in the laboratory?
- T8. How do hydrodynamic instabilities affect implosions to high energy density?
- T9. How can heavy ion beams be compressed to the high intensities required to create high energy density matter and fusion conditions?
- T10. How can a 100-million-degree-C burning plasma be interfaced to its room temperature surroundings?
- T11. How do electromagnetic waves interact with plasma?
- T12. How do high-energy particles interact with plasma?
- T13. How does the challenging fusion environment affect plasma chamber systems?
- T14. What are the operating limits for materials in the harsh fusion environment?
- T15. How can systems be engineered to heat, fuel, pump, and confine steady-state or repetitively-pulsed burning plasma?

The C-Mod program makes key, unique contributions to most of the recommended areas of US "opportunities for enhanced progress":

- Carry out additional science and technology activities supporting ITER including diagnostic development, integrated predictive modeling and enabling technologies.
- Predict the formation, structure, and transient evolution of edge transport barriers.
- Mount a focused enhanced effort to understand electron transport.
- Pursue an integrated understanding of plasma self-organization and external control, enabling high-pressure sustained plasmas.
- Study relativistic electron transport and laser-plasma interaction for fast ignition high energy density physics.
- Extend understanding and capability to control and manipulate plasmas with external waves.
- Increase energy ion pulse compression in plasma for high energy density physics experiments.
- Simulate through experiment and modeling the synergistic behavior of alpha-particle dominated burning plasmas.
- Conduct enhanced modeling and laboratory experiments for ITER test blankets.
- Pursue optimization of magnetic confinement configurations.
- Resolve the key plasma-material interactions, which govern material selection and tritium retention for high-power fusion experiments.
- Extend the understanding of reconnection processes and their influence on plasma
- instabilities.
- Carry out experiments and simulation of multi-kilo-electron-volt megabar plasmas.
- Expand the effort to understand the transport of particles and momentum.

Detailed discussions of how Alcator's specific topical science plans address the programmatic objectives are given in the respective sections of this Work Proposal.

Budget and Schedule

The baseline (A) budget for the C-Mod project in FY2007 is based on guidance from the Office of Fusion Energy Sciences, with total national project funding of $22.8M, including $20.2M at MIT, and major collaborations totaling $2.6M. These budgets will accommodate 15 weeks of research operations in FY2007. In the event of a 10% cut from the FY2007 guidance, research run time would be reduced to 10 weeks. For FY2008, we have taken a flat budget (relative to the FY2007A guidance) as the baseline case, with no increase for cost of living. This results in a plan for 13 research weeks. A 10% cut, relative to FY2007A, for FY2008, leads to a plan for 8 research weeks in the FY2008D case. The major items that the guidance and base budgets permit us to fund are shown in table 1.1.

Table 1.1: Major items funded in guidance budgets (FY06+FY07A+FY08A)

Item	Cost (k$)	Notes
Cryopump	520	Required for density control, lower collisionality; complete in FY2006
W divertor modules	75	ITER prototype
Lower Hybrid CD: 2nd launcher	900	Reduce power density (increase coupled power); allow compound spectrum
Polarimeter/Interferometer	850	j(r) at high density; ITER geometry
4-strap ICRF Antenna	750	Preserve full ICRF power capability with addition of 2nd LH launcher
DAC infrastructure	190	Data collection doubling on 2 year time scale
MPP Cluster Upgrade	25	Shared resource with PSFC Theory Group

Within the guidance budgets, run time is very constrained, and many important initiatives cannot be funded. We therefore also propose higher, national B budgets, totaling $28.7M in FY07 and $29.5M in FY08, which permit the following additions (in approximate priority order).

Table 1.2(a): Major items requiring budget increments (FY2007B)

Item	Cost (k$)	Notes
3 weeks additional run time	750	Total of 18 weeks research operation
Outer divertor upgrade	600	Power handling for >8MW, 5 seconds (complete in FY2008)
Active MHD upgrade	50	Add second location: toroidal mode number control
4th MW Lower Hybrid	600	Required for high current fully non-inductive AT (complete in FY2008)
Real-time matching (ICRF)	275	Increased productivity (complete in FY2008)
DAC Infrastructure	170	Increase data capacity; tools for personnel
SOL Thomson Scattering	250	Pedestal and SOL physics
7 weeks additional run time	1600	25 weeks total; full facility utilization
Spare 4.6 GHz Klystron	500	Currently have no spares for 16 klystron system

Table 1.2(b): Major items requiring budget increments (FY2008B)

Item	Cost (k$)	Notes
5 weeks additional run time	1100	Total of 18 weeks research operation
Complete outer divertor upgrade	200	Power handling for >8MW, 5 seconds
Spare 4.6 GHz Klystron	500	Prudent to have 3 spares
Complete and install 4th MW	500	Required for high current fully non-

LH		inductive AT
3 weeks additional run time	750	To 21 weeks total
High resolution x-ray upgrades	100	Additional tangential views for rotation, T_i profiles
ICRF real-time matching	300	Add to remaining antenna(s)
MSE second view	400	Direct E_r measurement
Advanced material divertor	550	ITER tungsten divertor
4 weeks additional run time	950	25 weeks total; full facility utilization

Table 1.3 summarizes the items which would be cut in the event of a 10% budget decrement (case D) for FY2008.

Table 1.3: Major items cut under a 10% decrement in FY2008D (relative to FY2007A)

Item	**Cost (k$)**	**Notes**
5 week decrease of research run time	1200	8 weeks research operation
Personnel cuts	800	2 Engineers, 1 Tech, 1 Scientist, 2 Students
LH 2nd launcher deferred	375	At least 1 year delay

Proposed facility research run time is given in table 1.4. In addition to the guidance cases, we show the incremental (program planning) (B) and the 2008D decremental case.

Table 1.4: Research operation for guidance (06-08A), incremental (07B-08B) and decremental (08D) budget cases

Fiscal Year	**06**	**07A**	**08A**	**07B**	**08B**	**08D**
National Budget ($M)	21.7	22.8	22.8	28.7	29.5	20.5
Research Operation Weeks	14	15	13	25	25	8
Research Operation Hours	450	480	415	800	800	255

Alcator C-Mod is operated as a National Facility, and includes contributions from major collaborations at PPPL and the University of Texas (Austin), as well as from a large number of smaller national and international collaborations. The present Work Proposal covers in detail the MIT responsibilities in the program, and assumes an integrated effort involving all of the collaborators. Sections 4.1 and 4.2 explicitly cover the contributions from Princeton and Texas respectively.

A summary of the planned facility schedule, assuming the guidance (FY07A) and level (FY08A) budgets, is shown in figure 1.4. Items shown in red require incremental funding. Planned research weeks are shown in the green operations blocks.

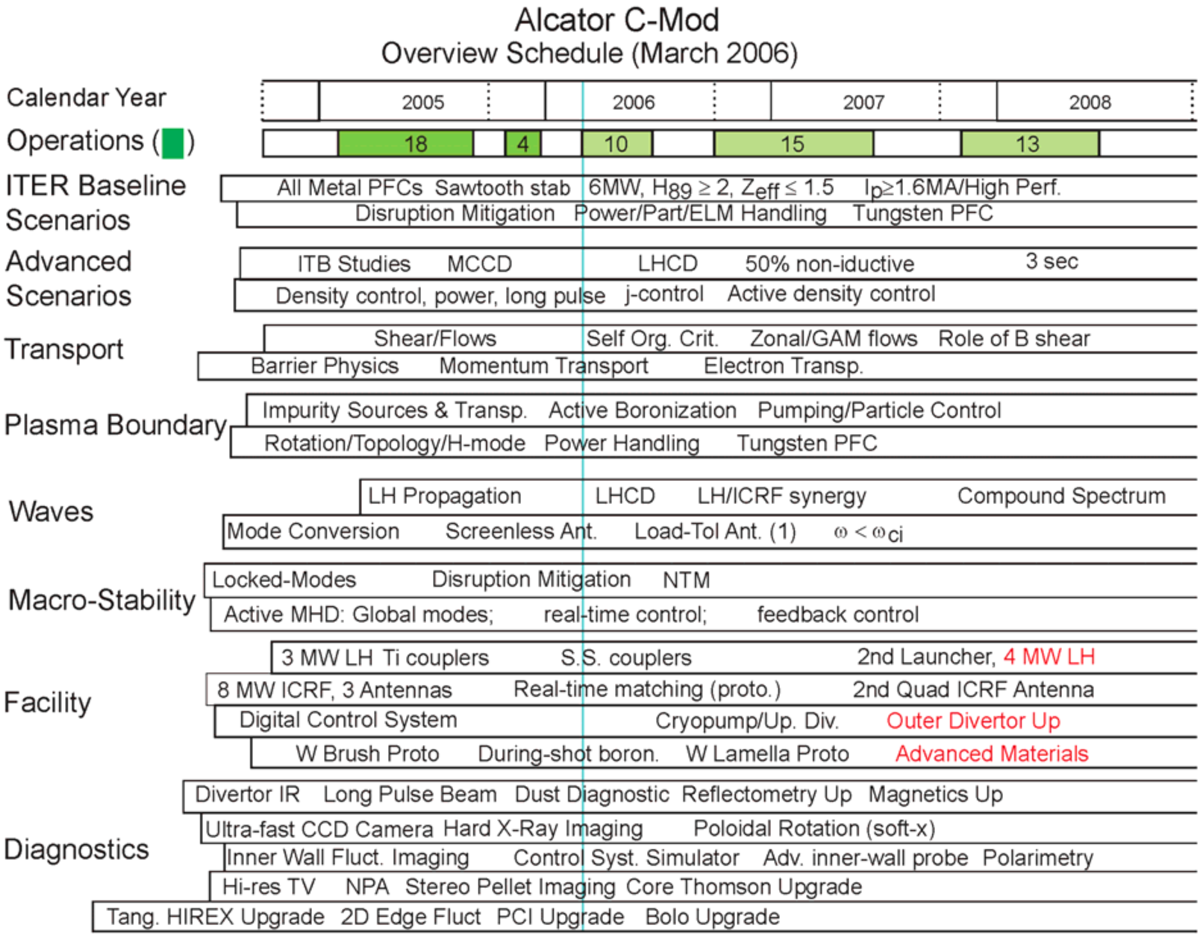


Figure 1.4. Schedule of Programmatic Emphasis and Major Installations [guidance (FY07A) and baseline (FY08A) Budgets]. Items shown in red require incremental funding.

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.

Disruption Mitigation of high pressure plasma [September 06]
Tokamaks are subject to major disruptions, which are sudden, undesirable terminations of the plasma discharge. Disruptions result in severe thermal loading of internal surfaces, large electromagnetic forces on conducting structures, and uncontrolled high-energy beams of electrons. These damaging effects will be particularly severe in burning-plasma-grade devices such as ITER. A number of methods have been proposed and/or tested to mitigate the consequences of disruptions, including injection of high-pressure gas jets. This technique has been shown to work in relatively low pressure, low energy density plasmas, but it is not at all clear that this method will work in high pressure, high energy density burning-plasma-grade discharges. Alcator C-Mod plasmas have absolute pressures and energy densities that are characteristic of those expected in ITER, and therefore will provide an excellent test bed for the gas jet disruption mitigation experiments planned in FY05-06.
Status: Experiments were begun in FY05 and will be continued in FY06. Initial results are very promising, and amelioration of disruptions at intermediate plasma pressures (<P> ~1 atmosphere) have been studied (see the Macroscopic Stability Research section of this proposal).

Sustaining Plasma Current Without a Transformer [March 07]
In standard tokamak operation, the plasma current is induced by a transformer coil, which limits the available pulse length. To operate steady-state, a tokamak needs other means, such as RF current drive and self-generated current. The long-term C-Mod objective calls for fully non-inductive sustainment, with 70% of the current self-generated. In the nearer term, as a first step, we intend to demonstrate discharges on Alcator C-Mod with at least 50% of the current driven non-inductively, using the newly installed antenna, which comprises Phase I of the 4.6 GHz microwave system. This should help to verify the theoretically predicted current-drive efficiency and our ability to control the various plasma parameters needed to optimize it.
Status: These experiments have been delayed as the Lower Hybrid systems were modified to change the plasma-coupling structure materials.

Current Profile Control with Microwaves [September 07]
These experiments are aimed at developing efficient steady-state tokamak operation by launching microwaves into Alcator C-Mod plasmas. The location of current driven by the “Lower Hybrid” waves we will use depends on their wavelength as measured parallel to the magnetic field. We will vary this wavelength and measure the location and amplitude of the driven current, with the intention of demonstrating an improvement of the plasma confinement through current-profile control. By adding independent plasma heating, 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 global instabilities.
Status: These experiments have been delayed as the Lower Hybrid systems were modified to change the plasma-coupling structure materials.

Active Density Control [September 07]
A new divertor cryopump will be installed in C-Mod and pumping properties will be tested during FY06. Beginning in FY07 the configuration will be evaluated for density control, particularly for target plasmas suitable for Advanced Tokamak regimes with efficient lower hybrid current drive combined with high bootstrap fraction.

Confinement at high plasma current [September 08]
The operational space of C-Mod in the plasma current range at and above 1.5 MA has not yet been extensively explored. The potential for improvements in plasma confinement and pressure can be exploited in this regime at magnetic fields of 5.4 tesla and above. With the successful implementation of the non-axisymmetric field error correction coils in FY04, this regime, which was previously precluded because of locked mode induced disruptions, has become accessible in C-Mod for study, and will be exploited in the coming campaigns. Elucidation of the implications of these results for extrapolation to burning plasma regimes, including ITER, will be a major goal of these studies.

Active control of ICRF antenna [September 2008]
To maximize coupled power through an ICRF antenna, the transformation or match of the antenna load to the transmitter needs to be maintained with low reflected power. The antenna load varies with plasma conditions that can evolve during the course of a discharge, especially for the long-pulse quasi-steady-state scenarios, and from discharge to discharge. One means to maintain the match is to use active tuning elements based on ferrite tuners. A system and its characteristics will be tested and evaluated for performance over a range of C-Mod operating conditions.

Goals Accomplished in FY2005

Commissioning of the Microwave Current Drive System [May 05]
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 experiments in FY 2005.
The target was completed on April 10, 2005. The Lower Hybrid Current Drive (LHCD) System has been implemented on Alcator C-Mod. The system consists of a transmitter consisting of 12 klystrons capable of producing a total RF source power of 3 MW at a frequency of 4.6 GHz, and a launcher that divides the RF power into 96 sub-height waveguides for delivery into the plasma. A sophisticated amplitude and phase control system permits precise control of the relative phase in each waveguide, which is

important in determining the propagation and absorption of the launched waves, and ultimately the distribution of RF driven current in Alcator C-Mod plasmas. In initial experiments, measurements of the coupling efficiency have been carried out as a function of phase progression and density at the grill mouth. The density is measured with an array of Langmuir probes which are designed into the coupling structure. The RF measurements were made at power levels of 100-200 kW and showed that very efficient coupling could be obtained. At the optimum current drive phasing of 90° and a plasma density at the grill of about $3x10^{18}$ m^{-3}, over 90 % of the power delivered to the antenna was radiated into the plasma. The dependences on both grill density and phase agree qualitatively with code predictions; more quantitative comparisons will be made when the experiments resume in the 2006 campaign.

Measure plasma behavior with high-Z antenna guards and input power greater than 3.5 MW. [September 05]
These experiments address issues related to first wall choices, and the trade-offs between low-Z and high-Z materials. This choice can affect many important aspects of tokamak operation, including: impurity content and radiation losses from the plasma; hydrogen isotope content in the plasma and retention in the walls; disruption hardiness of device components. All of these issues are significant when considering choices for next step devices to study burning plasma physics, especially ITER. Definitive experimental results will be compared to model predictions. This goal is an FY05 level 1 science target for the DOE Office of Fusion Energy Sciences.
The target was completed on September 20, 2005. All topics in the target have been addressed, and significant scientific results have been obtained through the studies leading to the target completion. Experiments have shown that the ICRF antennas with high-Z antenna guards have similar power and voltage handling capabilities to those envisioned by the ITER ICRF antennas. We also find that a properly conditioned wall, using boronization in C-Mod, is essential to attain best performance plasmas in an all-metal tokamak by lowering the impurity level. New boronization techniques, including between-shot boronization and glow-discharge boronization have been explored. The hydrogen to deuterium ratio has been better controlled in the all-metal configuration, which may partly account for a possible enhancement of ICRF absorption. Details of the experiments and results are documented in the report PSFC/RR-05-11, which is available online at:
http://www.psfc.mit.edu/library1/catalog/reports/2000/05rr/05rr011/05rr011_full.pdf.

Additional goals which were achieved in FY2005 include:

- Investigate massive gas puff disruption mitigation of high pressure plasmas
- Measure upper divertor chamber pressures with divertor baffle prototype and compare with previous results; assess implications for cryopump
- First operation of new long pulse Diagnostic Neutral Beam
- Utilize all-digital real-time control system for closed-loop feedback control of tokamak discharges; compare performance with that of the analog-digital hybrid control system
- Compare 50 MHz D(^{3}He) minority ICRF heating to 80 MHz D(H) minority ICRF heating at B~5T

- Implement all-digital fire-wire based miniature CCD cameras for improved sensitivity and higher spatial resolution imaging of in-vessel components during tokamak operation
- First C-Mod operation with prototype tungsten divertor tiles
- Measure the dependence of the threshold for error-field induced locked modes as a function of toroidal field at fixed q and Greenwald fraction; results provide important constraints on the size scaling for extrapolation to ITER

2. Alcator C-Mod Research

2.1 Boundary Physics

The C-Mod boundary physics program continues to make important contributions in a number of areas. This is in part due to the uniqueness of the plasmas studied (high density which leads to differences in collisionality), high-Z Plasma Facing Component (PFC) surfaces, and parallel power densities. The C-Mod boundary and divertor plasmas can uniquely approach those of ITER in a number of areas: parallel SOL power density; opacity of the SOL and divertor to neutrals; recombination and Lyman series absorption in the divertor. Overlaying these unique characteristics are research capabilities and emphasis that lead to a steady march forward in understanding in a number of areas – Transport (section 2.1.1), neutral physics including D retention (2.1.2), the effect of low-Z coatings on operation (2.1.2), Divertor physics (2.1.3), D retention (2.1.4), and Dust (2.1.5). In addition the boundary physics research staff continues to play a central role in machine improvements for high heat flux & particle handling (2.1.6).

Given its breadth and uniqueness the C-Mod boundary research program plays an important role in supporting general research. The specific overlap with ITPA high priority tasks, IEA-ITPA collaborative work and the priorities identified in the 2005 FESAC Panel report are covered in section 2.1.7, as well as in the general text.

2.1.1 Transport

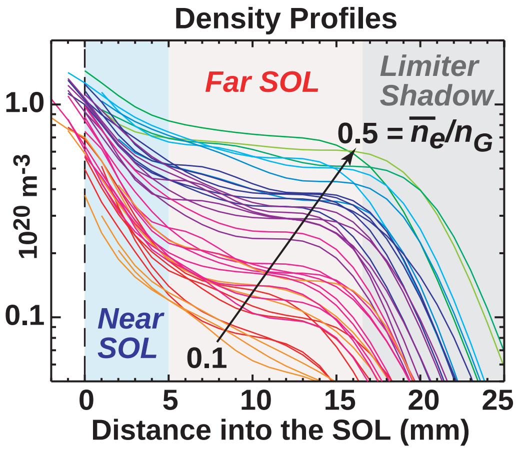


Figure 2.1.1:Density profiles in the SOL as the line-averaged density normalized to the Greenwald density is varied.

Accurate, physics-based descriptions of heat and particle transport are required in order to predict and optimize plasma-wall and plasma-divertor interaction (heat and particle loads) in ITER and power-producing reactors. This need is articulated by the ITPA high-priority research task, "Develop improved prescription of SOL perpendicular transport coefficients and boundary conditions for input to BPX modeling." For this reason, and because transport plays a crucial role in defining the conditions of the SOL and divertor plasmas, the topic of transport is a central focus of the C-Mod boundary physics program. Significant progress has been made in recent years in uncovering the underlying transport processes, moving closer towards the ultimate goal of a first-principles description of the transport physics. One key observation is that the SOL exhibits fundamentally different transport and fluctuation phenomena in the regions 'near' and 'far' from the separatrix. Figure 2.4.1 shows the typical break in the density profile at the transition between these two regions, with the density in the 'far' SOL increasing strongly with plasma density. As discussed below, this behavior may be understood as a by-product of the different transport properties in these zones:

electromagnetic affects appear to be important in the steep-gradient ‘near’ SOL region, while purely electrostatic transport mechanisms appear to dominate the ‘far’ SOL.

Our research on edge transport is organized into two complementary areas. The first is the study of time-averaged transport and plasma profiles (averaged over time scales greater than 1 ms) to characterize the level of transport, to develop empirical scaling relationships and, with guidance of theory and numerical modeling, to identify the key plasma parameters that ‘control’ the level of transport fluxes. Such a framework of understanding potentially points towards the underlying physics. Further, it allows transport comparisons to be made with other tokamak experiments and offers insight into what one should expect under reactor conditions, such as in ITER. Our second approach is to study the detailed time evolution of the underlying turbulence and how it leads to the time-averaged profiles and transport that we observe. These measurements are made with the help of state-of-the-art fast-camera and photodiode imaging systems, enabling direct comparisons with first-principles numerical simulations.

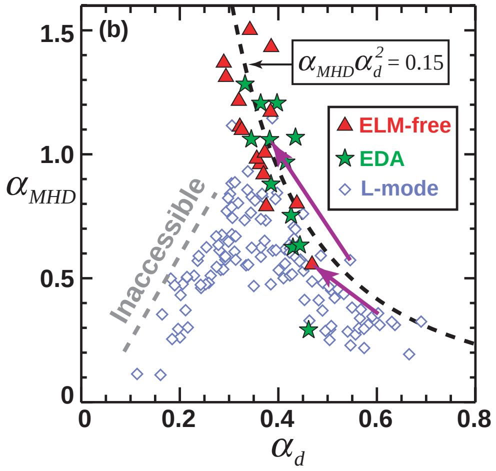


Figure 2.1.2: Normalized pressure gradient versus normalized (inverse) collisionality in the near SOL for a wide range of plasma conditions.

Time-averaged transport

Recent analysis of plasma profiles near the separatrix (i.e., the near SOL low-field-side region) has uncovered evidence that the pressure gradients there are set by electromagnetic fluid drift (EMFD) turbulence [1]. First-principles EMFD turbulence simulations [2, 3] identify 2 principal parameters that ‘control’ the level of turbulent-driven transport. These are: plasma poloidal β gradient (α_{MHD} or $\hat{\beta}$) and normalized inverse collisionality (α_d or $\hat{C}$). The edge plasma state near the separatrix in C-Mod is seen to lie within a well-defined region in this two-parameter space (α_{MHD}, α_d) over a wide range of discharge conditions – see Figure 2.4.2. These observations support the hypothesis that pressure gradients in the near SOL are set by a critical poloidal β gradient; for fixed normalized collisionality, attainable pressure gradients scale as plasma current squared. Pressure gradients in the H-mode pedestal are found to follow a nearly identical scaling [4]. Thus, the near SOL appears to form the base of the H-mode pedestal and may play a key role in the formation and development of the H-mode pedestal. We plan to explore further such important connections between the near SOL and pedestal region. We are presently in the process of mapping out the edge plasma ‘phase-space’ in more detail and over the wider parameter range that is available in C-Mod, including discharges with upper and lower single-null topologies. This research will continue to be active over the next 2 years, particularly when a new inner-wall scanning probe diagnostic begins operation at the beginning of FY 2007 (discussed further below). Figure 2.4.3 shows the B_t, I_p operational space that is presently under study.

Transport processes in the far SOL are also important since they determine the conditions under which a ‘shoulder’ forms in the profile, increasing the level of plasma interaction with main-chamber wall surfaces. Through collaborative effort, the technique used in the analysis of the time-averaged radial transport in C-Mod [5] has been applied to dimensionlessly similar discharges from DIII-D and JET and compared to C-Mod. The radial transport in the SOLs of these various tokamaks, given by effective transport coefficients, D_{eff} or v_{eff}, are found to be very similar, both in magnitude and radial scaling, indicating little or no dependence on ρ^*, ν^*, or β [6, 7]. We hypothesize that the differences in SOL profiles between JET and C-Mod are due to differences in the opacity of the SOL to neutrals [7]. Indeed, we believe that the differences in the C-Mod profiles as a function of density (see Fig. 2.4.1) are also due to differences in the opacity to neutrals. We have begun to explore lower density, reduced neutral opacity discharges (2005 campaign, Fig. 2.4.3) on C-Mod, with the aim of unfolding such effects. Analysis of the data should be performed this year. In addition, we will use data from the expanded operational space (Fig. 2.4.3) to: (1) Determine if the inferred weak dependence on dimensionless parameters holds over a much larger dataset; (2) Vary the neutral opacity over a much wider range, matching that of JET, and determine if the JET SOL profiles are achieved (without changes in transport); and (3) Compare the characteristics of the far SOL transport (v_{eff}) with that of local turbulence (e.g. phase velocities of striations/blobs). As pointed out recently by Sanchez *et al.*[8], a non-local probabilistic transport model may be a more appropriate description for transport in the SOL. We hope to explore these ideas through our continuing collaboration with Ben Carreras (ORNL). Augmenting the C-Mod work, we intend to continue inter-tokamak collaborations that include analysis of SOL data from DIII-D (H-mode), MAST and ASDEX-Upgrade. This work is part of IEA/ITPA collaboration DSOL-5 and the ITPA high-priority research task, “Improve understanding of SOL plasma interaction with the main chamber.”

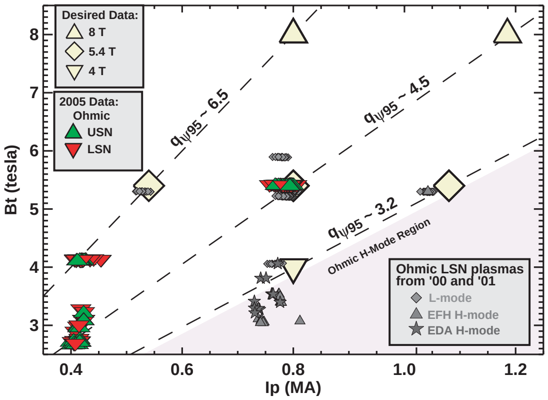


Figure 2.1.3: Proposed operational space for detailed studies of plasma transport.

Recent experiments in C-Mod indicate that strong plasma flows exist along field lines in the scrape-off layer and that these flows are driven by ballooning-like cross-field transport asymmetries[9]. Such flows may have important consequences for plasma discharges, potentially influencing the L-H threshold (via changes in magnetic topology[10]) and/or changing the flux-gradient relationships of local plasma transport in the near SOL[11]. We plan to explore this physics further in a series of dedicated experiments and in modeling efforts. One goal is to provide a measure of flow velocities independent of the Mach probe to confirm the probe-derived flows. Experiments are planned using both the Gas-Puff-Imaging (GPI) D_2 puff at the inner wall and the DNB to make CX measurements of B^{+5} ion flow velocities. A second goal is to determine how the strong parallel ion flow towards the inner divertor leg forms a ‘closed loop’ – Is there a mechanism of cross field ion convection into the closed flux surfaces or does neutral fueling from the inner divertor leg dominate? The convective channel will be investigated using impurity puffing while the neutral flow will be explored with camera imaging. Lastly, we will be utilizing 2-D fluid codes (through our collaborators) to determine if the postulated physics can provide the measured flows. The primary collaborators in this area are A. Pigarov (UCSD) and X. Bonnin (CNRS-France). Over the next year, A. Pigarov, using the UEDGE code, will be modeling flows and SOL profiles in a variety of C-Mod magnetic equilibria (upper-null, lower-null, double-null).

A number of upgrades will be made to the edge diagnostic set this year, which will help us answer these and other key physics questions in subsequent years. The current inner wall scanning probe will be replaced with one that is able to plasma measure flows and turbulence simultaneously, as a function of radius (a graduate student project). In addition, using the outer-wall scanning probe drive, we will test new ideas for advanced probe head geometries, which have the potential to improve measurement of poloidal flows and to better survive the heat fluxes that exist near the last-closed flux surface. Lastly, a new probe electronics package is being constructed that would allow fast voltage sweeps and thus direct measurement of density and temperature fluctuations (another graduate student project). Prototype tests of a working package are planned for FY 2006. Implementation on the C-Mod outer-wall scanning probe is targeted for FY 2007.

Turbulence

Plasma turbulence is responsible for the time-averaged plasma profile shapes and cross-field fluxes that are observed in the edges of tokamaks. Research on C-Mod has contributed greatly to the study of this edge turbulence, characterizing its spatial structure ("filaments" aligned with the magnetic field that appear as "blobs" when viewed along the field) and its dynamics (rapid propagation radially outward and poloidally). Over the past year we have continued to build on a number of innovative methods for diagnosing and understanding plasma turbulence in the SOL. An important tool in this arsenal is the set of "Gas-Puff-Imaging" diagnostics, utilizing fast cameras and arrays of fiber views. A radial array of fiber views was upgraded to a 2D

	C-Mod	NSTX
$B_{\text{outboard SOL}}$	4.4 T	0.2-0.3 T
$n_{e,\text{ SOL}}$	$2\text{-}20\times10^{19}$ m^{-3}	$0.2\text{-}2\times10^{19}$ m^{-3}
$T_{e,\text{ SOL}}$	10-80 eV	5-50 eV
"blob" size (radial)	0.6-1.0 cm	2-6 cm
"blob" size (poloidal)	0.7-1.5 cm	5-9 cm
"blob" velocity (radial)	<1 km/s	<1-2 km/s
"blob" velocity (poloidal)	<1.5 km/s	< 5km/s

Table 2.1.1. Comparison of SOL plasma parameters and "filament/blob" characteristics between C-Mod and NSTX.

(radial and poloidal) system and a 300-frame 250kHz ultrafast movie camera [12] was purchased and put into service. The recent focus of the research effort in this area has been characterization, scaling, and comparisons with model predictions of the filament propagation velocities [13-15]. This work will continue, with additional emphasis on *inter-machine* comparisons of the edge turbulence. An example of such a comparison between C-Mod and NSTX is illustrated in Table 2.4.1.

We are presently designing additional 2D GPI views. These views will be used for multiple purposes: for example, to examine in what way the dynamics of "filaments" change as a function of poloidal angle, to search for a region where an H-mode transition-precursor is observed, and to use these views *without* a local gas-puff to examine the plasma fueling (by measuring the Dα emission profile) as a function of poloidal angle. Coupled to one of these new views will be a new fast-framing camera that will record continuously with a maximum frame rate of ~100kHz. This camera will complement the existing 250 kHz camera so that two different views will be recorded simultaneously with high time resolution. In those cases where 100 kHz time resolution is sufficient, we will benefit from the new camera's continuous recording capability and its somewhat increased number of pixels (compared to the 250 kHz camera). Purchase of this camera is budgeted for FY07.

Future research in this area will also include comparing "filaments" in limited plasmas with simulation, examination of the parallel structure of the "filaments", comparisons of radial propagation speeds of the "filaments" and their scalings with theory/modeling predictions, in particular with modeling by Myra and D'Ippolito (Lodestar Corp.)[16] and by Naulin and Garcia (Riso – Denmark) [17].

ELMs in the SOL

The study of ELM characteristics with the goal of developing a predictive capability for their occurrence, size, and dynamics is a high-priority ITPA research task for the community. In addition, there is a crucial need to understand the effects of the ELMs on the first-wall structures and divertor. Recently, discrete ELMs have been routinely produced in C-Mod by making plasmas with high triangularity and low normalized edge collisionality, $0.2<\nu^*<1$. The structure and dynamics of these discrete ELMs are studied on C-Mod using optical diagnostics with high spatial and temporal resolution. Just after the ELM crash, plasma is observed to propagate radially outward through the SOL with velocities up to ~8 km/s, with the distribution of radial speeds of this "primary" ELM ejection peaked around 1 km/s. Qualitatively this is similar in many respects to the radial propagation of "filaments" discussed in the previous section. Continued study of the complex propagation dynamics is planned, with emphasis on the 3D structure of the ELM event. In addition, we have begun a collaboration with DIII-D to model the MHD stability of the pedestal in these ELMing plasmas and to compare the modeling results with observations.

2.1.2 Effect of low-Z coatings on operation (Boronization)

Present plans for reactor Plasma Facing Component (PFC) surface materials call for tungsten due to its low tritium (T) retention, capability to handle high heat fluxes with low erosion, and robustness to nuclear damage and activation. ITER, to provide required operational experience for DEMO, will likely at some point in its lifetime operate with all tungsten high-Z PFCs. At the moment there is minimal published divertor tokamak data detailing experience with **un-coated high-Z surfaces** thus making it difficult to ascertain whether the operational advantages of tungsten will hold true in a reactor environment. The present understanding is that current boronization techniques are not applicable to ITER. The capability of C-Mod to operate in this fashion provides timely and unique information about such requirements for ITER. A detailed description of experiments on C-Mod investigating performance with and without B coatings can be found in [18].

Boron coatings were removed prior to the FY 2005 run period to provide a unique comparison of operation with or without boronized molybdenum PFCs. Since the start of C-Mod operations, all high heat-flux PFC surfaces (divertor, inner wall and outboard limiters) have been solid Mo tiles. Boronization, applied using an Electron Cyclotron Discharge (ECDC) low temperature plasma in B_2D_6/He gas, was introduced in 1996, shortly after the first multi-megawatt ICRF heating experiments, and has been used routinely since that time to control Mo impurity influx. In 2005, it was decided to return the machine to the original all-metal PFC configuration: all boron coatings and boron-nitride tiles were removed.

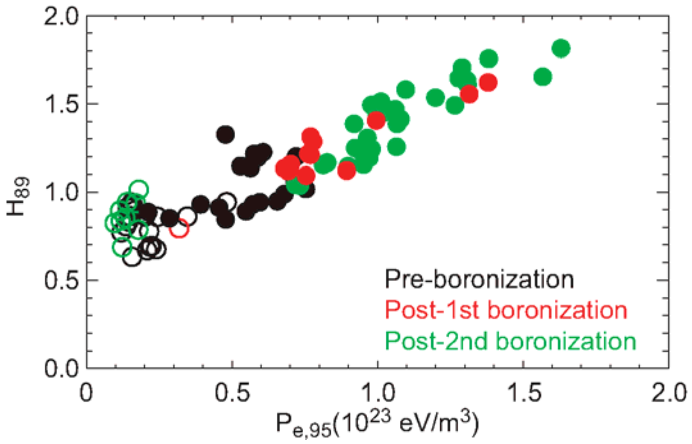


Figure 2.1.4. Energy confinement enhancement over L-Mode as a function of edge pedestal pressure.Solid (open) circles indicate H-mode (L-mode) plasmas.

After boron was removed from vessel and PFC surfaces, RF-heated H-modes were readily achieved although the resultant enhancement in energy confinement over L-mode was small. Molybdenum fractional densities approached 0.1%, rapidly rising after the H-mode transition, cooling the pedestal through line radiation, thus reducing global confinement and/or causing back H/L transitions. After boronization, the situation is dramatically changed: the Mo density was reduced by a factor of 10-100, and energy confinement doubled. Figure 2.4.4 shows the results of a set of discharges with similar plasma parameters (current, density, ICRF heating power). The energy confinement enhancement above the ITER-89P L-mode scaling is plotted as a function of pressure at the top of the edge pedestal, for three periods: pre-boronization, and after the first and second overnight boronizations of the campaign. The surface averaged boron coverage from each overnight treatment is about 200 nanometers.

The positive effects of boronization wear off, correlated with integrated input energy. Figure 2.4.5 shows this effect for a set of constant plasma current, density and field shots.

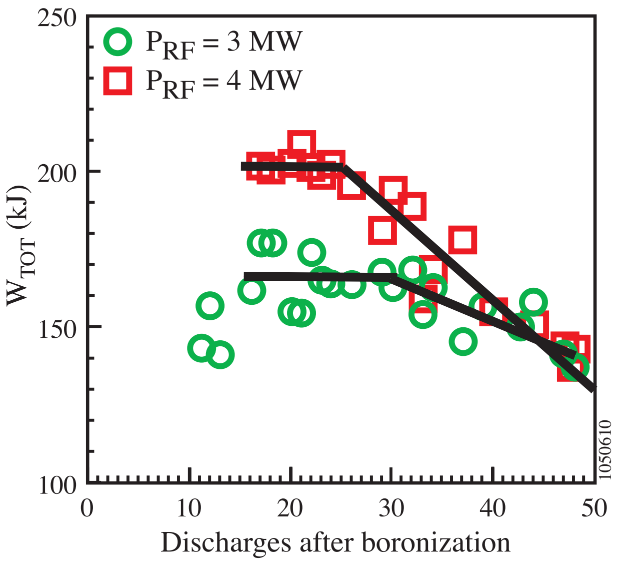


Figure 2.1.5: Degradation in stored energy as discharges are repeated. 25-30 discharges corresponds to ~ 50 MJ of input energy.

The ICRF power was stepped up during the discharge allowing the stored energy to be measured. For constant input power the stored energy is directly proportional to the energy confinement. The degradation in energy confinement occurs after an integrated input energy of ~50 MJ. Initial experiments have been started to assess whether the boron erosion is linked specifically to the RF power or more generally to total input power. Comparisons of ICRF H-modes with Ohmic H-modes point to RF-specific effects accelerating the erosion process. Because of these results we are planning further Ohmic - RF boron erosion comparison discharges. A reproducible target discharge will be produced via between-shot boronization and subsequent discharges will either have RF-heated or Ohmic heating only. The measure of the erosion rate is the number of discharges the thin layer of boronization lasts.

The detrimental erosion of boronization is clearly very localized and this has engendered efforts to identify those locations. Prior to the 2005 campaign the PFC surfaces had not been cleaned for a number of campaigns. Boron had accumulated over almost all PFC surfaces to thicknesses of order 6 microns, ~30x that of a single overnight boronization. However, in a few areas, the boron coating thickness was very small or non-existent. These locations include the outer divertor near the strike point, leading edges of each of the 10 outer divertor modules, outer limiter surfaces and gusset protection tiles at the top of the machine. Previous experiments showed that the outer divertor strike point, although a significant local source of Mo, was not affecting the core Mo levels in most cases. In an effort to better localize the primary B erosion location leading to enhanced core Mo levels (and make more reproducible discharge conditions) inter-shot boronization techniques were developed. The ECDC plasma used for boronization is formed at the cyclotron resonance, which is a cylinder at fixed major radius (f=2.45 GHz, B=.088 T). We thus control the resonance location by varying the toroidal field coil current during the deposition. In a series of inter-shot boronizations (~3 nanometer coverage each), the resonance location was scanned in steps, and the efficacy in each subsequent discharge was evaluated. The location near R=0.7 m appears to be the most effective, as judged using the radiated power during the high power ICRF phase just prior to the transition into H-mode. However, some uncertainty about the B erosion location remains because the ECDC plasma is not well confined, and we don’t expect the boronization layer deposition to be limited to the resonance radius.

Several strategies are being pursued to better identify the important B erosion locations. First, since early in the 2006 run campaign, new spectroscopic views of the outer divertor cover plate and outer limiter have been introduced. These allow us to monitor the B and

Mo source rates from these locations and correlate changes with RF power and boronization resonance location. Secondly, during the summer 2006 vacuum break, we plan to install commercially available quartz micro-balance (QMB) detectors. The QMBs will be movable (rotationally, about and along a major radius) to measure the boronization deposition profile for a given ECDC resonance location – for horizontal as well as vertical surfaces facing toroidal and radial directions. A third strategy is being investigated that involves changing certain tiles' surface composition. During the same summer 2006 vacuum break some of the tiles suspected as being primary sources of Mo affecting the core plasma will be coated with elements that, if introduced into the core plasma, will be easily observed spectroscopically and thus their absolute densities determined. Different types of tiles (e.g. gusset vs limiter, vs outer divertor leading edge) will be coated with different elements (thickness several microns). Thus, after a boronization, if a particular element appears in the core plasma we will know where it came from.

As part of our research into operational strategies for ITER we are also investigating *during-discharge* boronization. We have tested our diborane system and determined it can be used to fuel a discharge. We plan to inject varying amounts of diborane into Ohmic and RF-heated H-modes and determine if such a process slows or reverses the net B erosion.

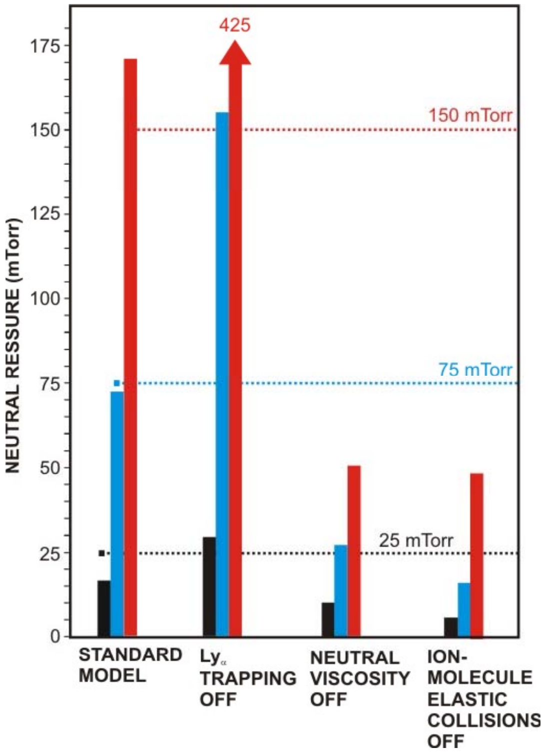


Figure 2.1.6: Turning processes off in EIRENE one at a time, to see what is important (static plasma solution) for the three neutral pressure cases.

2.1.3 Divertor physics

The codes currently being used to model the ITER divertor plasma are benchmarked against results from tokamaks running in low density regimes, where the neutral mean free path is long compared to the divertor and divertor plasma dimensions. Such comparisons with experiment fail to test 2 important aspects of physics in the models: 1) Lyman–α radiation trapping – which potentially strongly affects the ionization balance and ability to achieve detachment (required in ITER to dissipate the power flow to the divertor); and 2) the diffusive, or short neutral mean free path, regime – which determines the neutral densities in the divertor and thus pumping rates for D, T and He ash. Conditions in the C-Mod divertor plasma can address both of these key issues. A set of data for 3 C-Mod shots has now been identified and partial modelling utilizing the OSM-EIRENE code combination, in collaboration with the U. Toronto group (S. Lisgo) and U. Bochum (D. Reiter), is well underway. Preliminary results are shown in Figure 2.4.6. The three C-Mod discharges correspond to the outer divertor being attached (P_{DIV} = 25 mTorr), detached (75 mTorr) and detachment reaching the x-point (150 mTorr). The analysis of the 25 mTorr case is already done [19]. This work is part of the IEA/ITPA DSOL-5 collaboration with JET and has furthermore been specified by the ITER team as one of 5 calendar year 2006 ITER-US sub-tasks for US research to provide answers for ITER.

These C-Mod divertor conditions provide very good tests of the key physics. In the detached cases the photon absorption mean free path is ~ 1mm, smaller than the plasma fan, as will be the case in ITER. The high neutral densities will also provide tests of n-n and D^+ - D_2 collision physics for mean free paths small compared to the relevant plasma fan or divertor dimension (λ_{nn} ~ 1.3-7.8 mm, $\lambda_{p\text{-}D2}$ ~ 1-8 mm for the ranges in cases under study). As seen in Figure 2.4.6, photon and neutral collision processes play important roles in neutral transport and in determining the pressure in the divertor.

Another way of showing the importance of recombination is to compare it to the other ion sink in the divertor—ion current to the divertor plates:

Case	EIRENE recombinations	Ion flux to targets below the nose	recombinations/ion flux
25 mTorr	1.76×10^{22}/s	9.1×10^{21}/s	1.93
75 mTorr	2.32×10^{22}/s	6.03×10^{21}/s	3.7
150 mTorr	4.38×10^{22}/s	8.5×10^{21}/s	5.15

Furthermore, as the density is increased the effect of Lyman trapping on the solution (for pressure) becomes stronger: The ratio of 'no absorption' to 'absorption' for the three cases is 1.9, 2.2 and 2.5 respectively.

Although these results are a significant step towards proper testing of the physics in codes, more is required and planned. The results shown above for the 2 higher pressure cases are very new and will be studied and improved over the next few months. In addition to varying the specified densities and temperatures over the uncertainty range allowed, the full 3D divertor structure should be included because leakage, toroidally and poloidally, leads to factors of 2 reductions in the divertor pressure for the low pressure case. The next logical step for testing the code physics is to move from interpretive modelling, as done now, to predictive modelling, using the B2-EIRENE combination. This will provide a self-consistent solution, a more stringent test, and will be more usable for determining the importance of certain processes in the divertor solution. This work is being discussed currently as part of an expansion of the work by the U. Bochum group involving D. Reiter and Kotov.

2.1.4 D retention

Controlling the hydrogenic fuel inventory in Plasma Facing Component (PFC) materials will be necessary for the successful operation of burning plasmas that use tritium (T) as a fuel. In ITER, the in-vessel limit of tritium is 350 g for safety reasons, while ~100 g is fuelled into the vessel for a full power discharge, indicating the requirement for low global retention rates. As mentioned earlier, tungsten is currently projected for use in ITER and reactors partly due to its very low, compared to C, T retention [tritons per tungsten atom (T/W) ~ 10^{-5}] as measured in laboratory experiments. However, tokamak experience providing support for this result is lacking. We have undertaken to investigate high-Z PFC hydrogenic (H/D) retention and methods of controlling H/D wall inventory in a high-Z divertor tokamak.

Initial measurements of the D_2 retention in C-Mod discharges shows the surprising result that a large fraction of injected gas is retained in PFC surfaces, implying D/Mo ~ 10^{-3} - 10^{-2}, much higher than expected. The gas balance is accurately determined by closing all valves to pumps for the duration of the shot (and long afterwards) while measuring the quantity of gas injected and pressure in the chamber. An example of the large retention is shown in Fig. 2.4.7 for a set of identical 1 MA discharges where the Mo PFCs are not coated with B (boron coatings removed). D retention is clearly much higher than expected from laboratory results, at least in a global sense. The equivalent level of retention is also obtained when surfaces are coated with ~150 nm layer of boron during boronizations, implying D/B co-deposition is not an important process. While retention rates may evolve and depend somewhat on plasma parameters, the wall inventory does not saturate; large retention fractions are found for repeated discharges (Fig. 2.4.7). The retention is also dependent on plasma density, divertor strike point position and discharge duration; retention is likely localized in regions of high plasma flux and/or fluence.

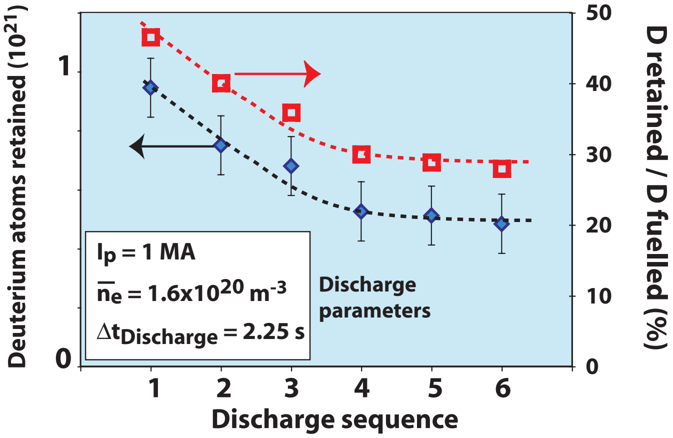


Figure 2.1.7: D retained by PFC surfaces per discharge for a sequence of discharges.

It is clear that further experiments are needed to investigate these phenomena as part of the general C-Mod emphasis on high-Z PFCs. Further constant discharge measurements of the D retention will be undertaken to determine if the behavior exemplified by Fig. 2.4.7 is repeated all densities. If flux and fluence are important variables, then strike point position and pulse length should be important, so further variation of these parameters is needed. Finally, in collaboration with groups at U. Wisconsin (D. Whyte) and UCSD (R. Doerner), we are attempting to create a laboratory environment (in DIONISOS at UW and in PISCES at UCSD) as closely approximating C-Mod conditions as possible in order to revisit, under more controlled conditions, the physics of D retention.

It is highly desirable to develop in-situ methods of controlling the recovery and retention of hydrogenic fuel species, in order to minimize delays in plasma operations in devices like ITER that might be required to remove accumulating T. For instance, an in-situ tritium recovery method has been proposed for ITER that uses planned radiative terminations [20]. Radiative disruptions cannot deliver sufficient energy density to the wall in present devices, due to their small energy density as compared to ITER. Therefore, planned disruptions aimed at depositing the plasma energy onto a fraction of the PFC surfaces have been used instead to provide the required surface heating. Fig. 2.4.8 shows the H recovered from PFC surfaces as a function of the energy density (which scales like the electron temperature) that can be delivered to those surfaces for a day early in the FY2005 run period. We see that there is a threshold energy above which the H removed becomes significant, in fact surpassing typical retention values (Fig. 2.4.7). When the experiments of Fig. 2.4.8 were repeated at a later point in the campaign, when the H levels in PFCs was lower, the threshold energy was increased indicating that a deeper region needed to be heated to promote H diffusion and removal. Fig. 2.4.8 is consistent with expectations that a critical surface temperature must be obtained to promote H diffusion and release on a ms timescale. Changing the disruption type and pre-disruption plasma shape varied the location of heating and hence removal efficiency. The amount of H removed during this day is ~ 1/3 of the H inventory measured to be absorbed into PFC surfaces. The large, post-vacuum-vent level of H in tiles is due to the vacuum vent with concomitant exposure of PFCs to H_2O laden air. Subsequently, all discharges and discharge-cleaning is performed with D_2. This demonstrates that planned disruptions can result in net global fuel *depletion* (i.e. the opposite of retention) of the wall, an important demonstration of H isotope control in a confinement experiment. The rate of H/D removal demonstrated in Fig. 2.4.8 has been determined to give a 5-10x higher global H/D recovery rate than either EC discharge cleaning or non-disruptive discharges.

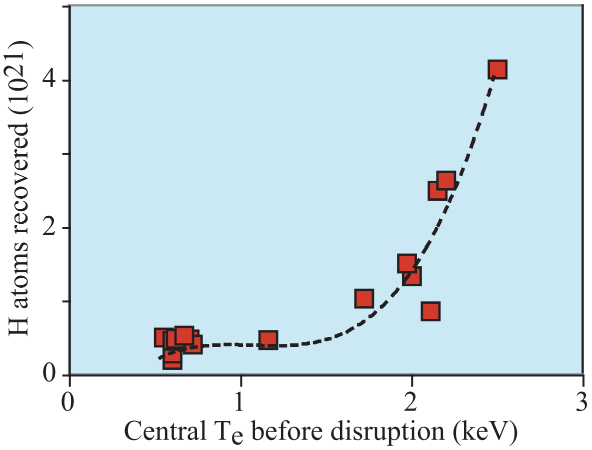


Figure 2.1.8: H removal as a function of core plasma Te (proportional to stored energy)

2.1.5 Dust

Dust in a reactor vessel may adversely affect both facility safety and plasma performance. As such, dust characterization and measurement techniques are high-priority ITPA research tasks. The C-Mod program contributes to the study of dust both by image analysis of dust as seen by visible and IR cameras and by characterization of the dust removed from the vessel after openings. We will continue development and analysis of our image-based measurements, as well as implement laser-scattering measurements using the Thomson scattering lasers. We also plan to investigate the origin of dust in C-Mod and how it affects the core plasma.

2.1.6 High heat flux & particle handling

The performance of C-Mod plasmas is intimately connected to performance of the divertor and first-wall components. Consequently, the boundary physics and operations groups work closely together in developing and optimizing these components. Over the past year, we have continued the development and testing of ITER relevant tungsten tile modules for C-Mod. Twelve 'tungsten brush' tile modules (based on a concept developed for FIRE by the Sandia group) were installed in the divertor at leading-edge locations. These tiles were exposed during the entire 2005 run campaign. No measurable tungsten content was seen in the core and the tiles survived without melting. However, a small number of the tungsten rods did fail mechanically; this has been traced to the method of fastening them to the base plate. We have recently changed to an improved lamella design (Fig. 2.4.9). These tiles are simpler and cheaper to manufacture, and are similar to the current concept being considered by the ITER team. Prototypes of these tiles have been tested for heat flux handling at Sandia and Julich, identifying further design refinements (e.g., use of molybdenum bolts and changes in fastener geometry). A full toroidal ring of 120 tiles will be installed near the outer divertor strike point during the summer vacuum break in 2006.

The plan for the next campaign is to monitor the tungsten lamella tiles with the divertor IR imaging system, particularly toward the end of the campaign, when the tiles will be deliberately exposed to extended high power fluxes. The levels of tungsten in the core plasma and the mechanical integrity of the tiles will be tested as well.

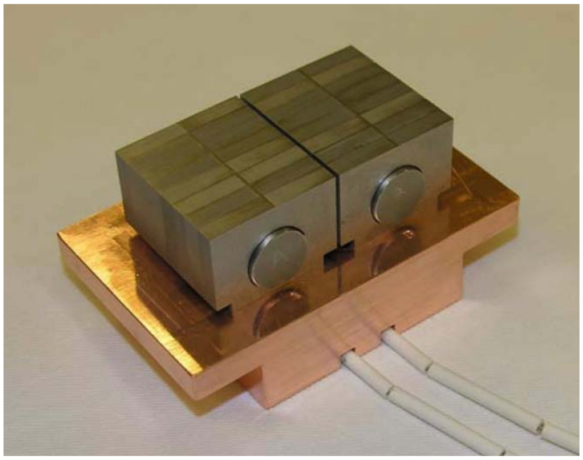
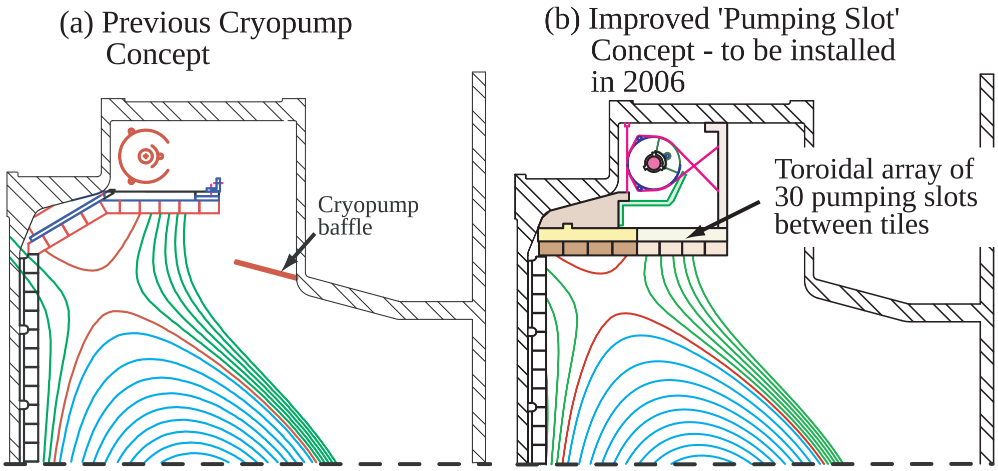


Figure 2.1.9: Prototype C-Mod tungsten lamella tile modules mounted on a test fixture for high heat flux testing.

Another area where the boundary physics group supports the C-Mod program is in cryopump development. The cryopump is an important tool for reducing the core plasma density and thus allow more efficient current drive. In addition lower density H-modes may provide access to different ELM regimes.

A series of experiments were performed in 2005 to investigate the optimal placement of the cryopump gas inlet and baffling systems. These experiments proved to be very valuable. They indicated that the original cryopump design (Fig 2.4.10a), with gas flowing around the upper divertor plate, would experience a sensitivity to the upper divertor strike-point location (factor of ~3 change in pressure). In contrast, data indicated that a design based on a regular array of 30 pumping slots (Fig. 2.1.10b) would be insensitive to strike point location and provide a higher gas throughput to the cryopump surfaces. A closer proximity of the tile surfaces to the upper null also improves heat-flux handling via increased flux expansion. A full set of molybdenum tiles and support hardware for the 'pumping slot' cryopump system is presently in house. Fabrication and testing of the cryogenic system is presently in progress with installation planned for the summer vacuum break in 2006.


(a) Previous Cryopump Concept

(b) Improved 'Pumping Slot' Concept - to be installed in 2006

Figure 2.1.10: Experiments in 2005 showed that the previous cryopump concept (a) would have a sensitivity to upper divertor strike-point location that could be avoided using a 'pumping slot' design (b). The latter concept will be used for the cryopump to be installed in 2006.

2.1.7 Relationship of the C-Mod boundary program to ITPA, FESAC priorities

The C-Mod boundary physics program addresses a number of **high priority ITPA issues** including:

- *Improve understanding of tritium retention & the processes that determine it.*
    - Understanding D levels on tiles and tile sides for B and Mo
    - Understanding removal of D at low tile temperature
- *Improve understanding of SOL plasma interaction with the main chamber.*
    - Wall flux measurements ('main chamber recycling')

  - Impurity influx and screening studies
- *Develop improved prescription of SOL perpendicular transport coefficients and boundary conditions for input to BPX modelling.*
  - Radial flux analysis - transport coefficients
  - Gradient scaling work (near SOL) connection to fluid turbulence theories
  - Dimensionless comparisons and scalings of SOL characteristics across tokamaks
  - SOL flows and effect on core
- *Understand the effect of ELMs/disruptions on divertor and first wall structures*
- *Development of measurement requirements for dust*

The C-Mod group is involved in a number of **IEA-ITPA collaborations:**

- DSOL-3 'Study of radial transport', B. Lipschultz organizer
- DSOL-4 'Comparison of disruption energy balance and heat flux profile', D. Whyte, J. Terry, contributor
- DSOL-5 'Role of Lyman absorption in the divertor', J. Terry, contributor
- DSOL-11 'Disruption mitigation experiments', R. Granetz contributor
- DSOL-13 'Deuterium co-deposition in gaps of plasma facing components B. Lipschultz participant
- DSOL-15 'Inter-machine comparison of blob characteristics' J. Terry organizer

The C-Mod Boundary physics researchers are leading one of the CY2006 ITER tasks requested of the US – subtask #3: '**'Effects of radiation transfer on divertor plasma'**

Lastly there is clear support of the FESAC priorities –
Topical questions:

- **T10**: "*How can we interface a 100 million degree burning plasma to its room temperature surroundings*"
  - C-Mod addresses this at the highest parallel power levels with Mo (& W) surfaces
  - Studies of transport in the SOL (parallel and perpendicular to magnetic field)
  - Plasma-materials interactions (Mo, W, boron coatings)
  - Materials development (e.g. W tiles) and operational experience

"Selected High Priority Activities":

- *"Carry out additional science & technology activities supporting ITER including diagnostic development, predictive modeling and enabling technologies"*
  - Predictive modeling - collaborative work with modelers to provide radiation transport and divertor plasma data at ITER conditions.
  - Enabling technologies - development of ITER-like bulk W tiles.
- *"Resolve the key plasma-material interactions, which govern material selection and tritium retention for high-power fusion experiments"*
  - D retention and removal studies in a high-Z environment
  - Compatibility with core (melting, impurities)
- *"Expand the effort to understand the transport of particles and momentum"*
  - C-Mod a world-leader in the study of cross-field transport of particles

The source of parallel flows, their transport and effect on core confinement are central studies

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2.2 **Transport**

Recent Highlights

Transport research on C-Mod emphasizes areas important for plasma science and of strong relevance to ITER. Imbedded in a national and international transport program, we concentrate on those issues where we have unique capabilities, where we run in unique parameter regimes and where we observe unique or unusual phenomena. Comparisons with theory and modeling form a critical part of the program, motivating and influencing the design of most experiments in the transport area. We have established close collaborations with theory and modeling groups at MIT and elsewhere. Topics of recent interest include rotation and momentum transport in plasmas with no external torque; internal barrier physics with equilibrated ions and electrons and no core particle or rotation sources, edge barriers, including threshold, confinement, pedestal scaling and mechanisms which control the pedestal such as ELMs and the quasi-coherent mode. There are obvious close connections with the integrated scenario thrusts, but also strong coupling to the boundary, stability and wave-particle topics. A good deal of the most interesting physics occurs at the interfaces between topical areas.

The study of edge transport barriers has continued, with specific focus in the areas of barrier formation, profile structure and edge relaxation mechanisms. Considerable emphasis has been placed on millimeter-resolution measurements that can resolve the small scale lengths of C-Mod pedestal profiles (*e.g.* edge Thomson scattering, scanning probes and neutral emissivity imaging), and on fluctuation diagnostics for characterizing edge turbulence (reflectometry, PCI). Recent efforts have been directed at diagnosing ion temperature and rotation, in particular with charge-exchange recombination spectra. Figure 2.2.1 shows an example of the ion temperature profile obtained. Promising results have been obtained near the top of the pedestal and inward, as discussed below. High-field-side measurements of ion temperature within the pedestal region have also been successful, showing good agreement with simultaneous electron temperature measurements. Improved rotation measurements, desirable for studies of L-H threshold physics and turbulence stabilization in the pedestal, are discussed below.

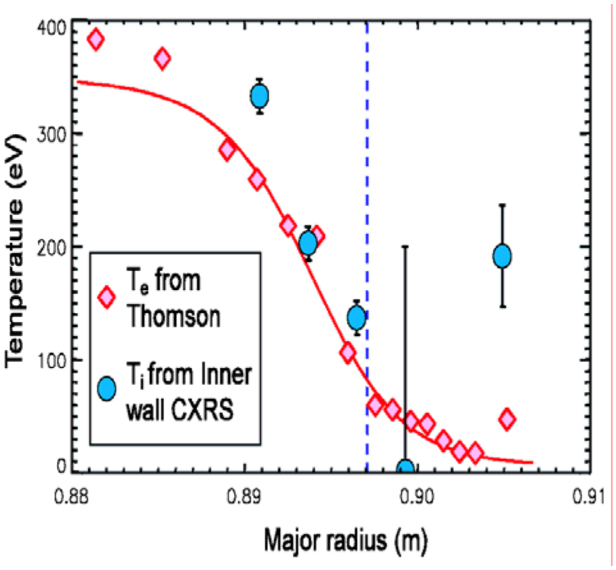

Fig. 2.2.1 High-resolution ion temperature profiles are now available for the pedestal region.

The physics determining pedestal structure has been explored through experiment and modeling. Experimentally, the density pedestal width varies little, while the height is set primarily by the value of driven plasma current and varies weakly with available neutral source. Indeed, a high degree of density pedestal resilience is observed even when aggressive gas puffing is applied to H-mode plasmas. The pedestal fueling process has been studied by coupling a one-dimensional kinetic treatment for neutrals to a diffusive model for plasma transport, allowing computation of the response of the density pedestal to perturbations in the neutral source. For typical C-Mod input profiles, modeling results agree qualitatively with the aforementioned empirical scalings, and provide a more detailed understanding of the neutral screening process that occurs in the high-density pedestal. The controlling physics responsible for setting pedestal plasma transport also has been examined, with consideration of whether a diffusive model is appropriate. The ballooning-like scaling $\nabla p \propto I_P^2$, observed empirically in H-modes with and without ELMs, suggests a role for gradient-driven transport governing the H-mode pedestal, in a manner similar to that observed in the near SOL of ohmic discharges (see fig 2.2.2).

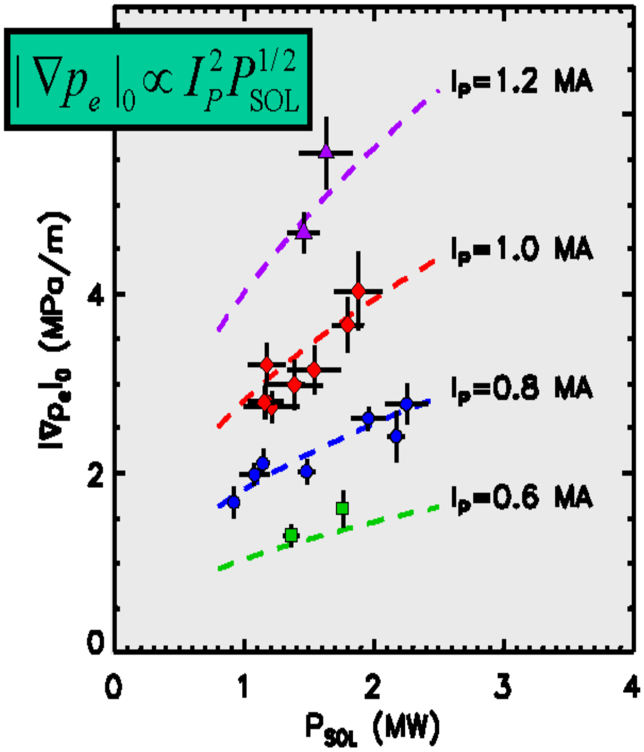

Fig. 2.2.2 Scaling of the pedestal pressure profile.

Exploration into the mechanisms that regulate edge transport, namely ELMs and the quasi-coherent mode (QCM), has moved forward. Properties of the QCM, which is responsible for the enhanced D-alpha (EDA) H-mode, have been studied using numerous fast diagnostics, resulting in clear harmonics, as well as coupling to higher-frequency core modes. Recently a joint experiment compared EDA H-mode to the similar HRS regime on JFT-2M, using matched poloidal cross-sections and scans of safety factor and collisionality. The region in this phase-space giving rise to continuous transport-driving edge fluctuations was similar across the two machines, increasing confidence that these regimes were produced by the same transport mechanism. An unexpected consequence of this experiment was the appearance of discrete ELMs, larger than normally observed on C-Mod. The shape and collisionality requirements for obtaining this ELM regime have been roughly determined, and characterization of ELM propagation has been performed using fast D-alpha diodes in the plasma edge.

Work on self generated rotation in torque-free plasmas has continued in several areas. New diagnostics have been commissioned to improve the radial coverage of the rotation profile measurements. Currently, most data come from, HIREX, an array of high-resolution x-ray spectrometers. These make measurements of ion temperature and velocity from the Doppler broadening and shift of x-ray transitions from argon and neon which are injected into each discharge in trace amounts. A new instrument, NeSoX makes analogous measurements at lower temperature (and thus farther out in the profile) using injected neon. This diagnostic has been tested and has shown good spatial and spectral resolution. Further measurements are now being obtained from charge-exchange recombination spectra in visible light. With the final commissioning of the DNB for long-pulse operation this should allow routine measurements of ion temperature in the pedestal and outer regions of the core plasma. In previous work, C-Mod published a scaling with toroidal velocity proportional to W_P/I_P or $<P>/I_P$, where W_P is the plasma stored energy and <P> is the average plasma pressure. Subsequent experiments on other devices, Tore-Supra and DIII-D for example, found a similar scaling. An attempt is underway, under the auspices of the ITPA, to assemble a database from several machines and to calculate an empirical scaling law. While that work has just begun, recent activity has focused on the identification of important physics variables. One result can be seen in fig 2.2.3 where the toroidal Mach number is plotted against, β_N the normalized plasma pressure.

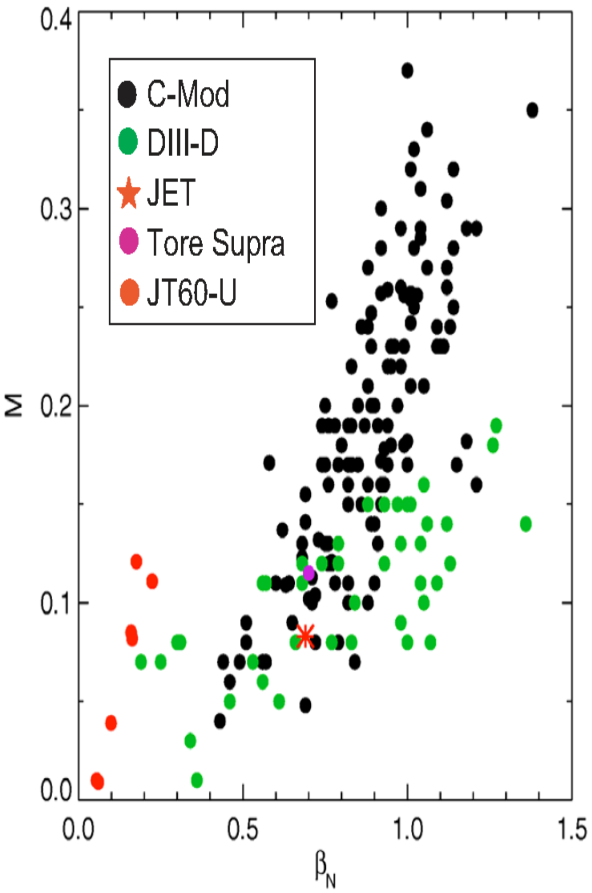

Fig 2.2.3 Dimensionless scaling of self-generated rotation for multiple machines has begun.

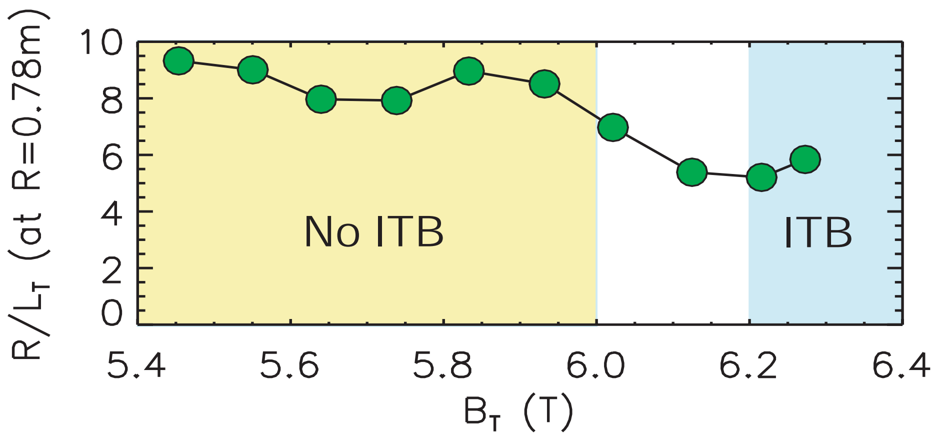

Fig. 2.2.4 Normalized temperature gradient is plotted vs toroidal field. The change in this parameter as the ICRF resonance is moved off-axis is consistent with our theoretical picture of barrier creation via ITG suppression.

Studies of internal transport barriers have focused on formation and control issues. In C-Mod, barriers are routinely

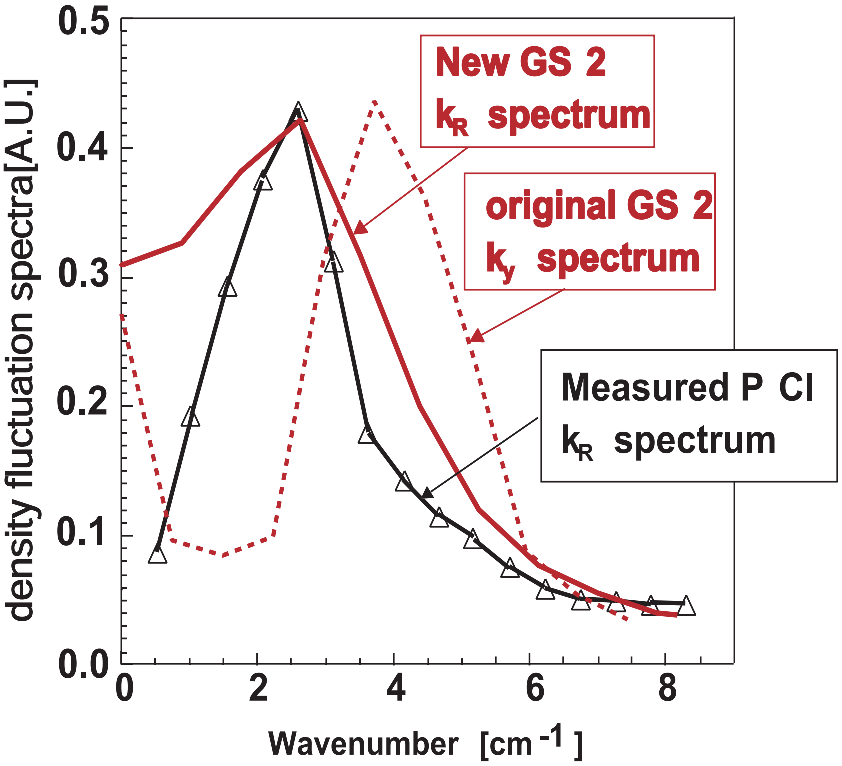

Figure 2.2.5 Using a new post-processor to produce a synthetic PCI diagnostic from the output of the gs2 code, qualitative agreement with experimental measurements was obtained. This result supported the conclusion that TEM turbulence is responsible for the regulation of particle transport in the C-Mod ITB.

produced by application of off-axis ICRF heating. These discharges have steep density profiles and sharply reduced thermal and particle diffusion over the inner half of the plasma. An early hypothesis, that the barriers were formed due to reduction of the core temperature gradient was supported by gyrokinetic simulations. This model had, however, some difficulty explaining the extreme sensitivity to RF resonance location. A change in the magnetic field by only a few percent could determine whether a barrier formed or not. A recent set of experiments making careful measurements of the temperature profile over the relevant range in fields, was carried out to investigate this issue. A result is shown in figure 2.2.4 where the normalized gradient R/L_T is plotted vs field. A significant and systematic change in the gradient is seen to occur near the barrier threshold field. Control of barrier strength has been explained by the excitation of trapped electron modes (TEM) which occurs as the density profile steepens. A sophisticated synthetic diagnostic using the output of the gs2 gyrokinetic code has been built to replicate all details of the actual PCI measurement. With this calculation, comparisons between gs2 and the experiment, which were previously qualitative, have now come into quantitative agreement (see Fig 2.2.5). Measurements of the same fluctuations with a heterodyne ECE spectrometer, operating in a refractive regime, confirm that these fluctuations are inside the barrier. Barrier position control has been explored via modification of the q profile. Systematic scans of toroidal field and plasma current indicate that q and/or magnetic shear play an important role in setting the barrier foot position.

Enhanced core fluctuation diagnostics are opening up new windows for core transport studies. The PCI (phase contrast imaging) diagnostic has been upgraded to allow some degree of spatial localization to what would otherwise be a line integral measurement.

An example is shown in figure2.2.6 where broadband fluctuations, localized to the core, are seen propagating in the ion diamagnetic direction while the quasi-coherent (QC) mode is localized in the edge and seen to propagate in the electron direction. A large part of the core mode rotation is due to Doppler shift from toroidal rotation. PCI has excellent sensitivity, high signal-to-noise and frequency response up to 5 MHz. Its *k* range has been expanded up to 25 cm^{-1}, with plans to go to 60 cm^{-1} in the near future. With these capabilities a search has begun for the fluctuations responsible for anomalous electron thermal transport. The first experiments performed looked at the low-density linear confinement regime for Ohmic plasmas. This regime has the virtue of decoupled electrons and ions with all heating and power loss through the electrons. While some changes in the fluctuation level were observed, the overall *k* spectra did not vary. Gyrokinetic analysis of these discharges is underway.

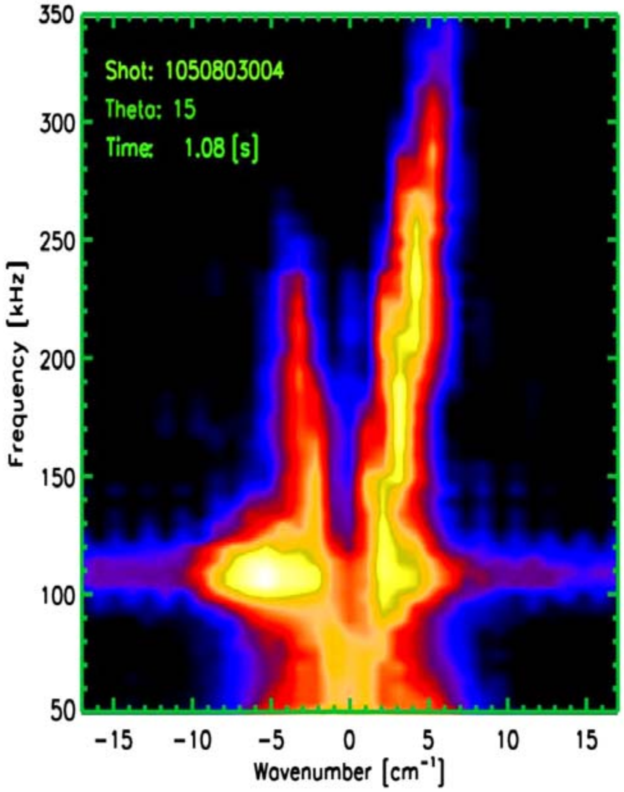

Fig 2.2.6 Enhancements to the PCI diagnostic allow spatial localization of the measured fluctuations. In this case, broadband fluctuations are localized in the plasma core and observed to propagate in the ion diamagnetic direction, while the QC mode is seen in the edge, propagating in the electron direction.

Research Plans

Transport research over the next two years will focus on areas motivated by emerging theoretical issues and enabled by new diagnostic and facility capabilities. As prediction and control are the ultimate goals of transport studies, validation of turbulence codes is an emerging theme in the transport community. Experiments and theory have progressed to the point where meaningful quantitative tests are now being made and theory plays a critical role in motivation, design and analysis of most experiments carried out on C-Mod. Facility upgrades will play a direct role in experiments targeted at the “standard” model for ion energy transport – that is ion-gyroscale drift wave turbulence regulated by zonal flows. LHCD will allow direct modification of the magnetic shear profiles while the planned cryopump should lead to high-performance plasmas at significantly reduced collisionality. Both parameters are predicted to have important effects on the marginally stable temperature profile. Studies of other transport channels will include further experiments into momentum transport and self-generated rotation; particle and impurity transport and investigations into the nature of anomalous electron energy transport. The manipulation of magnetic shear should open up avenues for creation and manipulation of internal transport barriers in steady-state plasmas. Finally, a substantial amount of experimental time will be devoted to edge barriers including investigations of the L/H threshold and its connection to equilibrium and fluctuating flows; edge barrier relaxation mechanisms especially small ELM and EDA regimes; pedestal scaling; momentum transport in the pedestal; transition and bifurcation dynamics and a search for evidence of turbulence spreading.

In the next two years, increased emphasis will be placed on plasma rotation in the pedestal, bringing to bear newly available measurements to answer fundamental questions regarding self-generated rotation. Rotation profiles with higher radial resolution will provide a comparison of measured poloidal rotation and that predicted by neoclassical theory, both with and without toroidal field reversal, and with grad-B drifts pointing toward and away from the plasma x-point. Time-dependent core rotation profiles following L-H transitions suggest inward transport of momentum from an edge source, and steady state core rotation are well correlated with SOL flows, demonstrating a strong coupling across the edge region. This will be studied in greater detail, with the goal of characterizing momentum transport through the pedestal region and determining its likely mechanisms, as well as the consequent modification to the edge shear layer. SOL flows are seen to have a significant impact on the H-mode power threshold, which is seen most dramatically in the threshold's topology dependence. It has been shown that the power threshold and flows are extremely sensitive to the magnetic balance of the discharge when near double-null geometry, and this will be further examined. Such experimental work should provide input for potential modifications to L-H transition and bifurcation theory.

The structure and scalings of the edge pedestal, examined over the typical operating range of C-Mod ($4.5<B_T(T)<6.0$, $0.6<I_P(MA)<1.0$), will be explored further, as work is in progress to expand greatly the range in operational parameters over which H-modes are obtained. Additional emphasis will be placed on pedestal scalings in regimes other than pure EDA, the H-mode regime on which prior work concentrated. Considerable increases in the ELM-free data set will be analyzed, and the structure of these profiles compared with neoclassical predictions from the XGC code that indicate a leading dependence of $1/B_T$ for the pedestal width. Pedestal structure evolution in ELM-free periods on C-Mod will also be studied and compared with the inter-ELM evolution on DIII-D. A systematic comparison of pedestals with favorable and unfavorable ion ∇B drift will also be done. Progress in modeling the neutral fueling of the density pedestal will be followed up with further analysis of the kinetic neutral results, comparisons to fluid modeling results, and incorporation of a critical-gradient assumption for the plasma transport. There will be continued examination of the critical-gradient hypothesis for H-mode pedestal structure, and an attempt to discern the physical connection between the observed H-mode pressure gradient scaling and that seen in the near-SOL of Ohmic plasmas. Dimensionless inter-machine comparisons of pedestals will be carried out, to assess the importance of factors, such as neutral physics, not considered in the standard Connor-Taylor formulation for scaling invariance. C-Mod plans to take part in such a comparison along with three other tokamaks, running in the coming year at 8T to obtain a dimensionless match to JET. An additional question to be considered is what mechanisms are responsible for the region of intermediate temperature gradient observed between the narrow pedestal region and the main core plasma. This region is observed to be 1-2 cm in extent, or roughly 20 poloidal ion gyroradii, and could be evidence of turbulence spreading analyzed by Hahm and Diamond.

Expansion of operational phase-space is useful not only in refining pedestal scalings and L-H transition thresholds, but also for exploring regimes of H-mode edge transport. Experiments will emphasize benign ELMy regimes on C-Mod and work toward an improved mapping of their existence criteria in both dimensional and non-dimensional edge parameters. Much of this work will be assisted by the addition of the C-Mod cryopump, allowing creation of plasmas with lower collisionality. Stability analysis with the ELITE code will proceed on both large and small ELMy regimes, in an effort toward physics-based characterization within the context of linear peeling-ballooning analysis. Further studies of precursors and ELM propagation dynamics will proceed, as will an experiment attempting to match the small ELMy regime of NSTX. The QCM in EDA H mode will also be studied further, as ongoing work attempts to uncover more information on the mode's radial structure and harmonics. Through comparisons of results with predictions of Kelvin-Helmholtz and analysis with the BOUT code, a more definite identification of the mode (resistive ballooning, K-H or other) will be sought, along with answers to persistent questions about the mode. In particular, the ability to drive strong particle transport while maintaining good energy confinement, the mode's saturation mechanism, and the coupling to core plasma modes will all be addressed

We plan to exploit the newly installed lower-hybrid current drive system in experiments aimed at exploring the role of magnetic shear, $\hat{S}$, in turbulent transport. There are theoretical predictions, backed up by gyrokinetic simulations, that very low or reversed magnetic shear can significantly reduce the drive for important micro-instabilities. Because of the relatively large ratio of discharge time to current penetration time on C-Mod, it is not possible to maintain modified shear configurations created in current-rise experiments. However, with LHCD it should be possible to vary this important parameter in a steady-state manner. Through small scans of the toroidal field, accurate measurements of the local temperature gradient length can be made, without recourse to absolute calibrations, and compared directly to the predictions of nonlinear gyrokinetic simulations made with the gs2 and gyro

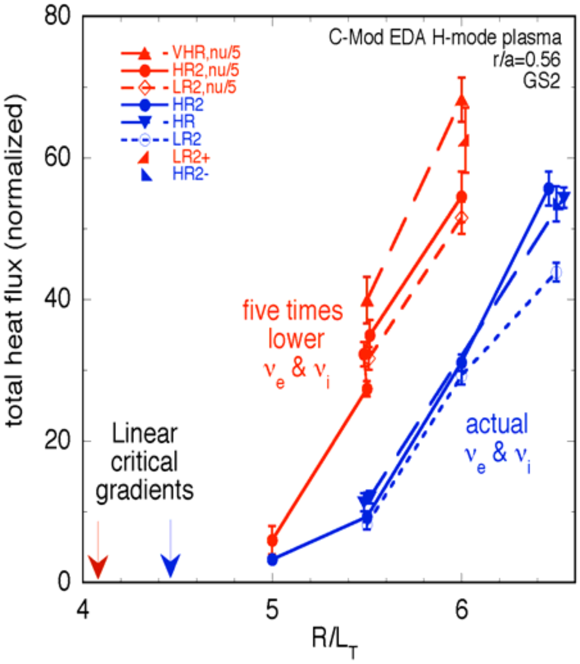

Fig. 2.2.7 Calculations of normalized heat flux vs normalized temperature gradient from the gs2 code. This analysis shows that the change in ITG drive is more important than collisional zonal flow damping in determining the overall dependence on this parameter.

codes. Modification of the magnetic shear will also be employed in tests of models developed for the creation and control of internal transport barriers. The C-Mod internal barriers, described above, are apparently created and controlled through the interplay of R/L_T and R/L_n. While these discharges have steep core pressure gradients and reduced thermal diffusivity, up to now, ITBs in C-Mod show only a weak effect on the temperature profiles. This may be due to the effects of electron-ion coupling, with the barrier reducing transport only in the ion channel. In general, results on a variety of machines show the difficulty of producing transport barriers which operate strongly on both ions and electrons. Exceptions generally involve experiments with very low or reversed shear. The object of this research is to determine whether we can drive $\hat{S}$ near or below 1, and create and maintain ITBs in both channels with strong central heating and in the absence of externally driven plasma rotation, characteristics of ITER plasmas. Since the LH system can vary the launched *k* spectra and thus move the current drive location, it may be possible to control the barrier location directly. The high efficiency of lower hybrid waves for driving off-axis current may allow the creation of discharges where a large portion of the plasma volume is within the ITB.

Collisionality is another quantity predicted to have an impact on turbulent transport. It enters the problem in both direct and indirect manners. First, the instability drive can depend directly on collisionality in addition to the pressure gradient, q profile and other parameters. Collisions can also enter indirectly via their effects on zonal flows, which are believed to be the main nonlinear saturation mechanism for drift-wave turbulence. In this case, collisions are competing with nonlinear dissipation of the zonal flows via additional turbulence effects. Early non-dimensional experiments on C-Mod at high levels of collisionality, ($\nu^* = 0.2 \rightarrow 1$), found a result which was superficially consistent with the linear zonal flow damping model. Normalized confinement time $B\tau_E$ scaled as $\nu^{*-1.0\pm0.2}$ in H-mode with $B\tau_E \sim \nu^{*-0.4\pm 0.1}$ in L-mode. That is, if collisions were regulating the zonal flows, then higher collisionality should lead to lower levels of zonal flows and consequently higher levels of drift wave turbulence and transport. This interpretation is in contradiction to results of nonlinear simulations which found that the reduction of the ITG drive was more important than zonal flow damping in setting the overall level of transport (see fig 2.2.7). These simulations were carried out at somewhat lower values of collisionality than the experiments however. It is also important to note that TEM driven turbulence has a somewhat different balance between instability drive and dissipation. C-Mod will continue to develop a unique diagnostic for zonal flows which is based on the motion of plasma ablated from lithium pellets. We have also begun a collaboration with McKee of U.W. in his experiments which look for zonal flow effects on DIII-D.

With the cryopump, it should be possible to extend C-Mod experiments to lower levels of collisionality. At fixed plasma pressure, even relatively small changes in density can have a large effect on ν^*; values as low as 0.02 may be reachable. This should provide a more definitive test of the predictions and would provide more overlap with other experiments. The role of collisions on particle transport is a topic of current interest for ITER. Experiments on ASDEX-Upgrade and JET have found a systematic increase in density peaking at the lowest values of ν^*, which is the regime of interest for ITER.

Modest density peaking is projected to have an overall beneficial effect on fusion performance in a burning plasma device. However, most experiments, performed so far, involve strong neutral beam heating and at least part of the effect can be attributed to beam fueling. With the cryopump, C-Mod will be in a position to test this issue in the relevant ν^* regime and with no core particle source, perhaps providing a clearer picture.

Plasma rotation can be important for stabilization of micro and macro-instabilities. However, as we move toward ITER and on to reactors, which will have little or no external torque, it is not clear how much rotation to expect. In this light, the observations of strong, self-generated flows on C-Mod are a significant result and one that needs to be better understood. To improve spatial coverage of the core rotation profiles, a new high resolution x-ray spectrometer has been proposed as part of the PPPL collaboration. Adopted as a design for ITER, this instrument will consist of a spherically bent crystal and a 2D position-sensitive detector. It would record spectra of Ar^{16+} from multiple sightlines, measuring $T_i(r)$ and $V_\phi(r)$ with spatial resolution ~1 cm and time resolution of a few ms. As an empirical approach to the problem, the multi-machine database will be assembled and formal scaling studies will be carried out. Long-term research will look at the mechanisms underlying the phenomena of self-generated flows, with an aim of extrapolation to future devices. The role of scrape-off layer flows in setting a boundary condition for the core rotation has been established. We will attempt to determine the relative importance of plasma processes such as turbulent Reynold's stress and collisional viscosity as well as the role of atomic processes like charge exchange in coupling momentum across the separatrix from the SOL to the confined plasma. Experiments will examine regimes where the neutral interactions could be modified, including variations in neutral density, fueling, cryopumping and studies in helium plasmas. Work will also continue on the role of these flows in the L/H threshold. As described above, a number of new and upgraded diagnostics will be deployed to "fill in" the regions of the profile where data are currently unavailable. The flatness of the core rotation profiles in steady-state H-modes suggests a steep gradient region, perhaps a rotation pedestal with a structure similar to the temperature and density pedestal. A related issue is the extent to which rotation gradients in this region contribute to the ExB stabilization of the edge and pedestal. The mechanism by which momentum is transported from the edge into the core will also be investigated, using gyrokinetic codes "instrumented" to output quantities relevant to momentum transport. In addition to profile comparisons, these studies may point to fluctuation measurements which may allow further tests of the simulations.

Further enhancements to core fluctuation diagnostics are planned. The PCI optics will be modified to increase the accessible $k_\perp$ range to 60 cm^{-1}. Hardware will be added to allow fast rotation of the mask used to isolate particular scattered wave polarizations enabling the spatial localization to be scanned rapidly throughout a discharge. To improve the level of sensitivity for turbulence studies, all reflectometer channels will be set up to isolate the phase signals from the upper and lower sidebands and outputting the individual phases (baseband) as well as the difference (AM). The main benefit of this method is that the signal is no longer acquired by subtracting two closely related signals. An additional channel at 140 GHz will be added, extending the density accessible by this diagnostic to $2.4 \times 10^{20}/m^3$, which should allow the core regions of advanced scenario plasmas to be probed. Radial correlation measurements would be carried out using swept-frequency source (Δf = 5-10 GHz) and poloidal correlation measurements made using poloidally displaced antenna and receiver sets.

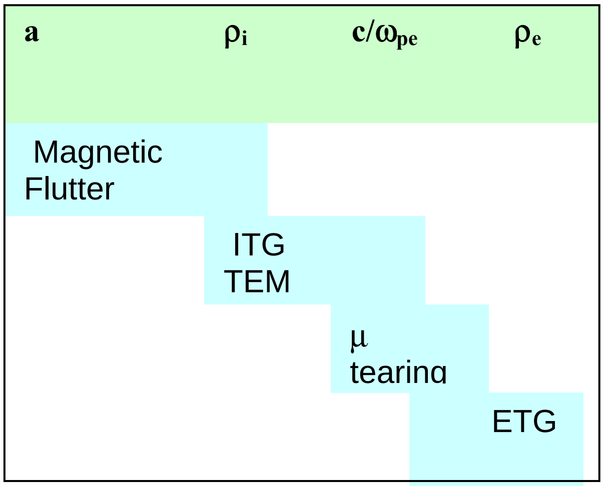

Fig. 2.2.8. Possible mechanisms for electron transport span the entire available spatial range.

The enhanced core fluctuation measurements will be employed to answer several important questions in transport physics. Anomalous electron transport is perhaps the most important unresolved issue in the transport area. In reactors like ITER, the ions and electrons will be closely coupled thus the electron channel cannot be ignored. Little is known about the mechanisms at work; there is not even a clear picture of what range of spatial scales are important. Figure 2.2.8 shows some mechanisms that have been proposed and the scales over which they are expected to operate. These span the entire range available. With the PCI diagnostic, which has excellent signal to noise and will measure fluctuations up to 5 Mhz with wavenumbers up to 60 cm^{-1}, we will attempt to identify the relevant spatial scales in various plasma regimes. The first experiments are aimed at very low density Ohmic plasmas where the transport channels are decoupled and where ion transport is near neoclassical levels. Discussions over joint experiments with DIII-D and NSTX have begun. Modifications to PCI will allow spatial localization of the observed fluctuations. For short wavelengths, this localization should be sufficient to allow measurement of the isotropy of the plasma turbulence. If short wavelengths modes, like ETG, are responsible for electron transport, they must develop extended radial structures which can increase the radial energy flux to the levels seen in experiments. This topic is currently a subject of great controversy and interest in the theory community. As shown above in fig 2.2.6, the PCI can distinguish core modes from edge fluctuations. We note that the broadband core fluctuations are found to propagate in the ion direction, due in large part to the Doppler shift from core rotation. Future experiments will look for residual mode propagation with plasma velocity near zero, for example in L-mode discharges near the L/H threshold. Measurement of the fluctuation propagation in the plasma frame would help determine the nature of the drift wave turbulence, ITG or TEM, that dominates such discharges and could be compared directly to simulations.

Spurred in part by the need to predict the density profiles and impurity content in ITER, there is renewed interest and emphasis on particle transport. As discussed above, studies will be carried out, using the cryopump to access lower collisionality regimes and investigate density peaking with no core particle source. Corresponding theoretical studies will assess the relative importance of TEM and ITG turbulence in this regime. The improved overlap with low-field tokamak experiments will allow more direct comparisons with machines which have reported agreement with particle transport theories based on turbulent thermodiffusion (TTD) and turbulence equipartition models. Currently, these models, which predict $\frac{\nabla n}{n} \propto \frac{\nabla T}{T}$ *and* $\frac{\nabla q}{q}$ respectively, do not appear to be consistent with C-Mod results. Transient transport experiments of particle transport will be carried out in both ITB and H-mode regimes, with the results compared to numerical simulations. The CXR diagnostics will enable studies of impurity transport, especially in the outer regions of the plasma. Finally, the LHCD system should allow investigation of the anomalous particle pinch. With $E_{\Phi} = 0$, as it would be for a fully non-inductive discharge, the Ware pinch is eliminated as a cause for inward convection.

In summary, with its capable and unique facilities, strong diagnostic set and wide collaborations with theory and modeling, the C-Mod experiment offers excellent opportunities to advance the state of transport science. In the coming years, all three of these components will be improved and expanded. The cryopump and LHCD represent important opportunities to extend C-Mod parameters in important directions and will be accompanied by significant upgrades in core profile and fluctuation measurements.

Support for ITER and Connection to ITPA Activities

C-Mod transport research makes or is planning important contributions to the following ITER High Priority Research Tasks:

- Improve experimental characterization and understanding of critical issues for reactor relevant regimes with enhanced confinement, by:
- Obtaining physics documentation for transport modeling of ITER hybrid and steady-state demonstration discharges
- Addressing reactor relevant conditions, e.g., electron heating, Te~Ti, impurities, density, edge-core interaction, low momentum input...
- Contribute to and utilize international experimental ITPA database for tests of the commonality of hybrid and steady state scenario transport physics across devices
- Encourage tests of simulation predictions via comparisons to measurements of turbulence characteristics, code-to-code comparisons and comparisons to transport scalings
- Assemble and manage multi-machine databases, analysis tools, and physics models
- Evaluate global and local models for plasma confinement by testing against the databases.
- Predict the performance of Burning Plasma Experiments using the models, and include an estimate of the uncertainty of the predictions.
- Construct a Profile database based on Intermachine experiments and perform tests of modeling using the profile database as TG work.

- Improve predictive capability of pedestal structure through profile modeling.
- Construct physics-based and empirical scaling of pedestal parameters
- Improve predictive capability for ELM size and frequency and assess accessibility to regimes with small or no ELMs.

C-Mod is currently involved in the follow set of transport related ITPA joint experiments

CDB-4	Confinement scaling in ELMy H-modes, ν^* scans
CDB-8	ρ^* scaling
TP-1	Steady state plasma development
TP-3.2	Investigation of transport mechanisms with Te~Ti
TP-4.1	Similarity experiments with off-axis ICRF-generated density barriers
TP-6	Obtain empirical scaling of spontaneous plasma rotations
PEP-7	Pedestal width analysis via dimensionless identity experiments
PEP-12	Comparison between C-Mod EDA and JFT-2M HRS regimes (recently completed)
PEP-16	C-Mod/NSTX/MAST small ELM regimes
TP-2	Hybrid regime development
TP-3.1	Sustained high performance operation with Ti ~ Te
TP-4.2	Low momentum input operation of hybrid/AT plasmas
TP-4.3	Electron ITB similarity experiments with low momentum input

The C-Mod transport program is also well aligned with the recommendations of the 2005 FESAC priorities panel. Top recommendations for additional funding for high-priority activities included the following areas of particular emphasis on C-Mod:

- *"Carry out additional science and technology activities supporting ITER…"*
- *"Predict the formation, structure and transient evolution of edge transport barriers."*
- *"Mount a focused enhanced effort to understand electron transport"*
- *"Expand the effort to understand the transport of particles and momentum"*

2.3 Wave Plasma Interactions

C-Mod exclusively uses RF for auxiliary heating (predominantly ion cyclotron range of frequency - ICRF) and current drive (lower hybrid range of frequency - LHRF) to heat a wide range of plasmas. Since current ITER plans include 20 MW of ICRF power, we seek to develop an understanding of the underlying technological and physics issues associated with high power ICRF operation. The single pass absorption in H minority heating in C-Mod can be more than 80%, similar to the expected single pass absorption in ITER; thus, C-Mod is an excellent platform to investigate a number of unresolved physics and technological issues. Furthermore, C-Mod discharges can be utilized to experimentally benchmark RF codes for heating and current drive, one of the five high priority CY2006 US tasks assigned by the International Team. The use of LH for off axis current drive is currently under consideration to enable advanced tokamak operation in ITER. The C-Mod LHRF system is optimized for off-axis current drive and is planned to operate at the ITER B-field and density, resulting in the same wave physics as ITER. Thus, the C-Mod experiments are well positioned to contribute to the evaluation of LHCD physics for ITER.

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2.3.1 ICRF

(i) Antenna coupling and Antenna/Plasma interactions

C-Mod provides unique opportunities to explore ICRF antenna design, operation, coupling, and antenna plasma interactions. We have a flexible ICRF system, access to advanced simulation tools like TOPICA and commercial electromagnetic solvers, and advanced diagnostics. Since we rely exclusively on ICRF for auxiliary heating, the ICRF system is operated over a wide range of plasma conditions and the addition of a cryopump for density control may increase access to large ELMs regimes. An ideal ICRF system would have the generator isolated from the load and/or would have the match resilient to load variations and efficiently couple power to the plasma with minimum negative impact on the plasma, particularly the plasma edge. We have identified two principal areas where C-Mod can efficiently and effectively contribute: understanding the underlying physics of antenna power and voltage limits and understanding RF-plasma edge interactions that lead to impurity production, enhanced sputtering, and localized hot spots.

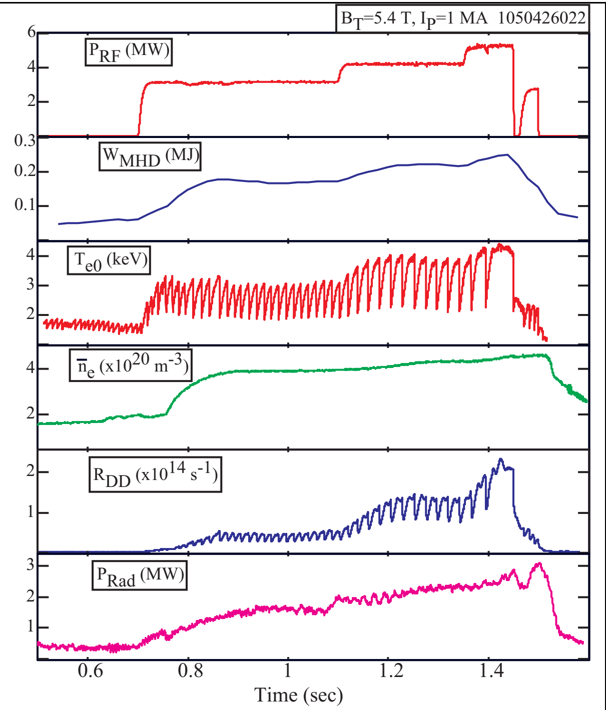

Figure 2.3.1.1: Record discharge summary showing 5.25 MW ICRF power injected, stored energy, central temperature, density, fusion neutron rate, and plasma radiation.

In C-Mod, we have reliably operated two 2-strap antennas, and one 4-strap antenna at high power density (>10 MW/m^2) with minimum impurity production and are planning to design and install a new four strap antenna in FY08. We have used the J antenna (4-strap antenna) to test specific issues like all metal protection tiles, Faraday screen-less operation, and modified power feeds. The compatibility of high power ICRF with all metal plasma facing components (PFC) with high plasma performance is a critical issue for C-Mod and future devices, such as ITER. We re-installed the Mo protection tiles on all antennas where J antenna was operated without a septum, and investigated plasma performance. We have achieved world record volume averaged plasma pressures (1.8 atm) in C-Mod with 5.25 MW of net injected ICRF power (5.5 MW-m^{-3}) in a boronized machine. The key to high plasma performance was determined to be the control of impurities, particularly Mo, through boronization. Since the boronization eroded with ~50 MJ of injected RF power and more quickly for RF than Ohmic H-modes, an RF mechanism is likely responsible for the enhanced erosion rate and is possibly a source of Mo impurities when operating without boronization. One possibility is enhanced sputtering due to RF sheaths and we plan to study this in the near term by investigating boronization lifetime dependence on antenna phasing, single pass absorption strength, and absorption mechanism. We will also investigate filament transport in the presence of RF heating and will begin by using the existing gas puff imaging diagnostic at low plasma current. Depending upon the results, we would like to expand this investigation to more standard 1 MA discharges in ’07 and ’08. Depending on availability of resources, a new, prototypical ITER ICRF antenna reflectometer (in collaboration with ORNL) is planned for installation with the new ICRF antenna (installation FY’08). This diagnostic will measure the local density near the top and bottom of the antenna to determine the extent of the up-down density asymmetry near the antenna. In addition, identifying the location of the important Mo sources will be crucial in developing a technique to improve the boronization lifetime or reducing the impurity influx. In the near term, through field line mapping we plan to identify regions where sheaths are likely and classify the type of field lines involved. For example, some open field lines that pass near the antenna in the scrape-off layer may connect the upper divertor and lower divertor. These field lines would be relatively unaffected by installing insulating limiters and require some other sheath mitigation technique. Starting in ’07 we plan to use marker tiles to help identify the areas of high impurity influx and evaluate potential materials and coatings in off-line experiments.

We plan to continue our collaboration with the RF SciDAC Group and R. Maggiora of Politecnico de Torino on the development of an electromagnetic solver that has a realistic ICRF antenna geometry, presently coupled to a 1-D, soon to be 3-D plasma field solver. To properly analyze and predict antenna performance, geometry is probably the single most important factor. Initial results from the modeling of the E antenna were encouraging and we have taken new data on the screen-less J antenna. In the near term, we will compare these measurements with the simulations and in the future we plan to simulate fast changes in loading to understand the antenna behavior during confinement changes associated with H-mode transitions and ELMs. We also plan to analyze the antenna fields to investigate the role of radial RF B-fields in impurity production from the antenna strap during the J antenna screen-less operation. If their importance is confirmed, we will analyze the new 4-strap antenna design to investigate antenna geometries to minimize these fields. The planned upgrade of the MARSHALL Beowulf computing cluster will greatly improve our ability to study the effects of different antenna geometries on predicted antenna operation.

Another area of research is to develop an understanding of the underlying physics of antenna power and voltage limits. In particular, we modified the J antenna feedthrus to test the effect of vacuum transmission line length on the antenna neutral pressure operation range and found the neutral pressure limit was unchanged. Investigations on a small, table top experiment found that multipactor induced discharge could be responsible for the reduced voltage handling at high neutral pressure. We measured the C-Mod ICRF antennas' susceptibility to multipactor induced discharge and found with ~0.1 T B-field the neutral pressure at which multipactor induced discharge began is nearly identical to the antenna neutral pressure limits. Recent work has also identified that a roughened Cu surface raises the pressure at which multipactor induced discharge is initiated. We plan to test this on the J antenna in FY07 and plan to modify the table top experiment to include magnetic fields and high RF power to investigate the influence of B-fields on RF breakdown. We also plan to characterize the secondary electron emission coefficient of different surface treatments by constructing a device to measure secondary electron emission by FY08.

Plasma load variations are commonly encountered during L/H transitions and edge localized mode activity (ELMs). One potential solution is to deploy a fast matching system and we plan to investigate the integration of a fast ferrites stub system into the E antenna matching network. We have received 2 fast ferrite stubs previously tested on AUG and have found the power supplies to be insufficient. We have identified new supplies commonly found in hospital MRI's and plan to perform tests using a double stub matching network in C-Mod discharges in FY07. If initial experiments are successful we will design an additional 3 sets of ferrite stubs. Implementation of these stubs in the future is budget dependent. With fast matching networks, either passive or active, arc detection becomes an increasingly important aspect of the antenna system. We will continue to investigate new techniques and strategies for arc detection and mitigation.

(ii) Propagation, Absorption and Mode Conversion Physics

C-Mod provides unique opportunities to explore ICRF wave propagation, absorption, and mode conversion physics. These investigations are facilitated by a flexible ICRF system, access to sophisticated ICRF simulation codes (through the RF-SciDAC Initiative and international collaborations), and the availability of advanced diagnostics for RF wave measurements. Realizing high heating efficiencies in D(^{3}He) discharges, where the single pass absorption is weak, is important for planned 2 MA, 8T operation in C-Mod and future experimental devices. We have confirmed that the heating efficiency is a sensitive function of the ^{3}He fraction, more so than expected from theory and more sensitive than the H concentration in D(H). In L-mode, the heating effectiveness compared to D(H) is similar but in H-mode the plasma performance obtained thus far has H89~1.4-1.5 and the improved performance resulting from boronization appears to erode more quickly compared to D(H). An outstanding question regarding D(^{3}He) is the role of parasitic absorption (e.g. boron minority resonance) loss mechanisms near the plasma edge. A similar parasitic minority resonance will be present in ITER from the planned use of Be coatings on plasma facing components (PFCs). In the near term, experiments are planned on C-Mod to move these edge resonances either into the plasma core or out of the plasma. We also plan to continue experiments where direct comparisons of D(H) and D(^{3}He) can be done in the same discharge using the D and E antennas at 80 MHz and J antenna at 50 MHz. Further experiments in FY07 will investigate the effect of additional heating power on heating efficiency. With higher power density, and consequently higher bulk plasma temperature, we expect to increase the single pass absorption in D(^{3}He).

We have extended the investigations of ICRF mode conversion and have made the first measurements of all three waves in the mode conversion region using Phase Contrast Imaging (PCI) diagnostic (DoE Diagnostic Initiative). Furthermore a synthetic PCI diagnostic has been implemented in TORIC in collaboration with the RF SciDAC Group. The data and simulation are in remarkable agreement suggesting that the physics model and numerical algorithm in TORIC models the mode conversion process very well. As part of the PCI upgrade, we plan to implement a masking technique that relies on magnetic pitch angle dependence of the scattered signal to provide localization information along the measurement chord. This will allow us to explore the predicted up-down asymmetries associated with mode conversion and make detailed local measurements of the short wavelength modes. Finally an optics upgrade will allow resolution of higher wavenumber (k) and when combined with the localization may allow observation of the rapid k up-shift associated with the mode converted waves. Taking advantage of this unique capability, studies will initially focus on D(^{3}He) and D(H), the primary species mix used in C-Mod. An important aspect of these studies is comparing the measured spectra with TORIC simulations. In collaboration with the RF Sci-DAC Initiative, we have local access to the MARSHALL Beowulf cluster to run up to at least 511 poloidal modes routinely and the planned FY06 upgrade to this cluster will enable more detailed analysis.

(iii) ICRF Mode Conversion Current Drive

While not expected to be as efficient as Lower Hybrid current drive, MCCD is predicted to be localized and can provide central current drive in scenarios where the LHCD is expected to be off axis. Furthermore, current drive using mode conversion has interesting physics questions that C-Mod can address, particularly the inherent up-down asymmetry associated with the mode converted waves. Recent experiments sweeping the mode conversion location from inside to outside the q=1 surface with heating and current drive have shown the driven current is localized and is consistent with the inherent up-down asymmetry in the mode converted spectrum. Due to this up-down asymmetry, TORIC predicts that heating phase will result in some net driven current and further experiments are planned to test this prediction. These experiments will provide a good test of TORIC and its associated current drive model. Previously, the current drive model coupled to TORIC was based upon the Ehst-Karney current drive efficiency parameterization. This formulation directly included particle trapping by convolving the local power absorbed with the current drive efficiency at that position and the parallel wave number variation is directly accounted. To improve the treatment of magnetic trapping and mode converted wave polarization, TORIC has been coupled with the Fokker-Planck code RELAX. Initial calculations show differences between the Ehst-Karney parametrization and the Fokker-Planck results and further study is planned to use another Fokker-Planck code that includes a more precise treatment of trapped particles near the trapped-passing boundary, critical for MCCD. Additional experiments investigating MCCD are also planned including injection into the initial current ramp-up and during the current flat-top. These experiments are planned with the J antenna at 50 MHz allowing MCCD experiments in D(^{3}He) plasmas at 5-6 T. The driven current will be deduced from analysis of the surface loop voltage, internal inductance, modification of the Alfvén spectrum, and measuring the changes in the current density profile with motional stark effect (MSE) and/or Faraday rotation diagnostics as they become available. In the near term, estimates of the driven current profile can be derived through modeling the sawtooth period variation using the Porcelli sawtooth model. The Porcelli sawtooth model is planned to be implemented in TRANSP by the end of FY07. As q-profile measurements become available, the driven current and profile will be compared with TORIC simulations. With the implementation of the cryopump for density control in FY07, a MCCD scenario with ~100 kA of driven current for 3 MW injected is possible.

For the central current drive case, predicted driven current exceeds the local Ohmic current density for low density and low Z plasmas. This may allow production of discharges with high-l_i, which have shown confinement improvement on other devices. For the off-axis case, the RF current density is a significant fraction of the Ohmic current density. This may allow a study of sawtooth stabilization as shown in the initial experiments and stabilization of pressure driven modes through local current profile modification.

(iv) Flow drive (MC)

Another important research theme, relevant to triggering and controlling transport barriers, is RF driven flows. Theoretical calculations are difficult because one must calculate the plasma response in addition to the RF fields and resulting force. Experiments may provide insight into which of the many terms in these equations are important. For example flows can be driven by pondermotive forces or Reynolds stress. In the former case, damping on electrons may result in driven current but in the second case damping on electrons will be small (electron to ion mass ratio is small) and ineffective. Depending upon species mix, deposition location, and plasma current, the power can be channeled to ions or electrons. With present diagnostics, the poloidal rotation, RF power deposition, and RF density fluctuation profiles can be simultaneously measured. These data will allow an assessment of the amount of poloidal flow, its profile, and its relation to RF wave propagation and absorption. Modeling indicates that the scenarios most likely to drive significant sheared flow are those that have significant damping on ions at the cyclotron resonance where the forcing term switches sign crossing the ion cyclotron resonance. This suggests current drive experiments where the plasma conditions are tailored to maximize electron damping, while flow drive experiments will require different plasma conditions so that ion cyclotron damping will dominate. Thus the first task is to identify a discharge scenario where significant power can be absorbed at the cyclotron layer. Once a target discharge is established, we plan to assess the poloidal flow characteristics. Depending on its success, RF driven flow shear can be investigated to determine RF power required to trigger or maintain internal transport barriers.

(v) Physics of Energetic Ions

Energetic ions physics is of obvious importance to burning plasma experiments. In particular, the influence of energetic particles on MHD mode stability and their effect on fast particle transport are important for ICRF heating and burning plasmas. A combination of new diagnostics, increased ICRF power, and access to new sophisticated codes has placed C-Mod in position to provide strong contributions in this area.

In the past, C-Mod energetic ion effects had been limited to experiments with relatively low density. With the successful operation of J antenna to 11 MW/m^2, energetic ion effects have become more ubiquitous. Furthermore, an active charge exchange neutral diagnostic has produced initial data providing a new ability to monitor the energy and possibly spatial distribution of the ICRF minority tail ions. For 0.8 MA discharges with central D(H) minority heating, the sawtooth period was observed to be antenna phase sensitive and insensitive to phase when the ion cyclotron resonance is near the q=1 surface. For 1 MA discharges, the sawtooth period is lengthened for all antenna phases. We plan to determine the parameter space under which small and monster sawteeth are obtained. These results will be analyzed with a new finite banana width Fokker-Planck code (ORBIT RF) with self consistent RF fields, developed through the RF Sci-DAC Initiative, to identify the role of fast particles in sawtooth stabilization.

Alfvén modes driven unstable by the energetic ions are another important physics topic with obvious implication for burning plasma experiments. In ITER, intermediate toroidal mode number (core localized) Alfvén modes are expected to the most unstable Alfvén modes. These modes could have an impact on the slowing down of alpha particles and impact energetic ICRF ion transport. In an ongoing effort, C-Mod data are being used to experimentally validate state of the art computational simulations and theoretical predictions in an effort to understand the underlying physics. With early injection of high power ICRF, reverse shear Alfvén eigenmodes or Alfvén cascades are readily obtained. Both global-like modes (measured by the edge B-dot probes) and core localized modes (measured by PCI) have been observed. These modes typically have an offset starting frequency and the different modes sweep upward in frequency at different rates as the minimum q decreases. Recent modeling with NOVA-k solving the finite pressure modified ideal MHD equations correctly predicts the offset starting frequency. In the near term, the temperature scaling of the starting frequency will be explored in detail through simulations and experiments. With additional high power RF a second harmonic of the cascade has been observed and comparison with theoretical predictions is planned. We have installed new current probes at the K port limiter and an additional set is to be installed in late FY06. These additional probes should allow a more accurate determination of the toroidal mode number beginning in FY07. Experiments will investigate the impact of deposition location and antenna phase on the excited modes.

(vi) **Connection to FESAC Priorities Panel Recommendations and ITP Activities**

C-Mod is also well aligned with FESAC Priorities Panel recommendations. Under the waves and energetic particle recommendations, a fundamental issue raised in the report is the interaction of electromagnetic waves with plasma. The proposed research strategy is three pronged: optimize externally-launched waves spectra and power coupling limits, develop detailed understanding of wave propagation, absorption and plasma responses required for practical applications, and elucidate the interaction between waves, stability, and transport for potential development for control and optimization of fusion plasmas. As noted above, research in all three of these areas can be readily identified from ICRF antenna development and modeling through off axis current drive with LHCD to influence plasma stability and transport. In addition, another area of interest is interaction of high energy particles with plasma. Here, C-Mod can address the interaction of energetic ions created via ICRF and energetic electrons driven by LHRF. The FESAC Priorities Panel recommends a parallel approach developing an understanding of the internal features of energetic particle-excited instabilities and synergistic behavior of alpha particle-dominated burning plasmas. C-Mod is well positioned to contribute on developing a fundamental understanding of energetic particle modes and these modes can be explored over a wide range of plasmas with access to sophisticated diagnostics and simulation codes

Although no dedicated ITPA organization exists to address heating and current drive, C-Mod contributes in joint experiments utilizing the RF for localized tailoring of temperature, momentum, or current profiles. In the near term, we contribute to MDC-5, comparison of sawtooth control methods for NTM suppression

2.3.2 Lower Hybrid Range of Frequencies

Recent Accomplishments

A Lower Hybrid Current Drive (LHCD) System has been implemented on Alcator C-Mod. The central objective is to use LHCD to supplement the bootstrap current in order to develop attractive steady-state regimes in Alcator C-Mod, namely those with high bootstrap fraction f_{BS} (up to 70%), high β_n (~ 3) and good confinement H_H ~ 1-2. If successful, these experiments will support the basis for near-steady-state operation of ITER and will be pivotal in deciding whether to implement an LHCD system on ITER.

The main components of the system are an RF transmitter composed of 12 klystrons, capable of delivering 3 MW of source power at a frequency of 4.6 GHz for 5 s, and a launcher consisting of 96 sub-height waveguides arranged in 4 rows of 24 waveguides each. The power from each klystron is split 8 ways and delivered to two waveguide columns, as indicated schematically in Figure 2.3.1. A single oscillator provides the excitation of all klystrons, but the amplitude and phase of the input to each klystron is electronically controllable by a vector modulator that provides programmable in-phase and quadrature-phase components to its klystron. In this way, the relative phase at the mouth of each pair of waveguides can be independently varied over the range 0 - 2π with a time response of ~ 1ms. This in turn allows a wide and rapid variation in the parallel index of refraction, e.g., 1-4, a critical parameter in determining the location of the driven current.

The launcher is movable over a distance of several cm, and is protected on each side by molybdenum limiters (see Figure XX.) Langmuir probes, mounted in the face of the coupler are used to measure the density and its gradient at the grill, which are critical parameters in determining the coupling efficiency.

In initial experiments during FY05, measurements of the coupling efficiency have been carried out as a function of phase progression and density at the grill mouth. The global reflection coefficient was determined from forward and reverse power measurements made at the outputs of the second power splitters shown in Figure 2.3.2.1. The measurements were made at the power level of 100-200 kW and showed that very efficient coupling efficiency could be obtained. At the optimum current drive phasing of 90° and a density at the grill of ~3e18 m-3, over 90% of the power applied to the launcher was radiated into the plasma. The dependences on grill density and phase agree qualitatively with code predictions; more quantitative comparisons will be made when the experiments resume in the Spring 06 campaign.

The first set of couplers was manufactured using a titanium alloy. The initial experiments were interrupted by severe interaction of the titanium with hydrogen gas and/or plasma, which led to loss of integrity of the couplers. While titanium is known to form hydrides in hydrogen, the reaction generally requires much higher temperatures than was experienced by the grills in Alcator C-Mod. An interaction with discharge-cleaning plasmas is suspected, and investigation of this possibility is continuing off-line. There could be potentially important consequences if this is found to be the case, since titanium is specified as the material for connecting blanket modules in ITER to the vacuum vessel. In the case of Alcator C-Mod, extensive dust in the machine identified as titanium hydride forced curtailment of LH experiments and removal of the launcher from the machine after having been installed in Alcator C-Mod for just over 3 months.

New coupler grills have been fabricated from stainless steel and the launcher is now reinstalled in Alcator C-Mod. Each of the 24 waveguides in the couplers has a ceramic window that provides a vacuum barrier. Brazing these windows into the stainless steel guides required an aggressive R&D program to solve problems created by the relatively large difference in the coefficients of thermal expansion between ceramic and stainless steel. Lower Hybrid experiments are now resuming and will continue throughout the Spring campaign.

Near Term Research (FY 06)

Lower hybrid experiments will resume in the Spring campaign, initially at the previous power levels of 100 – 200 kW which will permit quantitative comparison of coupling results with model predictions. While the new couplers were being fabricated, over 400 diode detectors were calibrated as a function of detected power level, a task that was necessary to accurately assess the absorbed power fraction. New control software developed for the purpose allows the data required to evaluate coupling efficiency as a function of waveguide phasing and density at the grill to be taken in only a few shots.

Our main goal during the campaign will be to raise the coupled power to levels in excess of 500 kW, where modeling predicts that effects due to a non-thermal electron distribution function should become evident (see Figure 2.3.2.3). A 32 channel X-ray spectrometer has been built and installed for the purpose of measuring the Bremsstrahlung produced by the non-thermal electrons. In addition, non-thermal synchrotron emission is also predicted to occur at this power level and should be observable on ECE diagnostics at frequencies below the second harmonic, as shown in Figure 2.3.2.4. These measurements should provide information concerning the propagation and damping of the lower hybrid waves, and yield an assessment of the efficiency of current drive even at a relatively low power level. Although the distribution function cannot be uniquely determined from the X-ray emission, modeling of the relationship between Bremsstrahlung emission and the electron distribution function allows quantitative information concerning $f_e(\vec{r},\vec{v})$ to be obtained using codes developed for this purpose, namely DKE (in collaboration with Y. Peysson and J. Decker (Tore Supra)) and CQL3D for which a synthetic diagnostic has been written in collaboration with R. Harvey. We are also assessing, through simulations, options for adding additional ECE views which are optimized for obtaining information on the non-thermal distribution.

Midterm Experiments (FY 07)

The next goal of these experiments will be to raise the power level to the maximum that can be handled by the grill, which is expected to be about 1.5 MW. At this level, up to ~200 kA of current driven by the LH waves should be possible. This will then provide a tool that can be used for control of the current profile, in particular for developing regimes with $q_0 > 1$. Varying the parallel index of refraction should enable some localized control of the magnetic shear and allow the study of its effect on confinement. It will be important in these experiments not only to infer the LH driven current via X-ray emission but also the total current via MSE and/or polarimetry. A key issue is the compatibility of the LH system with the ICRF antennas. Substantial ICRF heating power will be required to simultaneously raise β_n to ~ 3, a value which is needed to replace most of the Ohmic current drive with bootstrap current consistent with our goal of developing near steady-state regimes.

Figure 2.3.2.3 shows the results of modeling the effect of 2 MW of absorbed LH power for two values of the antenna phase progression in a low density Alcator C-mod H-mode plasma. The driven current drive at fixed phasing scales as P/n and the results show the importance of increasing the available power, as proposed below, and operating at moderately low density which should be facilitated by the new pumped divertor.

Longer Term Research (FY 08 and beyond)

The longer term goal is to increase the driven current capability by adding a second LH coupler and bringing the total power up to a source level of at least 4 MW. In the present configuration with 12 klystrons and one antenna, the available power is expected to be limited by the antenna to a level of ~ 1.5 MW. By adding a second antenna, up to ~ 3 MW could be coupled, thus increasing the current that could be driven by the LH waves to over 300 kA (depending on density) while maintaining a power of 1.5 MW per grill. A second antenna also allows flexibility; by launching waves with different parallel indices of refraction symbiotic effects due to compound spectra can be investigated. This would provide an additional means of controlling the driven current profile. Although 16 of the original klystrons from the Alcator C experiment are available, a few are in need of new cathodes. We plan to rebuild these tubes to use in conjunction with the new antenna. However, even if the refurbishment is successful there will be no spare tubes. Therefore we propose to purchase two new klystrons in FY08 under incremental funding. The experience in fabricating and operating the present antenna will be used to design and build the second antenna beginning in late FY 06. The fabrication should be completed in FY07 and installed in Alcator C-Mod in FY 08. Funds for the second antenna are included in the baseline guidance budgets for FY07 and FY08.

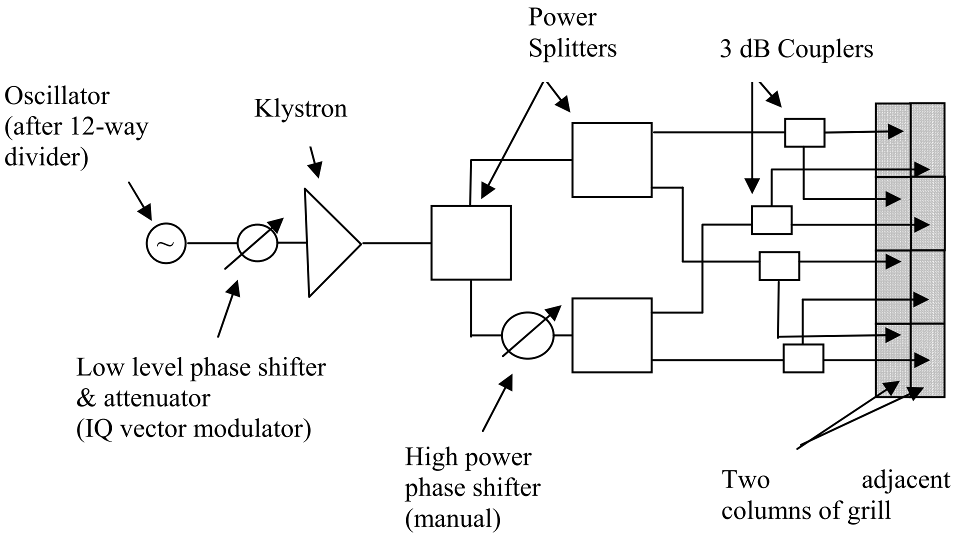

Figure 2.3.2.1. Schematic of way in which power from each klystron is split to excite 2 columns of waveguides in grill.

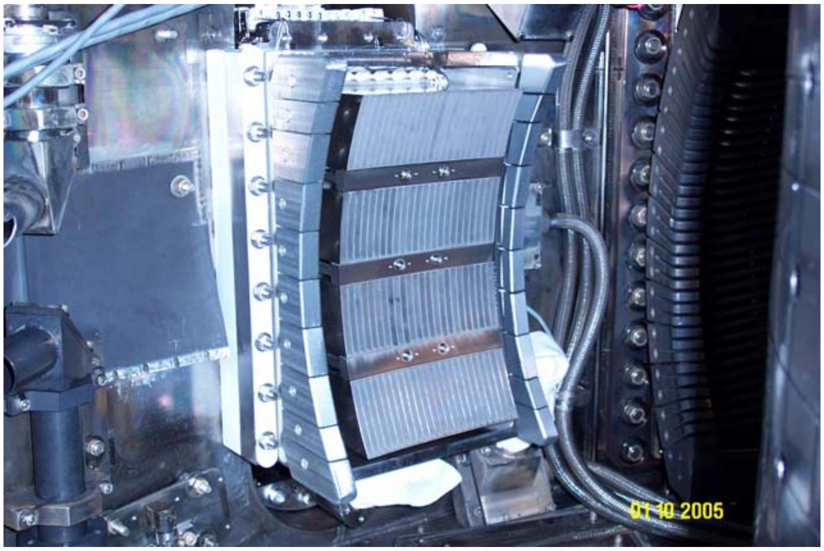

Figure 2.3.2.2. photo of the launcher grill as it appears installed in Alcator C-Mod. Note the protection limiters and 6 Langmuir probes imbedded in the grill.

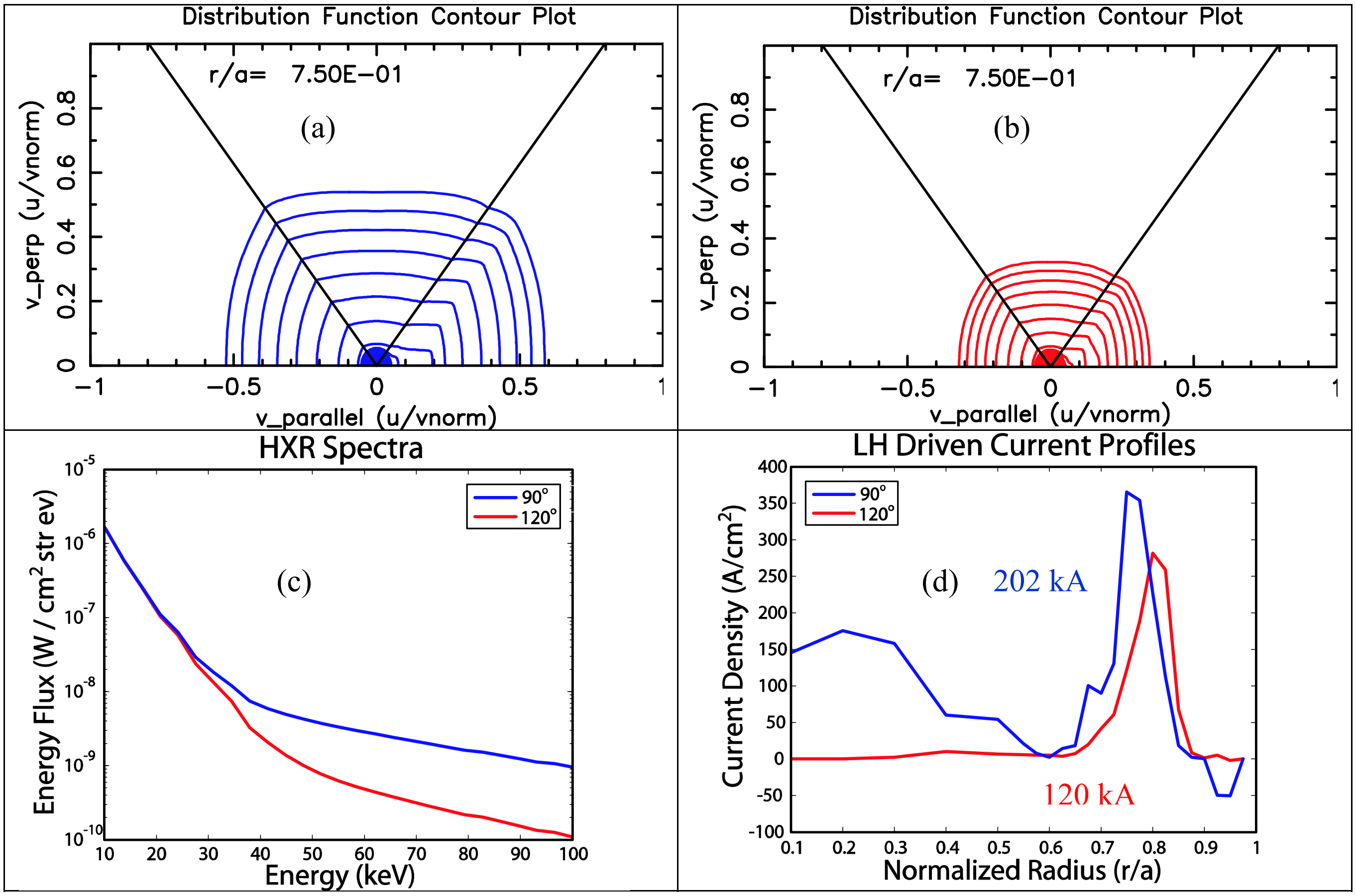

Figure 2.3.2.3. Results of using GENRAY and CQL3D codes to model performance of LHCD in a low-density Alcator C-Mod H-Mode discharge. Figures a) and b): the electron distribution function for 90 and 120 phasing; c) the corresponding Bremsstrahlung emission; and d) the distribution of driven current for the two phasings.

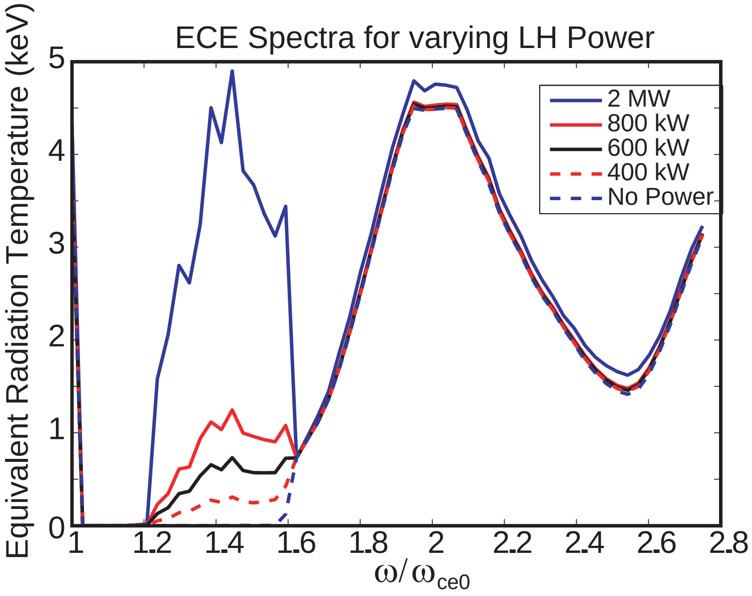

Figure 2.3.2.4: Simulations of horizontal electron cyclotron emission with LHCD, using the CQL3D and Horace codes. A strong non-thermal feature is seen below the second harmonic above ~400 kW, while the temperature measurements are little perturbed.

2.4 Macroscopic Stability

The Alcator C-Mod macrostability research program addresses issues relevant to the overall C-Mod program goals, as well as within the context of international research thrusts. A large fraction of the MHD research on C-Mod involves close collaboration with other facilities, many through official ITPA-coordinated joint experiments. This leverages C-Mod's unique region of parameter space to better determine scaling laws relevant to ITER and future reactors. The C-Mod MHD program also has excellent connections with theory and modeling.

Disruption mitigation

Disruptions are currently seen as one of the most urgent ITER physics issues. The development of practical disruption mitigation techniques is a critical item for any tokamak-based burning plasma experiment and reactor prototype. Disruption-related problems that are particularly severe for ITER-class devices include thermal damage (ablation/melting) to divertor surfaces, $J \times B$ mechanical forces on conducting structures arising from halo currents, and runaway electron populations generated during the current quench by avalanche amplification. Tests of disruption mitigation using high-pressure noble gas jet injection are specified as a high priority item by the ITPA MHD expert group. Experiments have recently begun on Alcator C-Mod and show very promising results. Due to its ITER-like plasma pressure and energy density, as well as its very high current density and fast disruption timescale, C-Mod provides a very challenging test of this technique, both in terms of the gas jet penetration, and in the ability to radiatively dissipate at the required power levels. Significant reduction of halo currents has been demonstrated with high-Z (neon, argon, and krypton) gas jets, as shown in Figs. 2.4.1 and 2.4.2. Substantial increases in radiated power fraction have also been obtained, resulting in decreased heating of the divertor surface, as shown in Fig. 2.4.3. These successful

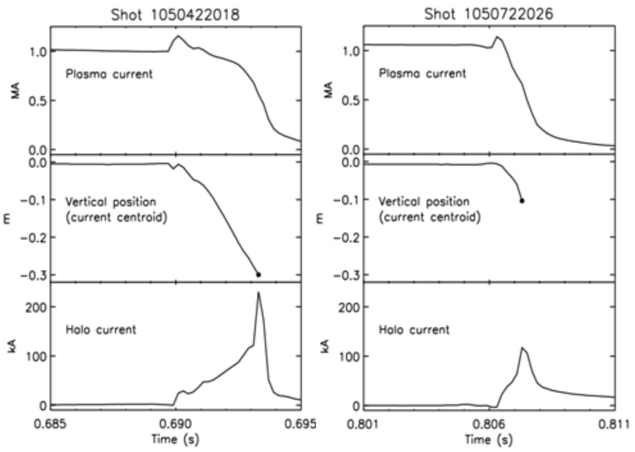

Figure 2.4.1 — Neon gas jet injection speeds up the current quench (right-hand case) compared to an unmitigated disruption (left-hand case). This allows less time for the plasma to move toward the divertor and results in less halo current.

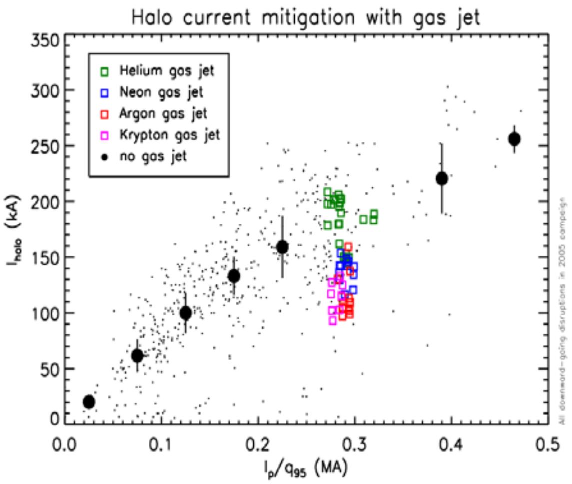

Figure 2.4.2 — High-Z (Ar, Kr) gas jet injection reduces halo currents by up to 50% compared to unmitigated (black points) cases

mitigation effects have been obtained despite the fact that the injected impurities do not penetrate far into the C-Mod plasma as neutral gas jets. Modeling of the gas jet effects using the 3D MHD code NIMROD reveals that rapid growth of tearing modes creates ergodic field regions over most of the plasma cross-section, which allows rapid transport of impurity ions into the plasma core and fast transport of thermal energy out to the radiating mantle. For these initial mitigation experiments on C-Mod, the gas jet system has been fired at a predetermined time into stable (i.e. non-disrupting, stationary) discharges. Over the next two years, this work will be expanded to study gas jet mitigation of unstable plasmas. Initially this will be tested on purposely-generated VDEs by turning off the vertical position feedback at a predetermined time, since this can still be done in a synchronous manner. Eventually a real-time trigger to fire the gas jet will be generated by the digital plasma control system (DPCS) using an internally-generated *z*-position error signal. The ultimate goal is to be able to recognize and predict an impending disruption due to a number of different causes, such as locked modes, impurity injections, β-limits, etcetera, in addition to VDEs. This may involve the development of neural network predictors, and will be a significant challenge given the characteristically short response times necessary to mitigate C-Mod disruptions. A redesign of the gas jet delivery system is planned to shorten the gas delivery time by moving the fast actuating valve closer to the port, and to increase the amount of gas delivered by replacing the valve with a new one being developed by ORNL which has a throughput diameter of 24 mm, in contrast to the present 4 mm valve. Near-term work will also concentrate on improving the computational modeling by incorporating realistic impurity transport and radiation physics from the KPRAD code into NIMROD, and benchmarking the calculations against experiments.

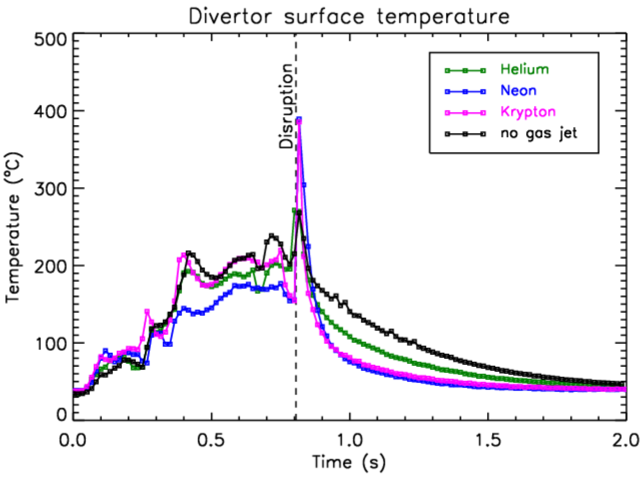

Figure 2.4.-3 — High-Z gas jets quickly radiate away most of the energy, resulting in less heating of the divertor surfaces.

Locked Modes and Error Fields

Locked modes and error field studies have become a major topic of research on C-Mod and are also an ITPA high-priority item because of the related issue of rotational stabilization of resistive wall modes. The desire is to understand the locking threshold, particularly its scaling with machine size (i.e. major radius, *R*), so that the maximum allowed error fields on ITER can be predicted and applied to the design constraints on its error correction coil system. The parametric dependence of the normalized error field locking threshold (B_{pen}) is usually expressed in the following form:

$$\frac{B_{pen}}{B_T} \propto n^{\alpha_n} B^{\alpha_B} q^{\alpha_q} R^{\alpha_R}$$

One way to obtain the *R*-scaling is to perform dimensionless identity experiments among machines of different size by matching the relevant plasma parameters (including the error field mode spectrum) and measuring the error field threshold for locking. C-Mod has been carrying out such experiments with DIII-D and JET under the framework of the ITPA, and plans to continue this work in the future. A second way to derive the size scaling exponent is to utilize the Connor-Taylor dimensionless scaling constraint that exists among three of the exponents in the previous equation:

$$\alpha_R = 2\alpha_n + \frac{5}{4}\alpha_B$$

A linear dependence with density (α_n=1) has been seen on all machines, including C-Mod. Therefore the size scaling is directly related to the *B*-field scaling exponent, which can be determined by performing a field scan *on an individual machine*. In principle, this second method may provide a more reliable answer, assuming the dimensionless constraint is valid, since it avoids the problems of possible unknown or uncontrollable differences between machines. Such a field-scaling experiment has been done recently on C-Mod, holding other relevant dimensionless quantities fixed at their ITER values. The result, shown in Fig. 2.4.4, is an inverse scaling with *B*, which implies a favorable scaling with size ($R^{0.68}$), and translates into a tolerable error-field threshold ($\tilde{B}/B \approx 10^{-4}$) for ITER ohmic target plasmas, which is well within the correction capabilities of the proposed ITER correction coils. However, this result is inconsistent with the size scaling determined from a similar analysis of the C-Mod data obtained from the dimensionless scaling experiments with JET, which yields a result that is much more pessimistic for ITER. The reasons for this discrepancy are not yet understood, but the parameter ranges, the plasma shape, the applied error field structure and magnitude, and the method for finding the locking threshold were all very different. Experiments aimed at understanding and resolving the different methods are planned. In particular, the difference is larger at high toroidal fields, so both experiments on C-Mod will be extended to 8 tesla. It may also be possible to redo the scans so that the applied error field dominates the intrinsic error, although it may require an upgrade in the A-coil power supply (possibly in the 2007-8 time frame) to increase its current output. In principle this would eliminate one of the most important sources of discrepancy. Dependence on error field mode structure will also be studied. Additional experiments to examine locked mode thresholds in H-mode plasmas and to further characterize the intrinsic and applied error fields are planned as well. Recent results from other machines suggest that non-resonant error fields may not be negligible in terms of their effect on plasma rotation braking, and we intend to look into this

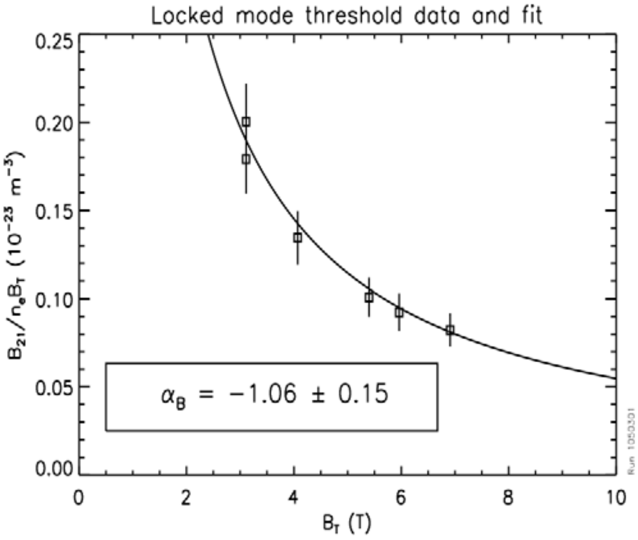

Figure 2.4.4 — An inverse scaling with *B*-field was found for the error field locking threshold under a particular set of controlled parameters on C-Mod.

question. Effects of the error fields on the equilibrium reconstruction, particularly the position of the separatrix, will also be investigated.

TAE modes, Alfvén cascades, and active MHD

The stability of toroidal Alfvén eigenmodes (TAEs) have been studied on Alcator C-Mod using the active MHD antenna system, which generates an ITER-relevant intermediate-n mode spectrum. The dependence of the TAE damping rate, γ, on various plasma parameters can be measured and used to guide and constrain theoretical models of the TAE instability. For example, in certain inner-wall limited plasmas an *n*=9 TAE mode at about 500 kHz was found to have a normalized damping rate of $|\gamma/\omega|$=1.2%. The standard EFIT-generated q-profile from this equilibrium was input to the Nova-K code, which was used to calculate the expected mode structure and real frequency for an *n*=9 TAE mode. Figs. 2.4.5 and 2.4.6 show some of the results from the Nova-K calculation. Both the real frequency (518 kHz) and the computed damping rate (0.01*ω) agree well with the measurements, and provide a good benchmark of the code, at least for these stable TAEs.

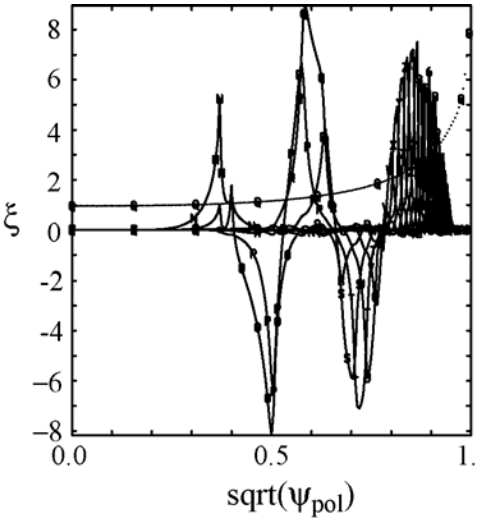

Figure 2.4.5 — TAE mode structure calculated by Nova-K using EFIT *q*-profile.

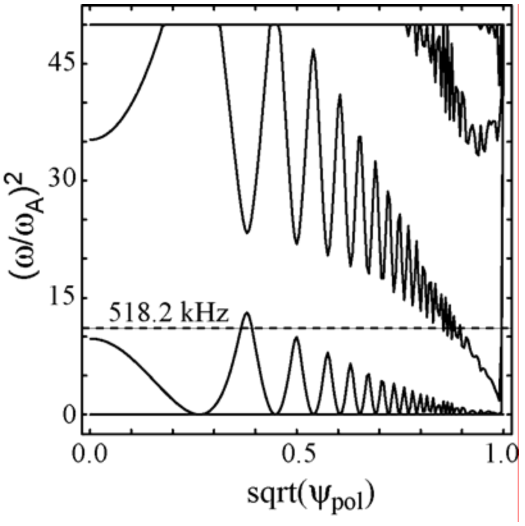

Figure 2.4.6 — TAE frequency gap structure calculated by Nova-K using EFIT *q*-profile.

A high-priority ITPA research topic is to determine if TAEs cause loss of the fast NBI or ICRF ions that couple the auxiliary heating power to the thermal plasma. Most other machines are concentrating on NBI, but an experiment is being planned on C-Mod to look for any reduction in ICRF heating efficiency due to driven TAEs. The active MHD antenna system, however, has relatively low power, so instead plans call for using the relatively strong beat wave produced by the interaction of the D and E ICRF transmitters. These are normally run at 80.0 and 80.5 MHz, resulting in a beat wave at 500 kHz which is clearly seen on the Mirnov pickup coils. With some effort, the frequency difference can be adjusted to 400 kHz, which is within the normal range of stable TAE modes in C-Mod. By varying the density, the TAE resonance can be excited at will with the strong beat wave, and any reductions of heating efficiency in phase with the TAE excitations will be searched for.

Reversed shear (RS) discharges on C-Mod can presently be obtained only during the I_p rampup, but LHCD experiments in the next few years are expected to produce near steady-state reverse shear profiles. In RS plasmas, Alfvén cascades (reversed shear Alfvén eigenmodes, or RSAEs) are observed with both the B-dot pickup coils and the phase constrast imaging diagnostic. The cascades are excited at integer and half-integer values of $q_{\min}$ and exhibit a characteristic frequency 'chirp' due to the mode's sensitivity to (q-$q_{\min}$), which is evolving throughout the I_p rampup. Modeling the RSAE modes with the MISHKA and NOVA-K codes shows good agreement with the observed frequency spectrum and mode numbers. Therefore these cascades can be used to deduce the evolution of the q-profile, particularly $q_{\min}(t)$. The addition of the lower hybrid current drive on C-Mod will extend the usefulness of this technique by allowing the determination of $q_{\min}$ during the current flattop.

RSAE cascades on C-Mod are observed to start at non-zero frequency (typically 150-250 kHz). In trying to understand this, we have found it necessary to incorporate additional physics, specifically finite pressure effects, into the Nova-K code. According to recent theoretical work, this introduces an offset term into the dispersion relation:

$$\omega^2_{AC}(t) = (m\text{-}nq_{\min}(t))^2 V_A^2/R_0^2\, q_{\min}(t)^2 + (2T_e/M_iR_0^2)(1+7T_i/4T_e)$$

which is interpreted as a geodesic deformation of the Alfvén wave continuum. Therefore the cascade starting frequency (when $q_{\min}(t) = m/n$) is predicted to scale like $T_e^{1/2}$ at the $q_{\min}$ location. With finite pressure included, Nova-K corroborates the theoretical prediction and reproduces the experimentally-observed behavior, as seen in Fig. 2.4.7.

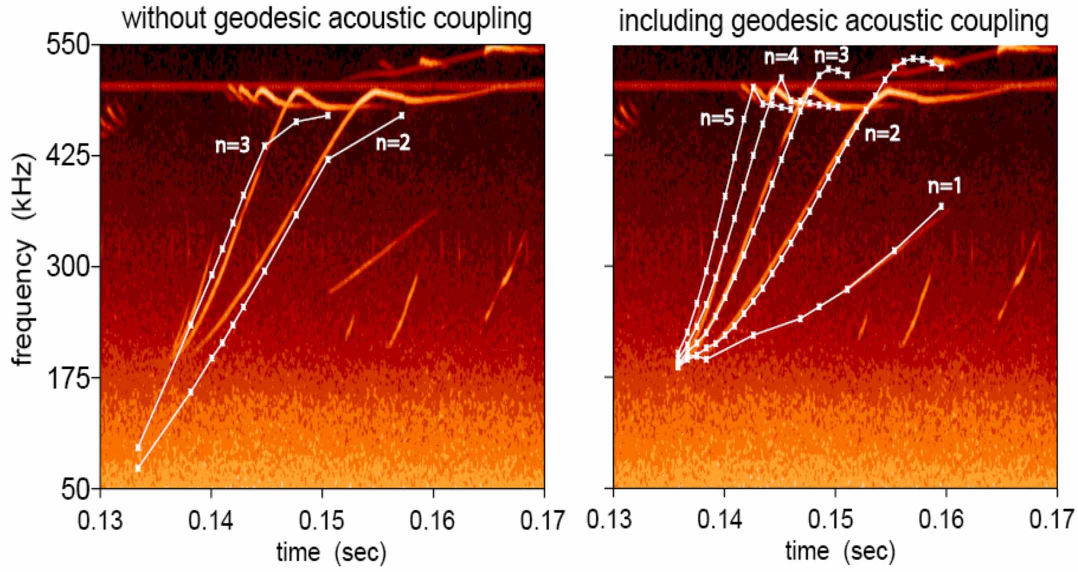

Figure 2.4.7 — Left: the white curves show the Nova-K modeling without the finite pressure term, showing poor agreement at the minimum frequency. Right: inclusion of finite pressure and its gradient in the upgraded Nova-K assures matching of the minimum cascade frequency (200 kHz).

Experimental investigation of the cascade starting frequency does indeed find that it scales approximately like $T_e^{1/2}$. Coupling this information with the measured $T_e(r)$ profile could help determine the location of the $q_{\min}$ radius. Additional measurements and comparisons with theory are planned.

High β instabilities and NTM's

NTM's are the focus of a planned ITPA collaboration with DIII-D and JET to determine the β_N threshold in discharges with matched dimensionless parameters (ρ_*, ν_*, q_{95}). JET and DIII-D have already made non-dimensionally similar discharges that demonstrate the onset of a 2/1 NTM at $\beta_N \approx 3.8$ on both machines, but this is beyond the reach of C-Mod. Therefore a different set of dimensionless parameter values has been proposed, for which DIII-D finds a threshold $\beta_N \approx 1.7$. This should be attainable in C-Mod, particularly with the addition of the lower hybrid power. The corresponding dimensional parameters for C-Mod are $B_T = 5.3$ T, $a = 22$ cm, and $n_e = 2.5 \times 10^{20}$ m^{-3}.

Research on the control and/or elimination of NTM's with RF is also planned. There are several schemes that can be tried, either to reduce or eliminate sawteeth, which would remove the seed island trigger, or by direct stabilization with current drive in the islands at the resonant surface. The possible RF techniques for doing this (ICRF, FWCD, LHCD, MCCD, ICCD) are discussed in detail in the RF chapter.

Digital Plasma Control System (DPCS)

In the past year, several non-linear algorithms have been developed for the DPCS to provide control signals for various purposes. For example, one of the DPCS channels now outputs an estimate of the Alfvén frequency at the q=1 radius, which is used to drive the frequency sweep generator for the active MHD antenna system, allowing the frequency to be varied around the Alfvén resonance even as the plasma equilibrium evolves. The DPCS computes the frequency continuously using the density and toroidal field input signals. During the next few years, algorithms for the DPCS will be developed to recognize and predict the onset of some types of disruptions. This is necessary in order to provide a real-time trigger for the gas jet injection system, which is a priority task for C-Mod and for the ITPA. Due to the relatively short disruption timescale on C-Mod, it may not be possible to generate a useful real-time trigger for all types of disruptions, but our efforts will concentrate initially on generating VDE warnings and locked mode warnings. The VDE predictor will use the DPCS computed error signal on the plasma vertical position (unlike the Alfvén frequency, which just uses input signals The locked mode predictor may initially use combinations of B_p signals, but it may be more robust to develop a non-linear algorithm to recognize the cessation of sawtoothing on the ECE or soft x-ray signals. In the long term, particularly with reversed shear operation at high beta$_p$, neural networks may be developed and trained to predict disruptions. In addition, non-disruption related tasks could also be developed, such as real-time EFIT reconstructions.

2.5 Research for Integrated Scenarios on ITER

This research activity includes experiments and modeling aimed at reaching attractive operating points, generally cutting across multiple science topics and often involving interaction and compatibility issues between different plasma processes or regions. As such, it *integrates* work described in the four preceding topical science sections, and corresponding Scientific Campaigns as defined by the 2005 FESAC panel on Scientific Challenges, Opportunities and Priorities for the US Fusion Energy Sciences Program.

Given the imminent construction of ITER, C-Mod is focusing its integration work to an even greater degree than previously on the target scenarios which are to be demonstrated and explored on ITER. The three main planned ITER scenarios are[1],[2]:

1. *Conventional H-Mode "Baseline" Scenario.*

   This scenario, arguably the most important since it is relied on to provide the target Q=10 fusion performance, features an edge transport barrier but no core barrier and has positive shear, without external current drive; non-inductive fraction of ~25% comes primarily from bootstrap current. Target parameters are $q_{95}=3$, $\beta_N=1.8$ and density ~ 10^{20} m^{-3}, all similar to current C-Mod values.

2. *"Hybrid" Scenario.*

   This scenario, so-named because it is in a sense intermediate between the H-Mode and steady state scenarios, has weak core shear and q_{min} ~1. This has been shown to provide modest core peaking and confinement improvement, though the mechanisms for the sawtooth avoidance and transport reduction are as yet unclear. Higher levels of external heating and current drive are used to reach $\beta_N=2.8$ and 50% non-inductive fraction, projecting to Q=10 performance at reduced current, $q_{95}=4$, and for increased pulse lengths.

3. *"Steady-state" scenario.*

   This 'advanced' scenario relies on strong off-axis current drive to produce weak or reversed shear profiles with q_{min}~2 and q_{95}~5. Projected confinement improvements (H_H~1.2) lead to $\beta_N=3.0$, close to the ideal stability limit, and allow 100% non-inductive current drive and long pulse (~3000 s) operation at a reduced fusion Q = 5. Successful demonstration of this regime on ITER would be an important step towards an attractive tokamak DEMO design. Lower hybrid current drive is under serious consideration for this and the hybrid scenario, as it has the highest off-axis CD efficiency.

It should be recognized that, in ITER experiments as on C-Mod, ranges of parameters will be explored in each scenario. However, reaching these ambitious targets serves as a

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[1] ITER Technical Basis Plant Description Document, published by IAEA (2001)

[2] W. Houlberg et al, Nuclear Fusion 45 (11) 1309 (2005).

useful goal to focus attention on the challenging combination of conditions which must be simultaneously met on burning plasmas. This research therefore contributes most strongly to the second overarching theme of the US fusion sciences program, *"Create a Star on Earth"* (i.e. burning plasmas), as expressed by the 2005 FESAC panel.

Recent Research Highlights

Recent experimental work on C-Mod has focused principally on the first, "Conventional H-Mode", scenario, tackling several of the key remaining issues important for extrapolation to ITER. A notable success, described in more detail in Section 2.1 (Boundary Physics), was in the understanding of the H-mode behaviour with all-metal walls and the deposition, erosion and effects of boron coatings. The choice of low or high-Z wall materials for various parts of the ITER PFCs, and phases of operation, is still under consideration, as each has advantages and concerns; less is known about metal walls since most current machines use carbon PFCs. During an extended C-Mod experimental campaign starting with all boron removed, it was well documented that effective boronization is required for high performance H-modes. This research encompassed most topical science areas; the effect of boronization was seen most directly in the rate of rise of radiation, which in turn affected the pedestal stored energy and ELM regime. Through stiffness in the core energy confinement, consistent with turbulence models, this in turn affected the core temperature and global energy confinement. It was also found that there is a continuum in performance with the thickness and condition of the boron coating. The erosion and subsequent degradation varies markedly with input RF energy, suggesting an RF-edge effect contributes; the location and mechanism for this is under investigation and may lead to further optimization. However, gradual degradation occurs even with ohmic plasmas, so that regular reapplication of boron will be needed.

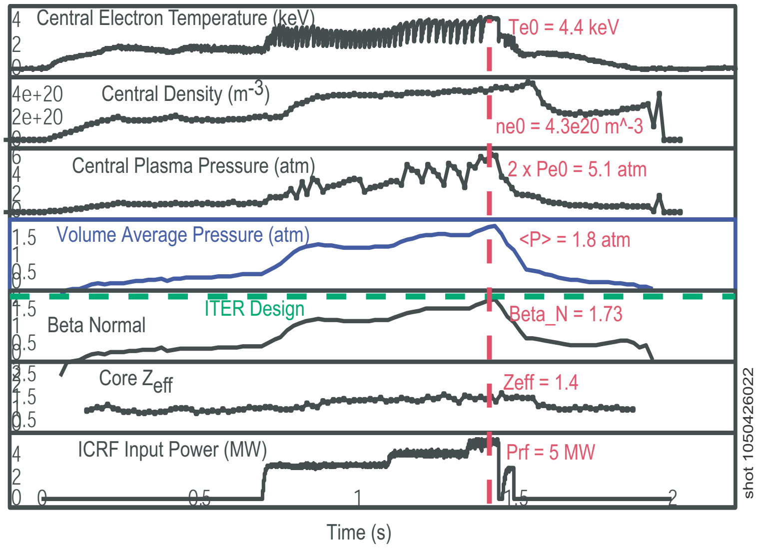

Figure 2.5.1: High performance C-Mod H-mode plasma produced in 2005 with world record plasma pressure <P> = 1.8 atm, at values of toroidal field, 5.4 T, and normalized beta, β_N=1.73, very close to those on ITER.

A key result of this integrated research was an increase in C-Mod performance. Increases in stored energy, 250 kJ, and volume averaged pressure, <P> = 1.8 atm, were achieved with 5 MW of ICRF, shown in Figure 2.5.1. This is not only a C-Mod but a world record for <P> and , significantly, was achieved at the values of toroidal field, 5.4

T and normalized beta, β_N=1.73, very close to those planned on ITER. Good cleanliness was maintained, with Z_{eff} =1.4, below the ITER target. Results in this scenario are therefore encouraging for ITER and provide a relevant regime for many detailed topical physics studies, such as pedestal and ELM research, SOL and divertor studies, disruption mitigation, core MHD modes and RF-plasma coupling studies. The ITER central team has accordingly assigned several urgent physics tasks, on disruption mitigation, divertor radiation transfer, RF code benchmarking and past particle confinement, to the C-Mod group.

Work in FY05 on the development of advanced scenarios, including both 'hybrid' and 'steady state' modes of operation, has focused primarily on the advancement and understanding of profile control tools, in particular for current and transport profiles. These will then be integrated to expand our operational range. A key milestone was the commissioning of the first lower hybrid launcher. Scenario modeling for ITER, an ITPA activity in which PSFC personnel continue to play important roles, has demonstrated that far off-axis current drive is essential for these scenarios and that LHCD offers the best efficiency. Crucial issues of wave coupling, accessibility and damping and current drive efficiency need to be assessed experimentally. C-Mod is the only device capable of performing such experiments in advanced regimes with edge and core barriers, at the ITER B_T and n_e range, and corresponding ω_c and ω_p . The ratios of ω, ω_c and ω_p are the key parameters for wave physics. First low power results showing wave coupling variations with launcher phase and radial position, in qualitative agreement with model predictions, were highly encouraging. Plasma-material interactions leading to erosion of the titanium grill precluded high power current drive experiments in 2005, which are expected with the newly commissioned SS launcher in the spring 2006 campaign.

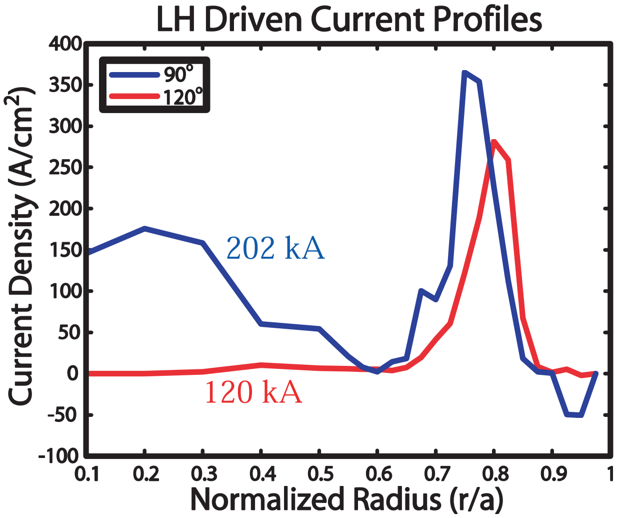

Figure 2.5.2: CQL3D simulations of LH driven current profiles expected on C-Mod with 2 MW of coupled power. Deposition is far off axis (r/a 0.7-0.85) and varies with launcher phasing. Actual H-mode target plasmas were used in this simulation [J. Liptac, APS 2005]

In parallel with this experimental development, tools and diagnostics for modeling LHCD in advanced scenarios were greatly enhanced. The state-of-the-art CQL3D code, a fully self-consistent 2-D (velocity space) Fokker-Planck model which has been shown to predict higher LHCD efficiency in the ITER, and thus C-Mod, wave regime than earlier adjoint codes, was installed at MIT in a collaboration with CompX and is now in routine use, allowing many more parameter scans. Realistic scenarios based on existing H-mode target plasmas were modeled and indicate excellent off-axis localization of j_{LH} at r/a~0.7-

0.85, depending on parallel refractive index $N_{//}$, as seen in Figure 2.5.2. Important new synthetic diagnostics were implemented by graduate students, increasing confidence that we will be able to measure and interpret non-thermal emission using our Hard X-ray and ECE diagnostics. The code will also be useful for ICRH modeling.

Progress was also made in the development and understanding of Mode Conversion Current Drive, which modeling has shown could be a useful supplemental current profile control tool in some scenarios. Clear modification of sawtooth periods, in agreement with expectations, was shown when phase and location of current drive were varied near the q=1 surface (Figure 2.5.3). The amount of on-axis current drive was limited by difficulties in producing discharges with simultaneous low values of n_e and Z_{eff}. Low values of both are needed, as will be the case for LHCD, thus illustrating the need for integration of density and impurity control.

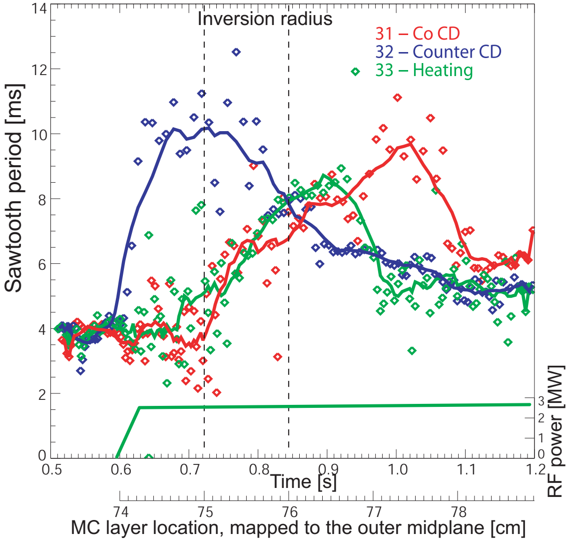

Figure 2.5.3: Modification of the sawtooth period effected by Mode Conversion Current Drive. The period varies substantially as the MC deposition layer is scanned across the q=1 surface. The effect reverses with co vs counter injection, clearly indicating local current drive [A. Parisot, APS 2005]

In the area of transport profile control, the operational space of scenarios with internal transport barriers, which have been discovered on C-Mod to occur with tailored off-axis RF deposition, was extended. Exploiting the new capability of 50 MHz ICRH, barriers were obtained at significantly lower B_T (2.82 T), corresponding as in past experiments at higher B and RF frequency to a deposition location at r/a =0.5. This brings the total range to B_T=2.8-6.4 T, and has enabled a clearer scaling of the barrier width; the barrier foot scales with safety factor q_{95}, not simply with B_T. At low q, very wide barriers were obtained.

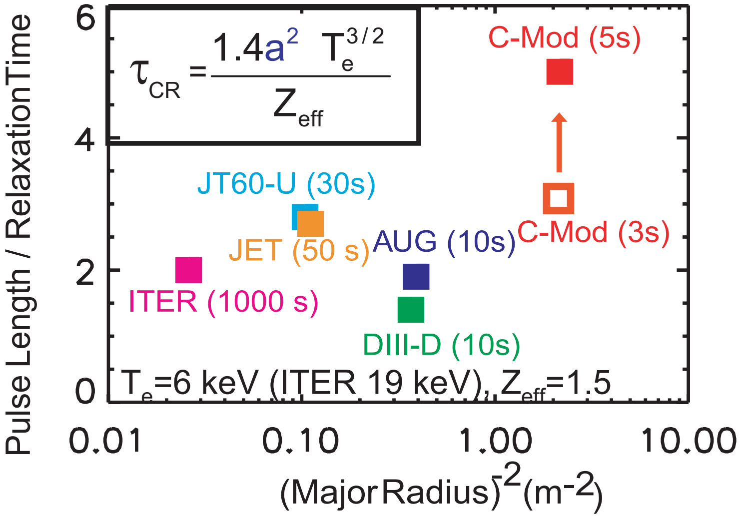

Figure 2.5.4: Normalized pulse length τ_{pulse}/τ_{CR}. Current relaxation times are consistently calculated using Z_{eff}=1.5 and T_e=6 keV for present machines (19 keV for ITER).

A critical strength of C-Mod for advanced scenarios is the ability to run routinely pulse lengths many times the current relaxation time, allowing studies of fully relaxed, quasi-steady state current profiles. The TF system is capable of running five second pulses at 5 T, which will require partially non-inductive current drive, while τ_{CR} is naturally short (~0.1-1 s) due to the compact size of C-Mod. Figure 2.5.4 compares normalized pulse length with current and planned values on some other world tokamaks. To exploit this capability, issues of long pulse heating and power handling also need to be addressed. Experiments performed in 2005 succeeded in making significant advances on this front, increasing pulse lengths in L and H-mode RF-heated plasmas to 3.2 s flat top and nearly 4 seconds total pulse length, and input energy to 6.3 MJ, all C-Mod records (Figure 2.5.5). Newly commissioned fast cameras were used to monitor ‘hot spots’ in the divertor, and these were moved through strike-point sweeping, increasing confidence of use of these techniques for our advanced scenarios with LHCD. No significant difficulties in power handling of either the molybdenum tiles, or of prototype tungsten brush tiles which are being tested to aid possible designs for ITER, were encountered in these experiments.

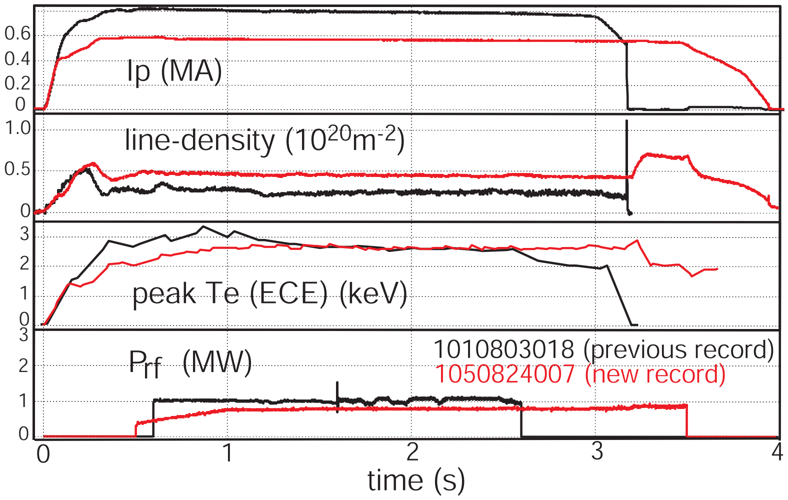

Figure 2.5.5: Long pulse experiments in 2005 (red) significantly extended the durations of the current flat top (top trace) and RF power (bottom trace) compared to prior limits in 2003 (black)

All of these research areas, and others, will be extended in future C-Mod campaigns exploiting new capabilities as described below.

Research Plans

Development of Control Tools

Further development of several crucial control tools will be important for all integrated scenarios. A key addition in 2006 will be a cryopump for density control. This will be installed in FY06 and commissioned early in FY07. The novel upper divertor 'slot' design should allow effective pumping in a wide range of shapes planned for both H-mode and advanced scenarios.

A focus of the FY06 and FY07 campaigns will be the utilization of the recently commissioned Lower Hybrid Current Drive system, with the new stainless steel coupler grill. This work, described more fully in the Section 2.3.2, will begin with further low power coupling studies, and progress to determination of power limits. Extensive scans will then be carried out, to measure the current drive as a function of plasma density, grill phasing and other parameters. These will be compared with modeling calculations from CQL3D and other RF codes, which will be used in conjunction with experiments to optimize the magnitude and location of the wave damping and driven current. The Hard X-ray camera, as well as detectors of non-thermal Electron Cyclotron Emission, will provide important measurements of the LH-driven fast electrons for these studies. The new synthetic diagnostics for CQL3D should be a particular help in this work. A MSE diagnostic should be able to measure changes in local current and a polarimeter for j(r) measurements is also under development.

Installation of a second launcher is planned in FY08. This will reduce the grill power density substantially, allowing optimal use of the 3 MW source power for long pulses of up to 5 seconds, which corresponds to several current relaxation times even at T_e=6-7 keV. Importantly, it will also enable experiments with simultaneous launching of spectra with two $N_{//}$ peaks, which has been shown by both other experiments and modeling to give greater control of the deposited waves and driven current profile. A second upgrade, to be carried out in FY07 under incremental funding, would bring the source power to 4 MW, with a target of 3 MW coupled. This power level, as discussed below, is needed for the most attractive non-inductive scenarios.

We will continue to exploit the flexible source frequencies of our ICRF heating and current drive systems, which have 8 MW source power, half tunable from 50-80 MHz. Near-term experiments will include further assessment and optimization of $D(^3He)$ heating, needed for 8 T H-modes, and optimization of MCCD. A second four-strap antenna will replace the current two-strap antennas, maintaining the full heating capability when the second LH launcher is installed.

H-mode scenarios

Building on the successful high-power H-modes demonstrated in FY05, with ITER field and β_N, H-mode scenario development will focus on key ITER issues and on creating conditions which are closer to those on ITER in other respects. Pedestal and confinement studies will focus more on the regime of small ELMs in addition to enhanced D_α, which is attained at high power and pressure; this seems most promising for ITER. To this end we will increase plasma current and explore the limits of q_{95} for which this regime can be attained; our highest pressure H-modes to date were at $q_{95} = 3.9$, while the ITER reference scenario is at $q_{95} = 3.0$. Similarly, we will use the cryopump capability to reduce density below that set by natural evolution from the L-mode target; at fixed pressure, the resulting increase in T_e should cause a substantial reduction of ν^*. A key issue is how the ELM behavior will respond. Below some limit it is likely that larger, Type I ELMs will occur, which would open new avenues of research. Research into neoclassical tearing modes will be another topic of increasing importance. High performance H-modes are already close to predicted limits. With modest increases in β_N and decreases in ν^*, C-Mod will be positioned to provide important tests of NTM thresholds and RF stabilization techniques.

Following successful demonstration of efficient $D(^3He)$ heating in H-modes at 50 MHz and 5.3 T, some of our future high performance H-mode experiments will be carried out at 8 T. While higher than the ITER value, this would extend ρ^* scalings, and facilitate important ITPA intermachine experiments with larger, lower field tokamaks including JET, DIII-D and Asdex Upgrade.

Research on testing PFC materials and coatings, and their impact on H-mode performance, will continue. C-Mod experiments feature divertor heat fluxes of ~0.5 GW/m^2, approaching that of ITER. Understanding and optimizing boronization application techniques and erosion mechanisms, including RF effects, will be a priority. The long term aim is to develop intershot, and ultimately *during-discharge,* boronization techniques to maintain optimal H-mode confinement and cleanliness. Such techniques would be extremely advantageous, and perhaps necessary, for ITER. Research will extend to include testing of new tungsten tile designs over a greater portion of the PFCs. The duration and input power of long-pulse experiments will be progressively extended, enabling even more demanding tests of all PFCs. Interaction with, and effects on, the core plasma will again be a key part of the experiments.

Other H-mode experiments with direct relevance to ITER burning plasmas include burn control simulations. The newly upgraded digital plasma control system gives the capability to vary input RF power as a function of plasma temperature or neutron production. This will allow us to simulate the excursions which would occur in a burning plasma. The challenge will be to maintain performance during various plasma phenomena such as sawteeth or ELMs. These experiments will be carried out in collaboration with researchers at General Atomics.

Research to understand the physics of radiation trapping in the high n_oL C-Mod divertor will continue. C-Mod is the closest to ITER in this regard, and modeling predicts this will have important effects on detachment. In FY08 we will explore the prospects for radiative divertor H-mode scenarios on C-Mod at higher power than was available in past investigations. High radiated fractions are envisaged in most ITER experiments to reduce divertor heat loads. The issue and challenge here is to maintain high pedestal pressure and global confinement.

Hybrid Scenarios

Once LHCD has been demonstrated and understood, the program will progressively work during FY07 and FY08 to combine it with other current drive tools including MCCD and FWCD and to develop optimized scenarios with higher non-inductive fraction and global confinement. An early focus will be on demonstrating the "hybrid" scenario in C-Mod, which will be a crucial test of whether the relevant physical mechanisms still give improved performance in the ITER-relevant conditions with coupled electrons and ions and without momentum input. As noted above, this Q=10 ITER scenario requires $q_{95}=4$, so that present C-Mod H-mode scenarios with small ELMs make an ideal starting point. Reduction in density using the cryopump would likely be advantageous. Most experiments worldwide employ tailored current rampups to create the needed flat current profile and avoid sawteeth. On C-Mod, due to the short current relaxation time, LHCD will be needed to broaden the current profile and raise q_{min} above 1 for extended periods. This will obviously require less LH power than reversing the shear.

Hybrid scenario research will be supported by an increased effort in integrated scenario modeling to quantify these requirements and optimize the target and LH parameters. Past modeling has been done using the ACCOME code as well as time-dependent TRANSP simulations which used LSC to model LHCD. This will be supplemented in FY07 with modeling by a recently coupled TRANSP-TSC model, in collaboration with PPPL researchers. In parallel, work will be initiated to couple the more accurate CQL3D wave model to TRANSP, which will be an important asset for both LH and ICRH simulations.

Experiments and modeling on the hybrid scenario will be carried out in close collaboration with those on other tokamaks, under ITPA joint experiments SSO-2.1, 2.2 and 2.3 and corresponding TP-2, TP-3 proposals. Unique contributions are anticipated to several High Priority ITPA Needs identified by the Transport and Steady State Operations Topical Group, described below.

Steady State Scenarios

In parallel with the above H-mode and Hybrid scenarios, we will conduct research aiming toward demonstration of high bootstrap fraction, quasi-steady state scenarios. The long-term goals for C-Mod are very similar to the ITER steady-state scenario, with 100% non-inductive current drive, 70% bootstrap fraction, achieved using modestly enhanced global confinement, reversed shear and β_N~2.9, just below the no-wall limit with optimized shaping and profiles[3]. This will again be achieved in the ITER relevant regime with coupled electrons and ions and low rotation, in contrast to other AT experiments worldwide. Modeling shows that the higher power enabled by the second LH launcher will be needed to reach this ambitious target. The compound spectra enabled by the second launcher are also important, providing greater flexibility and control of the deposition profile. Our interim goal for the operational period with one LHCD grill is 50% non-inductive current drive. Both L-mode and H-mode target plasmas will be used. The new capability of the cryopump is needed for density control, which is critical for maintaining good LHCD efficiency and deposition control in a range of confinement regimes.

Optimization of the bootstrap current will be equally important for current profile control, and will require good confinement and control of kinetic profiles. To this end, our studies of internal transport barriers will be continued. In FY06, we will work to optimize and understand in more detail barriers triggered with off-axis ICRH. Starting from the lower density H-modes demonstrated to be attractive target plasmas for LHCD, we will investigate whether lower density double-barrier discharges can be produced by this technique; these could be very attractive for advanced scenarios. In order to gain experience with the challenges of non-inductive operation, near-term experiments at lower current will be conducted, in collaboration with researchers at GA and PPPL, using two-frequency ICRH with the aim of achieving up to 100% non-inductive (bootstrap) current drive.

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[3] P.T. Bonoli *et al.,* Plasma Physics and Controlled Fusion **39**, 223 (1997).

In FY07 and 08, taking advantage of the LHCD and MCCD capability, the ITB research emphasis will shift towards investigations of the effects of current profile and local shear on barrier formation, location and control. It is anticipated based on results on other experiments that once shear is reduced or reversed via LHCD, it will be possible to form ITBs at a larger radius and with higher on-axis power. This would give a substantial increase in the bootstrap fraction. ACCOME and TRANSP simulations also show an important synergy between bootstrap current and LHCD; when LHCD is applied the bootstrap fraction increases due to the weaker poloidal field near the axis.

The complexity and number of control tools and non-linear interactions between RF and transport physics make scenario modeling a key part of the advanced scenario development program, used for both planning and interpreting experiments. As discussed above, our time-dependent scenario modeling effort will be expanded through multi-institutional collaborations.

As the plasma β increases due to increased input power and improved confinement, issues of MHD stability will become more important. In the FY07-FY08 period, our plan is to modify shaping and plasma profiles so as to maximize the no-wall ideal stability limit. Stability calculations show that $\beta_n \sim 3$ is achievable. In parallel, we will, in collaboration with Columbia University, begin numerical studies of possible methods to stabilize resistive wall modes and exceed the no-wall limit.

Increased power, heat loads and pulse lengths will also increase the challenge of power handling on the metal walls and divertor of C-Mod. As discussed above, FY05 long pulse (3-4 second) discharges with higher RF power and using improved diagnostics were encouraging. Compatibility of high heat loads with the lower densities needed for optimal non-inductive current drive will be a particular challenge for steady-state scenarios through FY08; our results will be important in planning advanced scenarios for ITER. They will also determine the need for outer divertor upgrades on C-Mod, which are anticipated in the FY08 time frame to enable full power, five second operation.

Contributions to ITPA High Priority Tasks and to FESAC Fusion Science Campaigns

C-Mod experiments are uniquely ITER-relevant in many respects, including:

- an all-metal wall,
- high heat loading,
- equilibrated ions and electrons,
- no external momentum or internal particle sources,
- an opaque divertor and
- high magnetic field.

Furthermore, C-Mod is the only world experiment that can test LHCD in these conditions. Research aimed towards demonstrating integrated scenarios thus contributes strongly to a range of ITER needs, as identified by the 2005-2006 list of *ITPA High Priority Research Tasks* recently compiled by the ITPA Coordinating Committee. A summary of the most important contributions expected in the time frame of this proposal is given below, by Topical group. Some of these have been described in more detail in preceding sections but are included here for completeness; the key challenge is to *simultaneously* address the many physics and technology issues which will be encountered on ITER. Many of these issues will be addressed partly through participation in ITPA joint experiments, as listed in Section 1.

MHD:

- Investigate underlying NTM physics including seeding.
- Construct new **disruption database including conventional and advanced scenarios** and heat loads on wall/targets.
- Develop **disruption mitigation techniques**, particularly by noble gas injection.
- Understand **intermediate-n Alfvén eigenmodes**; losses of fast particles from AEs; and perform theory-data comparisons on damping and stability.

Steady State Operation:

- Focus the modelling activity on **ITER Hybrid and Steady state cases**, using standard (and common) sets of input data.
- Assessment of **real-time control of advanced scenarios in ITER**, with collaboration on experiments and modelling.

Transport Physics:

- Understand and optimize **transport properties of hybrid and steady-state demonstration discharges**.
- Address **reactor relevant conditions and dimensionless parameters e.g., electron heating, T_e~T_i, low momentum input, He/impurity transport, edge-core interaction**.
- Utilize international experimental databases in order to test commonality of **transport physics in hybrid, steady state scenario and reactor relevant conditions**.

Confinement database and modeling:

- Resolve the differences in β scaling in H-mode confinement
- Define a program to **understand the density peaking**
- Develop a **reference set of ITER scenarios for standard H-mode, steady-state, and hybrid operation** and submit cases from various transport code simulations to the Profile DB
- Resolve which is the most significant confinement parameter, ν^* or n/n_G

Pedestal and Edge:

- **Improve predictive capability of pedestal structure** through profile modeling of joint experimental comparisons
  - **Dimensionless cross machine comparisons** to isolate physical processes; rotation, E_r, shape, etc
  - Measurement and modeling of inter-ELM transport
  - Establish profile database for modeling joint experiments
- Physics based empirical scaling

    - Collaboration with CDBM to improve scalar database characteristics and utilization
- Predict ELM characteristics and **develop small ELM and quiescent H-mode regimes** and ELM control techniques

Divertor and SOL:

- Understand the effect of ELMs/disruptions on divertor and first wall structures.
- **Improve understanding of tritium retention** & the processes that determine it and **development of efficient T removal methods**.
- Improve **understanding of SOL plasma interaction with the main chamber**.
- Develop **improved prescription of SOL perpendicular transport coefficients** and boundary conditions for input to BPX modeling.

While C-Mod research on integrated scenarios is targeted particularly towards ITER needs, it should be noted that it also makes strong general contributions to fusion science. Many issues at the forefront of fusion research are those involving interaction of multiple topical areas. This is highlighted by the *Recommended areas of US 'opportunity for enhanced progress'* in the 2005 FESAC report, which identified 12 top priorities for fusion program resources. Of these, the Integrated Scenarios thrusts contribute most directly to

> *Integrated understanding of plasma self-organization and external control, enabling high-pressure sustained plasmas,*

which in fact summarizes well the overall goal of the C-Mod scenarios research.

Other FESAC priority areas in which the integration research makes strong and unique contributions are:

- *Predict the formation, structure, and transient evolution of edge transport barriers.*
- *Extend understanding and capability to control and manipulate plasmas with external waves.*
- *Resolve the key plasma-material interactions, which govern material selection and tritium retention for high-power fusion experiments.*

2.6 Run Planning

The C-Mod Research Program is coordinated by the Experimental Program Committee, which consists of the leaders of the Topical Science and Operations Groups and the Thrusts and Task Forces, representatives of the major collaborating institutions and the C-Mod Project leadership. The EPC sets overall research priorities, determines the run-time schedule and allocations, and reviews and approves MiniProposals for individual experiments.

Planning of experimental campaigns at Alcator C-Mod is a multi-step process which draws on input from the entire community. The C-Mod Ideas Forum provides a mechanism by which research proposals are solicited from the entire community. These Forums are typically held at intervals of 20 to 30 Research Weeks, usually keyed to the availability of new C-Mod capabilities. The last Ideas Forum was held in September, 2003, and the next is planned for Summer of 2006, following completion of the current run campaign. Prospective resources and program priorities are advertised prior to the Forum. Research ideas are presented in a brief, conceptual format in open sessions, which are available remotely using a variety of videoconferencing tools. Follow-up sessions for each of the Task Forces and Topical Groups, including all interested participants, discuss and refine the ideas, and prioritize the proposed experiments.

An initial allocation of run-time among the various Topical Groups, Thrusts and Task Forces is established prior to the beginning of each experimental campaign, based on programmatic requirements, facility capabilities, and available resources. The allocations may be revised as the campaign proceeds as priorities and capabilities change, or in recognition of new opportunities.

Approval of specific experiments is based on submission of detailed Miniproposals, which describe the purpose of the research and the plans for carrying it out. Miniproposals are intended to cover well-defined topics requiring one or several days of dedicated experimental time. These are usually submitted under the auspices of one (or more) of the Research Groups or Task Forces. The EPC meets periodically to review and approve MP's. Criteria for acceptance include programmatic relevance, technical feasibility, and scientific importance. Preference is also given to proposals which contribute to students' thesis research.

Scheduling of experimental time for approved Miniproposals is generally done on a short term basis, typically weekly. This approach has proven most successful in terms of providing optimum use of facility capabilities and conditions as they evolve during a campaign, as well as providing flexibility to accommodate unanticipated research opportunities. An exception is made in the case of experiments involving off-site collaborators who require travel arrangements to be made well in advance, and for other time-sensitive experiments; in these cases an experimental schedule may be established well in advance. Similarly, some experiments which depend on specific facility configurations, such as those requiring a particular ICRF frequency or operation with reversed field, will normally be assigned a block of time during which the appropriate conditions are to be provided.

The FY06 Experimental Campaign (now underway) is planned to include 14 Research Weeks (76 Research Days). The total run time is determined, primarily based on budgetary considerations, in consultation with the funding agency. Four research weeks were accomplished in Nov-Dec 2005. The effort during this brief "mini-campaign" was largely devoted to technology issues, especially evaluation of the operation of a screenless ICRF antenna and continuation of wall-conditioning studies begun during the FY05 Campaign. A breakdown of the sixteen research days among the different topical areas is shown in Table 1.

Table 1

ICRF (screenless antenna evaluation)	6.5 days
Wall conditioning	3
MHD	1
Transport/Diagnostics	3.5
SOL / Divertor	2

Initial allocations for the remaining ten weeks of the campaign are shown in Table 2. The allocation to "Lower Hybrid" reflects the level of effort required to commission the launcher, which was re-installed in before the resumption of experiments in February.

Table 2

Transport Physics	8 days
Edge/Divertor/Wall Studies	7
ICRF Physics and Technology	4
MHD	7
Lower Hybrid	7
Integrated Scenario Studies	7

The proposed run plan includes a number of multi-machine "Joint Experiments", including ITPA activities, Internal Transport Barrier (ITB) experiments featuring heating with on- and off-axis ICRF resonances to control the barrier formation, ICRF physics studies including low-single pass D(He^3) minority heating and mode conversion current drive (MCCD). The allocation also includes Integrated Scenario experiments in support of ITER. The allocations shown do not include 8-12 days required for startup and machine conditioning activities, which normally occupy the initial phase of each campaign.

The proposed ten-week campaign requires a number of configurational changes which impact scheduling. Two changes of frequency of the tunable ICRF transmitters #1 and 2 (J-port antenna) are proposed. The initial part of the campaign will be conducted with these transmitters at 78MHz; this configuration provides the maximum available on-axis heating power, with the 4MW from J-port combined with 2MW each from D- and E-port antennas, which are powered by fixed frequency transmitters at 80 and 80.5 MHz . During this phase, at least one week would be carried out with the toroidal field and plasma current direction reversed, a change which requires a few days to accomplish. The J-port ICRF frequency would then be changed to 50MHz in order to carry out elements of the RF physics research program. The frequency would then be changed again, to

70MHz, in support of ITB physics experiments. Each change of the ICRF frequency normally requires about three days for changes to the transmission line hardware and retuning of the transmitters.

3. Operations

3.1 Facilities

The FY2005 run campaign concluded in September 2005, after 18.4 weeks of research operation, which exceeded the 17 week JOULE milestone. 2088 discharges were produced during this run period.

Highlights of machine operation during the last year included a world record volume averaged plasma pressure of 1.8 atmospheres, a C-Mod record stored energy of 250 kJ, and commissioning of the lower hybrid system.

Contribution to ITER Technology

C-Mod is very similar to ITER in terms of magnetic field, density, and heat flux to the divertor. A wide range of ITER technology issues can therefore be studied on C-Mod. In addition, the development of conferencing technology, as well as data acquisition software, will be of great importance to efficient operation of the ITER program. In Table 3.1 we list major contributions we plan to make to the ITER effort.

Cryopump

Density control during H-Mode and AT operation on C-Mod will be greatly improved with the addition of the upper divertor cryopump near the end of FY2006. A liquid helium cooled inconel tube in the upper divertor region of the vacuum vessel will provide the pumping. A liquid nitrogen shield surrounds the helium tube and keeps the helium cooled pump from seeing room temperature surfaces (figure 3.1). A novel feature of the pump will be 30 radial slots in the upper divertor protection hardware that will allow good flow of neutral particles to the pump in a manner insensitive to plasma strike-point location (figure 3.2). Feedlines and baffles needed to contain the pumping volume are shown in figure 3. The effective pumping speed is expected to be >10,000 l/s for deuterium.

All upper divertor protection hardware is in-house and components have been put through a successful in-vessel test fit-up (figure 3.4). The pump will soon be undergoing testing in our cryopump test chamber (figure 3.5 and 3.6). Pump installation will be complete FY2006 with first operation in FY2007.

Advanced Divertor Tiles

During FY2005 the tungsten brush tiles performed well in C-Mod. Now a new more ITER relevant design is being developed in which 4 mm thick tungsten plates are bolted together to form a tile (figure 3.7). This lamella design has been tested to 6.7 MW/m^2 for 7 s pulses (figure 3.8). A toroidal belt of tiles will be installed in the outer divertor during FY2006 with first operation in FY2007.
A picture showing the location of the new tiles and also the tungsten brush tiles currently installed in-vessel can be seen in figure 3.9.

Lower Hybrid System

The lower hybrid system became operational during the FY2005 campaign. During the first three months of the campaign important plasma coupling data was obtained. In May of 2005 an interaction with deuterium gas and/or the plasma with the titanium couplers forced removal of the launcher. It was clear on inspection of the couplers that titanium was not an acceptable material for C-Mod or most probably for fusion grade experiments generally. A review of the use of titanium for invessel ITER components is now in process as a result of the C-Mod experience. As part of a graduate student thesis, we also continue to document and test the titanium used for the couplers in an effort to better understand the failure process.

To replace the titanium couplers, new stainless steel couplers were developed and installed on C-Mod. These new couplers required development of novel brazing techniques to accommodate the large difference in the thermal coefficient of expansion between the alumina vacuum windows and the stainless steel coupler guides. Testing of the new couplers into plasma has just begun.

The last year has also seen extensive development work on protection and phase and amplitude control systems. Fault detection by monitoring the 3rd harmonic power generated by a waveguide arc is a novel technique being developed to protect the C-Mod launcher.

A second lower hybrid launcher will be fabricated starting in FY2007 and continuing into FY2008. The new launcher will reduce the rf power density on both launchers and allow compound spectra to be investigated.

Long Pulse DNB

The long pulse DNB was installed on schedule Feb/Mar 2005 by Budker Institute and MIT personnel. The system has worked reliably with 150 ms pulses into C-Mod at near the design parameters of 50 kV and 7 A. Long pulse operation into C-Mod for 1.5 s with a 100% duty cycle and 3.0 s with 50% modulation requires completion of an interlock system to protect the inner wall. This system integrates signals from an optical pyrometer, the two-color-interferometer, pressure gauges, magnetics, visible bremsstrahlung, and visible light to control beam operation. Long pulse operation of the beam will begin in FY2006.

ICRF Systems

Experiments have been done to better determine the need for a Faraday screen on ICRF antennas. The cost of a new antenna could be reduced if we find that the Faraday screen is not required. The complexity of the antenna would also be reduced, possibly adding to the reliability of the antenna.

As a test, the J-Port antenna Faraday screen was removed, and the antenna was retrofitted with a septum between each strap. From a voltage and power handling standpoint, the design operated well, but local impurity generation was sufficient to degrade the overall plasma performance, particularly during H-Mode, so the Faraday screen was re-installed in a short in-vessel entry in January 2006. Follow-up tests with a modified antenna strap geometry that could have much lower impurity generation characteristics are being considered.

A new 4-strap ICRF antenna, with a Faraday screen, is planned for installation in FY2008. This new antenna will replace both the D- and E-Port antennas without losing any ICRF power handling capability.

Ferrite stub tuners fabricated by Advanced Ferrite Technologies, and originally used at DIII-D and ASDEX-U, will allow a match between the transmitter and the antenna to be maintained in real-time. The original power supplies shipped with the tuners were found to be inadequate for use on C-Mod, but a new vendor for acceptable supplies has been identified. The new vendor has agreed to loan us a supply so that a more informed decision on supply characteristics necessary to achieve the desired system response can be made. An RF power test of the tuners and supply is planned in FY2006 and installation and operation in the E-Port antenna matching network is planned for FY2007. Operation with these tuners will yield valuable information regarding the design of new tuners for the two 4-strap antennas we plan to have in place by FY2008. Incremental funding will allow the design, procurement, and installation of these tuners for FY2008.

To determine how best to improve ICRF availability, all the ICRF systems have been carefully examined for reliability. The ignitron based protection crowbar system for the final and driver amplifiers have been identified as the major system most degrading availability. The crowbar, using very dated technology, is a very complex system that is subject to numerous fault conditions. A replacement IGBT based device that would greatly simplify the tube protection circuit and reduce high fault current stress on the transformer-rectifier supply has been identified. Another ICRF system upgrade would modify the screen and grid power supplies and fast opening switch. This upgrade would allow high power operation over a wider range of plasma loads. Finally, the transformer-rectifier set is insufficient to deliver the necessary current to deliver the rated power of the final amplifier tube. Upgrades to the transformer-rectifier system are being considered. Given tight budgets and physics priorities, these upgrades can only be accomplished with incremental funds.

MIT Alternator

The alternator ran very reliably over the past year. A new monitor to measure oil content in the oil cooling water was recently installed and has added to the environmental safety of the alternator system. As part of the continuing preventative maintenance program for the alternator, a Hi-Pot of the stator will be performed at the end of the FY2006 run campaign. A full alternator inspection has been scheduled for FY2008.

Data system

Approximately 1.8 GBytes of data is taken during each shot cycle on Alcator C-Mod with our data handling requirements doubling every 2.15 years (figure 3.10). To keep up with this demand we have increased our ISCSI raid array storage from 6 to 11 TB. We have deployed a LT03 based tape library for archive and backup. Six new data acquisition infrastructure servers have been brought online. We have continued to migrate from CAMAC to CPCI based data acquisition hardware. This level of hardware upgrades will have to continue during FY2007 and FY2008 to maintain our ability to access all C-Mod data quickly, and to not allow the data acquisition process to impact the machine shot cycle.

To support better access to data and improve record keeping, a web based logbook interface was implemented that has now become our standard logging tool. Development of web based user, network, and hardware administration and record keeping tools will continue during FY2007 and FY2008.

Both on- and off-site support of MDSPlus will also continue over the next two years. MDSPlus is now used at more than thirty sites throughout the world.

We have continued to expand our use of the new Digital Plasma Control System (DPCS). This system allows much more complicated control algorithms to be implemented than were possible with the Hybrid system. In addition, the number of inputs increased from 96 to 128, and the number of outputs doubled from 16 to 32. Over the next two years we will replace linear observers with improved non-linear algorithms, extend the range of controlled parameters, and move toward real-time MHD equilibrium modeling and control. We have recently implemented an algorithm to ramp down the power supplies in a more controlled fashion following disruptions or when the plasma ramp-up is not successful, which has reduced stresses on key machine components. Control of ICRF power using the neutron rate as a control parameter is also planned for near term implementation.

Outer Divertor Upgrade

The heat load to the outer divertor will continue to increase as we add more lower hybrid rf power and extend discharge pulse lengths. A conceptual design for a new outer divertor with an extended vertical cylindrical section has been developed to handle these increased heat loads. The new design will be easier to manufacture than the current divertor, and also be much easier to install with precision in-vessel. Alignment of the tiles relative to the field lines is a critical issue to be addressed if we are to minimize peak heat loads on the tiles. There will also be no toroidal gaps or leading edges where enhancements to the heat load typically occur. The tiles used for the divertor are expected to follow from our advanced divertor work.

If incremental funding becomes available, a detailed design of the outer divertor is planned for FY2007 with procurement and installation completed in FY2008.

3.2 Diagnostics

Several new diagnostics have been brought on-line over the past year as well as upgrades to existing ones. We show in Table 3.2 a list of diagnostic accomplishments and plans.

During FY2005 important progress was made on several diagnostics. A mask was installed over the phase plate in the PCI diagnostic that allowed the location of the fluctuations to be localized along the beam path through the plasma. The k-space range of PCI was extended to 25 cm^{-1} with plans to push to 60 cm^{-1}. The Neon Soft X-Ray spectrometer (NeSOX) has been installed and is making measurements of impurity rotation near the plasma edge. A great deal work has been done in a effort to understand the MSE data, and the new long pulse DNB will allow new MSE calibration techniques to be developed. The polarimeter/interferometer 10.6 μm single chord prototype diagnostic made very good plasma density measurements using retro-reflectors on the inner wall. This poloidally viewing polarimetry geometry is planned for use on ITER. The CXRS systems also provided very useful data over the past year with ion temperature and plasma rotation profiles being provided for the first time.

Table 3.1

C-Mod Contributions to ITER Technology

Tungsten Lamella Tiles	Toroidal belt of tiles will be installed during Summer 2006 up-to-air Design similar to those being considered for ITER
Real-Time ICRF Fast Ferrite Tuners	Maintain antenna match during ELMs and across confinement mode transitions
Wall Conditioning and Coating Technology	C-Mod results indicate low-Z coatings are required for good performance in machines with high-Z PFCs Techniques need to be developed to continuously maintain low-Z layer
Dust Detection	Dust may become a major safety issue for ITER Dust may also provide a source of impurities Scattering system to detect dust being developed for C-Mod. Video techniques to view dust also being developed.
Polarimetry	FIR poloidally viewing polarimeter being developed for C-Mod Geometry similar to that proposed for ITER Fields and plasma parameters similar to ITER

Gas Jet Disruption Mitigation	Disruptions in ITER could be very damaging events Techniques are needed to detect and protect the machine from disruptions A high pressure gas jet is now operational on C-Mod and has already greatly influenced our understanding of how these systems affect disruptions Experiments will continue during FY2006-8
Remote Conferencing Technologies for ITER	Off-site session leaders routinely use audio, video, and MDS-Plus tools to communicate with on-site personnel and direct C-Mod run We are working directly with US and international teams to define remote conferencing technologies for ITER
CODAC Definition and Design for ITER	We have begun to work with the ITER team to help define CODAC requirements and a conceptual design for ITER
National Fusion Collaboratory	Effort aimed at developing collaborative technologies for current research programs, but with ITER in mind

Table 3.2

Diagnostic Development

PCI Diagnostic	Vertical localization capability has been demonstrated using a phase plate masking technique ITG-like turbulence has been measured High-k measurements have been tested up to 25 cm^{-1} New phase plates will extend high-k range up to 60 cm^{-1} Scanner will provide vertical scans at a 100 scans/s rate
Bolometry	Midplane foil array increased from 16 to 24 channels for 2005 campaign Midplane 22 channel AXUV diode installed to allow better separation of impurity species ($r/a > 0.36$) in FY2007 2D toroidally viewing pinhole camera will be installed to investigate polodial asymmetries in radiated power and x-ray emissivity in FY2007
Neon Soft X-Ray Spectrometer	New diagnostic provides impurity poloidal rotation profile in plasma edge Measures line emission from He-like neon Purely poloidal view ~20 cm below midplane
X-Ray Imaging Crystal Spectrometer (PPPL)	2D position-sensitive detector Will measure $T_i(r)$ and $V_{\square}(r)$ with 1 cm spatial resolution Initial tests of diagnostic FY2007
Correlation Reflectometer (PPPL/MIT)	Install 130-140 GHz swept frequency Gunn diode for density correlation. Operation FY2007 Install and operate ITER prototype 180 GHz, X-mode system with operation in FY2008.
MSE (PPPL/MIT)	APD detectors will be installed to replace PMTs with degraded sensitivity for FY2007 Long pulse DNB will enable new calibration technique that sweeps plasma edge past each MSE channel in FY2006/FY2007

CXRS Diagnostics (Utexas/MIT)	Passive measurements of B^{+4} have yielded T_i, V□, and V□ for 0.85 < r/a < 0.9 Active measurements of B^{+5} have yielded poloidal rotation and ion temperature profiles. 22 each, core and background toroidal chords, 0.72 m < R < 0.88 m, < 10 mm resolution, will be added for FY2007 25 outer edge toroidal chords 0.86 m < R < 0.90 m, < 3.5 mm resolution, will be added for FY2007
Prototype Polarimeter	FIR polarimeter/interferometer system will complement the MSE effort A single chord prototype system at 10.6 □m has been installed to test edge scrape-off effects, determine noise levels, refine experimental techniques, and assess survivability of in-vessel optical components New laser, shutter system, and improved retro-reflectors will be installed for FY2007 operation Procurement of FIR laser and detectors underway Multi-chord FIR system available FY2008
Core Thomson Scattering	In FY2006 increased number of core channels from 11 to 16 – enhanced resolution in vicinity of ITB New compact polychromators (8 new channels in FY2007) New fiber bundles with improved transmission in near-IR FY2007 New fast charge digitizers FY2007
Dust Detection (ITPA high priority)	Developing laser scattering techniques to measure dust for FY2007 Automation of visible and IR camera image analysis to search for dust will be further developed in FY2006 and FY2007
New Penning gauge installation	Gauges can be easily mounted on vessel wall Quantify pumping/cryopump operation Install FY2006 with FY2007 operation
Advanced Inner Wall Scanning Probes	Measure plasma flows, fluctuations, and profiles in the high-field side scrape-off layer. Installation in FY2006, operation in FY2007.

Quartz Microbalances	Measure deposition and erosion Quantify boronization layers Install FY2006 with FY2007 operation
Compact Neutral Particle Analyzer (CNPA)	Si detectors measure energetic ion tail (50 to 350 keV) Passive and active detection of neutrals demonstrated (FY05)
Hard X-Ray Pinhole Camera (HXR)	Poloidally viewing 32 channel CZT detector array (20 to 250 keV) (operational from FY06) Measure supra-thermal electron distribution during lower hybrid current drive experiments FY2006/FY2007
Gas Puff Imaging in Lower Divertor (PPPL/MIT)	New GPI view in divertor will allow comparisons to be made with transport effects seen near plasma midplane FY2007
New Magnetic Pick-up Coils	Six new coils to improve low toroidal mode number resolution of poloidal field measurements FY2007
Non-Thermal ECE	Essential tool to understand electron distribution during lower hybrid current drive experiments CQLD modeling of present and possible new views continuing FY2006 Design and implement optimized system in FY2007 with operation in FY2008

Figure 3.2 Cross-section of cryopump is shown. Liquid helium flows in central tube and is surrounded by a liquid nitrogen cooled shield. Baffle (green) protects nitrogen shield from warm neutral particles. Protection support hardware and tiles also shown.

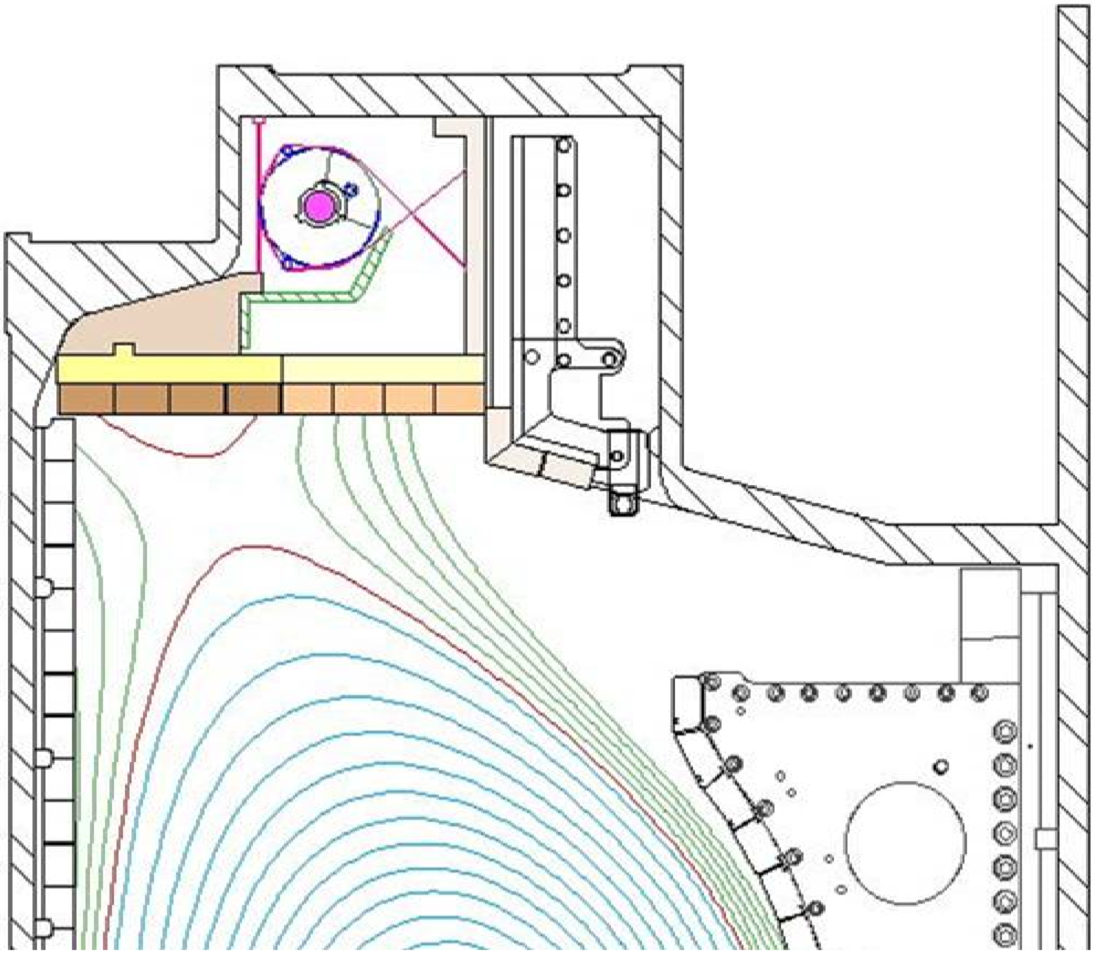


Figure 3.3 Thirty radial slots in the protection hardware allow neutral particles to enter cryopump pumping volume. This configuration allows the pumping rates to be insensitive to the plasma configuration.

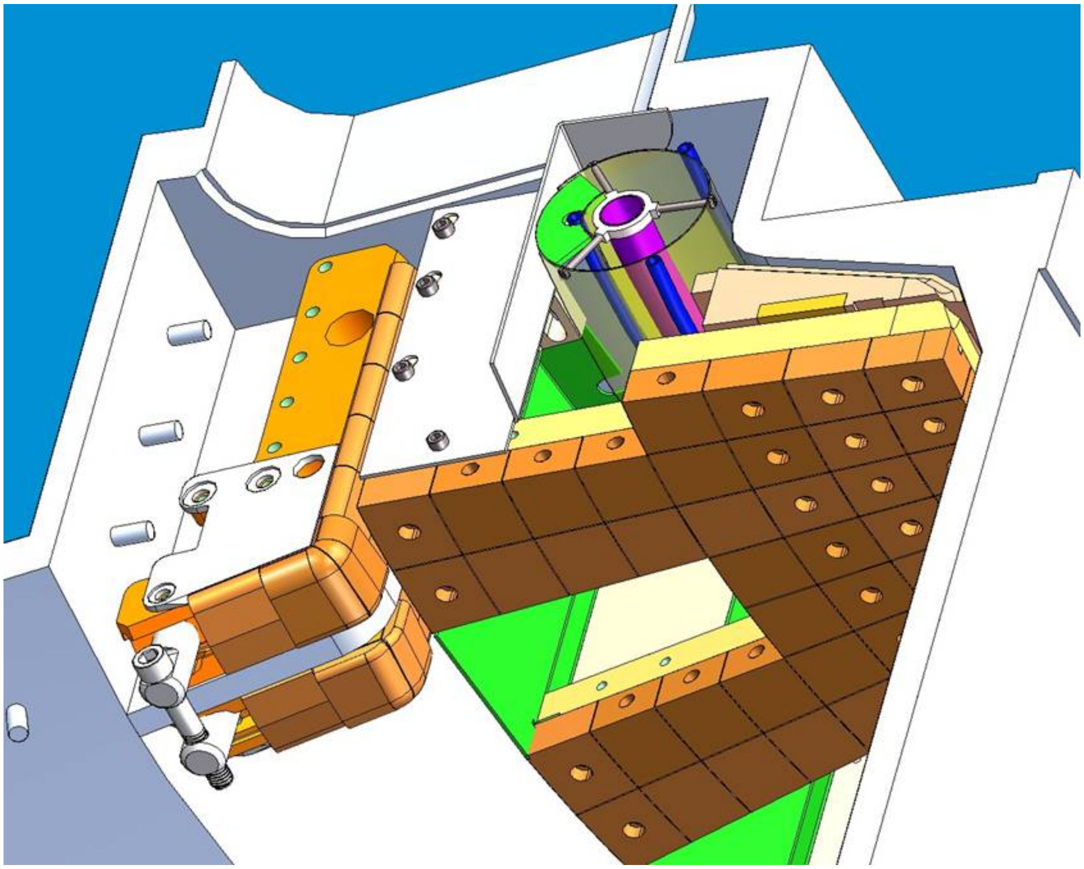


Figure 3.4 Liquid helium and liquid nitrogen feedlines are shown as are support plate hardware. Baffles (shown in green) define pumping volume.

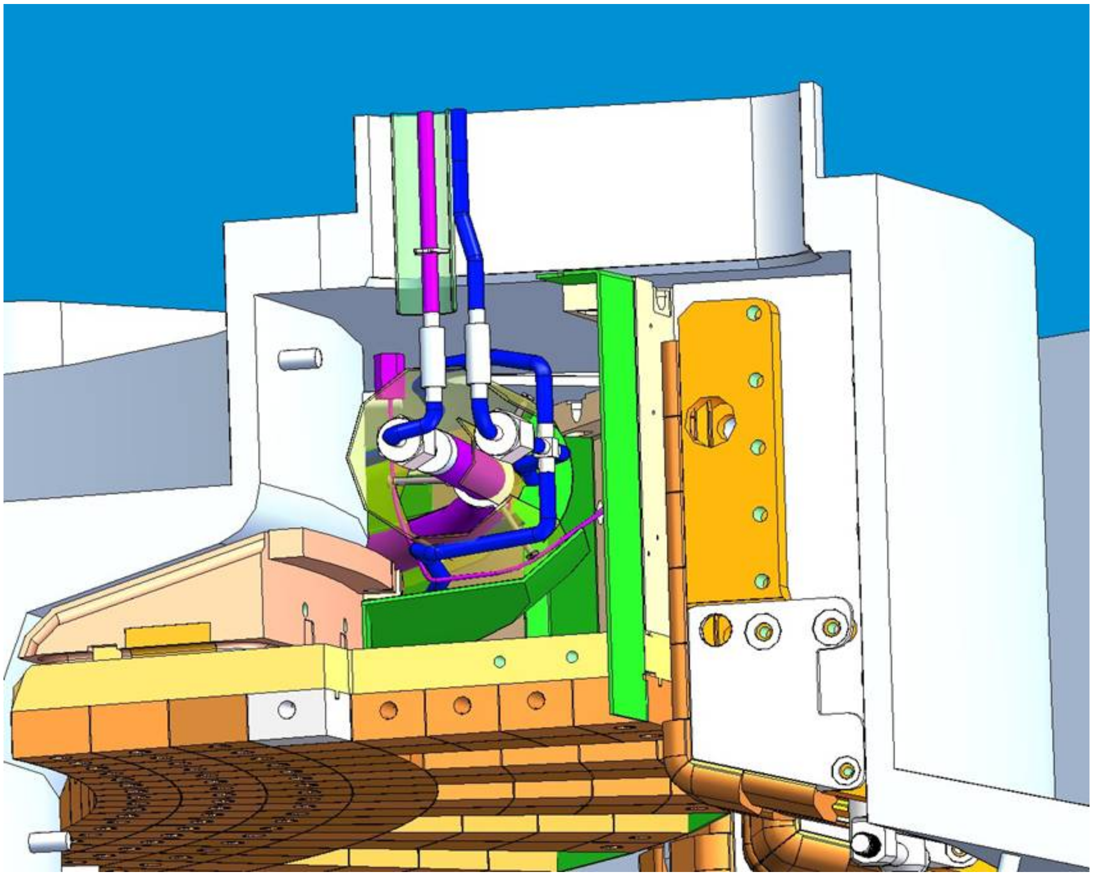


Figure 3.5 All protection hardware is in-house and components have been successfully fit-up in-vessel.

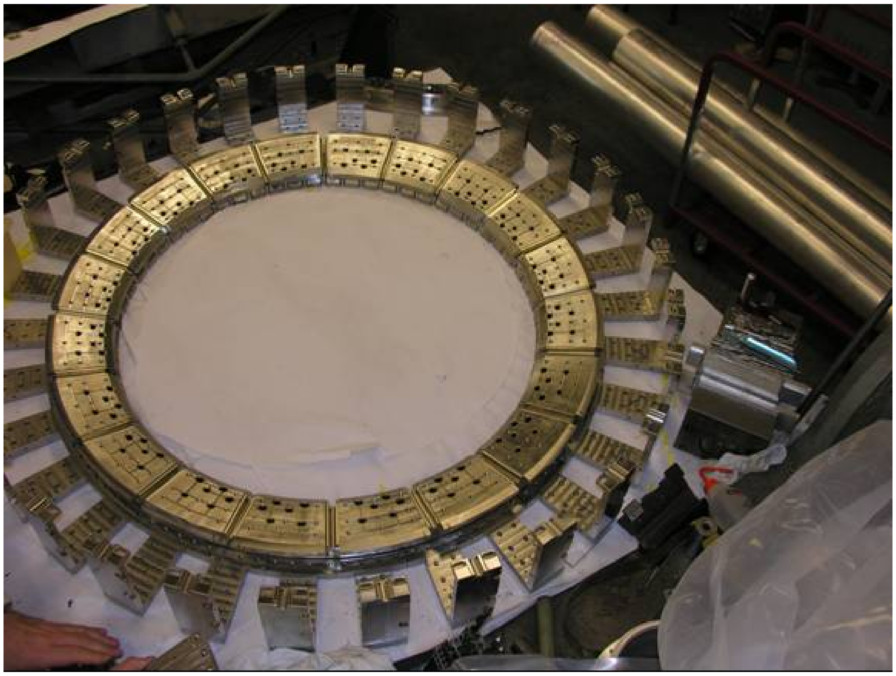


Figure 3.6 The cryopump test stand will mock up pumping geometry expected in C-Mod. The pump control system, pumping rates, and maximum gas inventory will be tested on the stand before installation on C-Mod.

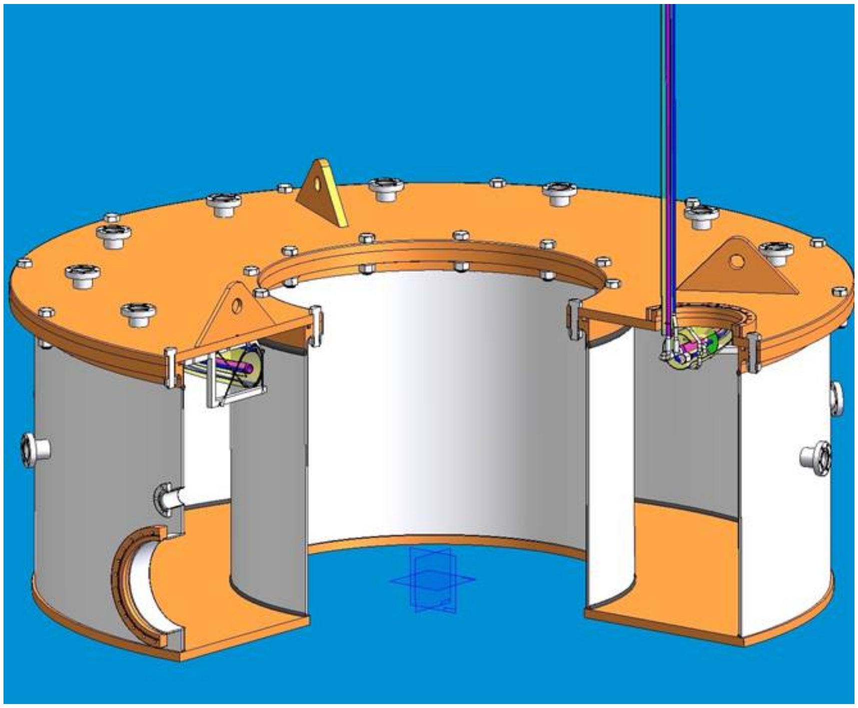


Figure 3.7 Cryopump test stand vacuum chamber and pumping station

Figure 3.7 W-tile lamella design is shown. Eight tungsten plates are tied together with a single TZM bolt. Threaded hole in bolt provides anchor to tile support plate and allows the new tile module to directly replace the moly tile.

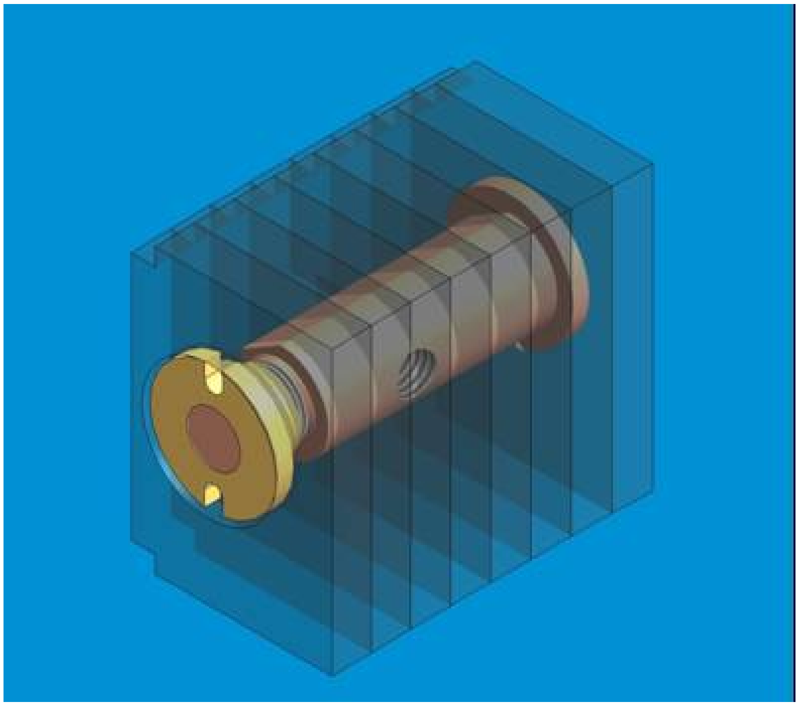


Figure 3.8 Two lamella W-tiles ready for high power testing. Power levels of 6.7 MW/m^2 for 7 s were applied to the tiles without damage. Thermocouple leads are also shown.

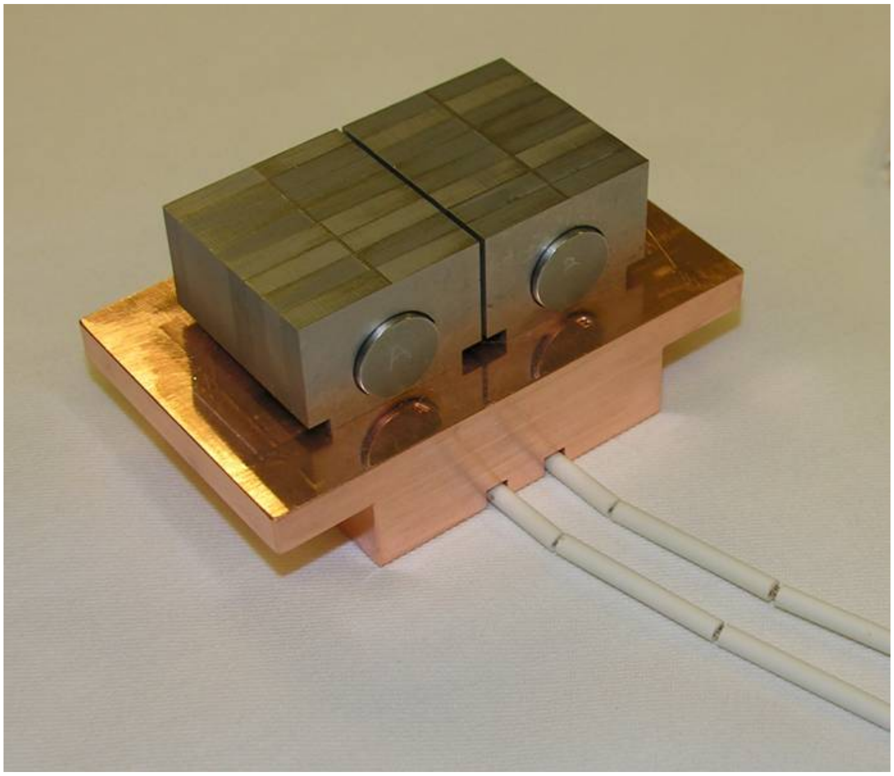


Figure 3.9 The new lamella W-tiles will form a toroidal belt just below the "nose" tiles of the outer divertor. Some of the W-brush tiles successfully tested in the FY2005 campaign can be seen near the middle of the picture.

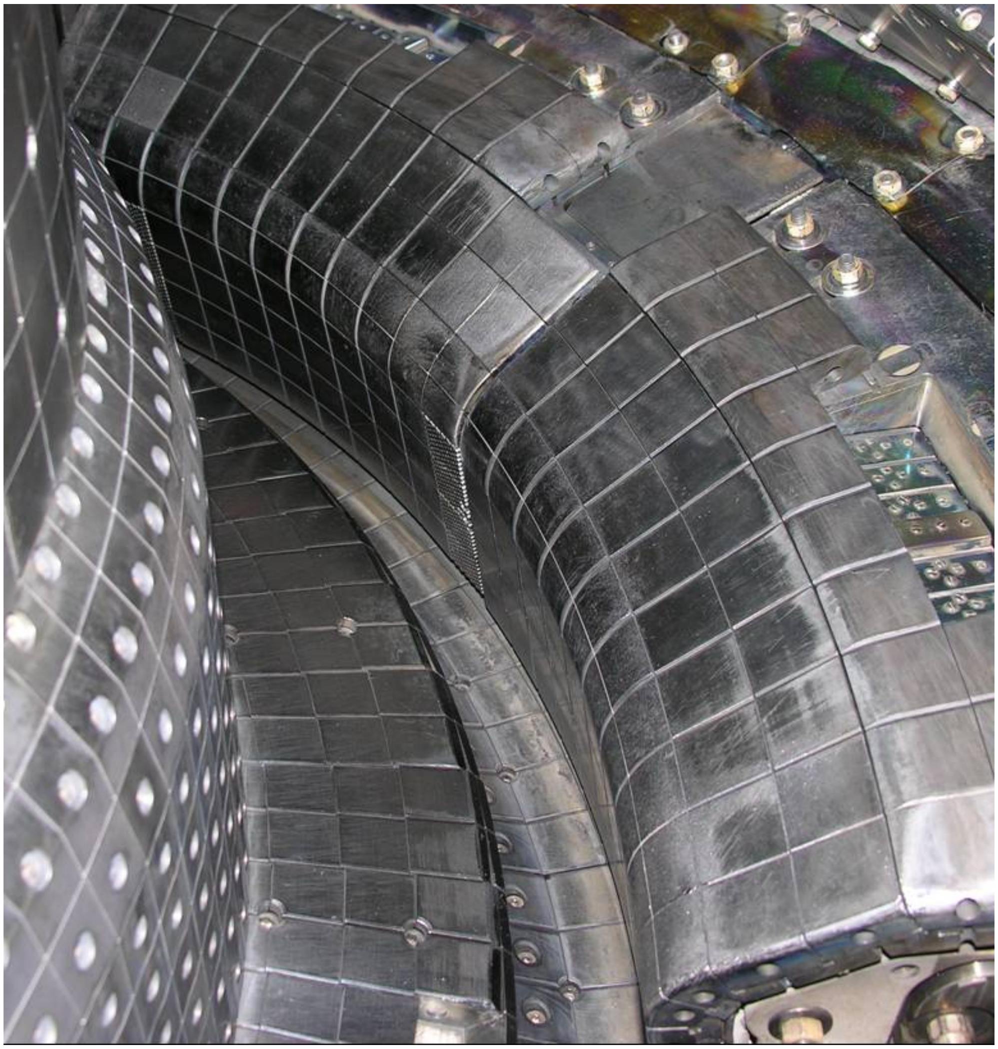
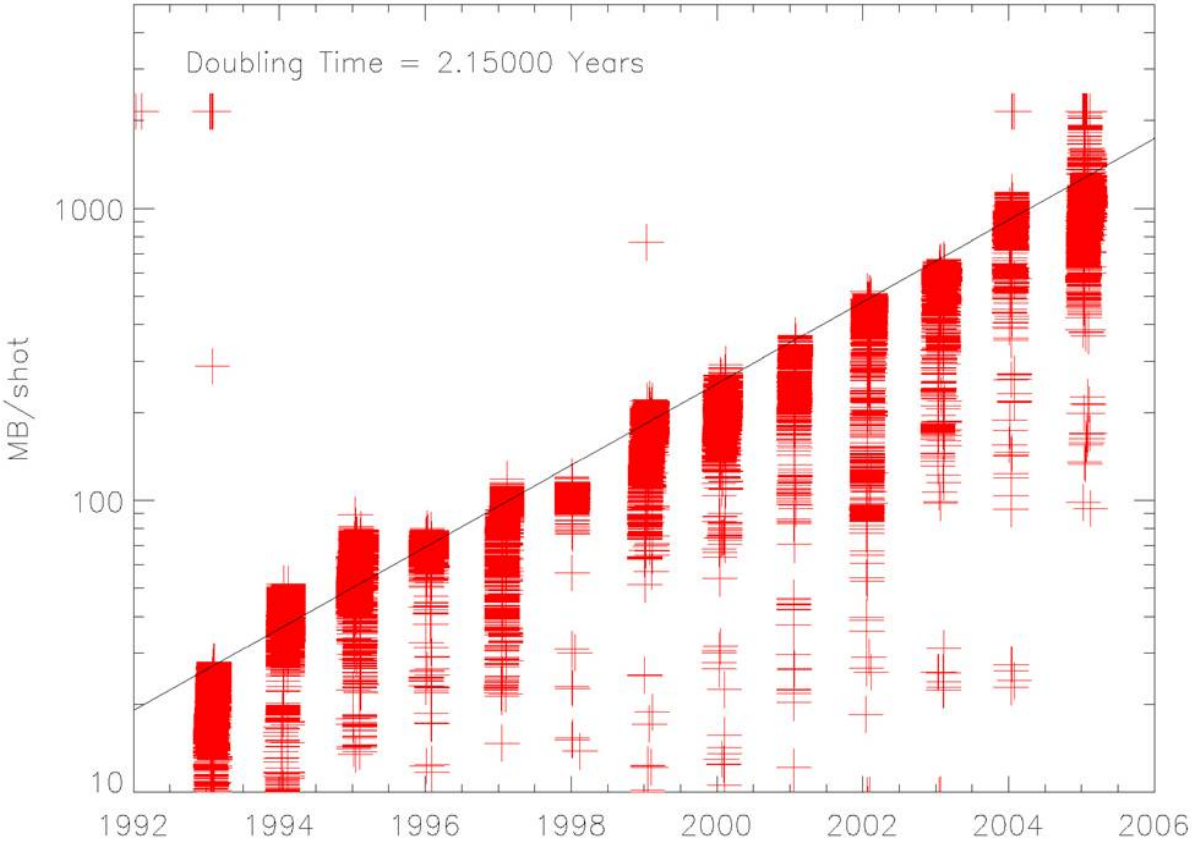


Figure 3.10 The amount of data taken per shot is plotted since Alcator C-Mod began operation. A data doubling rate of somewhat more than 2 years is indicated.



4. Alcator C-Mod Collaborations

4.1 PPPL Collaboration:

Purpose:

The purpose of the PPPL C-Mod collaboration is to conduct and enable forefront scientific research on the Alcator C-Mod tokamak and to perform engineering/technical support for the C-Mod team.

Research aims

- Determine the effectiveness of plasma current and pressure profile modifications to achieve enhanced plasma performance in the Advanced Tokamak regime through off-axis current drive via Lower Hybrid current drive and high power ICRF heating and current drive;
- The experimental study of basic ICRF plasma-wave interaction processes and their comparison with theory to gain predictive capability for heating and current drive in reactor-grade experiments;
- Creation and understanding of internal transport barriers through off-axis ICRF heating;
- The study of plasma turbulence and its effect on core confinement; and
- Plasma edge turbulence visualization and determination of its effect on transport.

Hardware upgrades (recent and proposed)

- Ongoing improvements to the 4-strap ICRF antenna for plasma heating and current drive;
- A LHCD launcher and coupling hardware for control of the plasma current profile through current drive;
- Current profile diagnostics to increase understanding of current drive and plasma behavior;
- Further improvements to edge diagnostics (edge fluctuation measurements at the plasma periphery with reflectometry and 2-D imaging of edge turbulence) to increase understanding of turbulence and transport in the scrape-off region and the pedestal.

- Spectroscopic diagnostics (curved x-ray crystal spectrometer) to measure the radial profiles of ion temperature and toroidal rotation speed with high spatial resolution.

Engineering and technical support for RF power systems include:

- Engineering assistance in tuning and maintaining the ICRF transmitters, and

- Engineering participation in the design, fabrication, and installation of the Lower Hybrid launcher as part of the current-drive system

In these scientific and technical areas PPPL provides assistance in areas where PPPL has competence and capabilities needed by the C-Mod program while enhancing cross-cutting research opportunities for PPPL scientists.

Approach:

Members of the PPPL research staff participate in experiments on C-Mod at MIT as integrated members of the C-Mod research and operations team. These scientists are supported by core teams at the laboratory for theoretical support, data analysis and modeling, and for coordination with other PPPL research endeavors through the PPPL science focus groups. In addition, PPPL provides a team of engineers and technicians for the design and construction of upgrades, and for technical support at C-Mod.

Wave-Particle Studies

The interaction of radio-frequency waves with the plasma components can result both in localized plasma heating and the generation of a locally driven current. Studies and understanding of the basic physics processes will allow extrapolation of these results into the reactor-grade plasma regime. Numerical simulation of current drive by Lower Hybrid waves will be extended to include effects from non-Maxwellian electrons.

Radio-frequency heating studies in the ion cyclotron range of frequencies will investigate various aspects of heating mechanisms:

- Compare the heating efficiency of strong single-pass absorption heating, hydrogen minority ion species in a deuterium majority D(H) with weak single-pass absorption in helium-3 minority in a deuterium majority D(3He); and

- Investigate the rich spectrum of phenomena associated with fast wave mode conversion.

Launching an ICRF directed wave allows us to drive plasma current in the core by fast wave current drive (FWCD):

- Explore further plasma rotation with directed waves without external momentum input;

- Develop the capabilities to affect the radial electric field through toroidal rotation.

Transport Studies

Our studies of plasma transport are focused on optimizing the experimental design to yield measurements that can be compared with nonlinear gyrokinetic computational turbulence simulations to gain insight into the plasma microturbulence:

- Tests of marginal stability predictions involve the comparison of data from C-Mod fluctuation diagnostics with nonlinear gyrokinetic simulations using the codes GS2 and GYRO, Quantitative interpretation of the experiment is expected to require simulations of the reflectometers used to measure the turbulent fluctuations.

- A technique for measuring small changes in the electron temperature gradient scale length will be used to compare experimental and theoretical dependencies of the electron temperature gradient on parameters such as the q profile.

- A new technique has been proposed to delineate which plasma parameters are involved in the formation of internal transport barriers.

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Plasma Boundary Studies

The study of plasma edge physics has been enhanced through the addition of a new fast camera to obtain 2-D imaging of edge turbulence:

- 300 frame "movie" images of edge turbulence structures ("blobs") are obtained with 4μs exposure, giving growth/decay and radial/poloidal motion information, and

- This behavior is being compared with a variety of edge turbulence models.

Integrated Scenarios research for ITER

We plan to study Advanced Tokamak plasma regimes by modifying the plasma current profile with off-axis Lower Hybrid current drive and on-axis ICRF fast wave current drive. The plasma pressure profile will be modified through the application of high power ICRF on- or off-axis heating.

The Lower Hybrid power system is based on the 4 MW, 4.6 GHz system originally used on Alcator C. The PPPL Lower Hybrid team designed and fabricated a titanium waveguide launcher and constructed a novel power divider, high power phase control and custom waveguide assembly that provides precise spectral control of the launched wave spectrum (n = 1.5-3.0). The splitting system allows launching waves in both forward and reverse direction.

Modification of the current profile, whether by FWCD, MCCD or LHCD, requires a measurement of the resulting current profile for analysis. This will be achieved through the motional Stark effect (MSE) diagnostic. The optical system, electronics, and software have been supplied by PPPL; the diagnostic neutral beam (DNB) generating the signal is supplied by MIT. Initial measurements of magnetic pitch angle have been made, and extensive in-vessel calibrations and analysis are in progress in order to derive accurate current distributions from them.

In the area of high power plasma heating and current drive in the Ion Cyclotron Range of Frequencies (ICRF) our approach is twofold:

- Use the increased heating power and current-drive capability to expand the C-Mod physics operating regime and enable understanding of a wider range of plasmas, especially relevant to the high-field approach to burning plasmas; and

- Increase the understanding of the physics of ICRF heating and current drive at high power.

FY05 Accomplishments

Wave-Particle Studies

- A dielectric tensor module that is appropriate for nonthermal species in the finite Larmor radius limit was developed for use in the TORIC code. Simulations of the effects of a nonthermal electron "quasilinear plateau" velocity space distribution function on lower hybrid wave propagation and absorption in C-Mod were presented at the 2005 APS-DPP meeting in Denver. [RF SciDAC and C-Mod collaboration]

- The TORIC5 code has been integrated into the TRANSP plasma analysis and simulation code.

Transport Studies

- We have explored the prospects for experiments to probe the species dependence of transport and turbulence. Suitable RF heating scenarios have been identified for plasmas with hydrogen and helium-4 as the majority species; these can be run with the same toroidal field strength as H-minority heating in a D-majority plasma. Using an upper-single-null configuration should produce L-mode plasmas with density profiles that enable core reflectometry to measure the turbulent fluctuations.

Plasma Boundary Studies

- A Princeton Scientific Instruments PSI-5 fast camera with as short as 4 μs exposure time and 300 frame capability has been added, allowing improved edge turbulence visualization to increase our measurement of edge turbulence growth and motion.

- Evaluation of the velocity field of turbulent motion in the edge has been performed, showing dominantly outward radial motion outside the separatrix, and dominant poloidal motion inside the separatrix.

- Comparison of edge turbulence with modeling by the Risø group is in progress.

Integrated Scenarios research for ITER

- In FY2005, the titanium LH launcher was operated for the first time at low to moderate power levels (< 200 kW). Initial studies of LH coupling as a function of phase and density were carried out.

- Despite difficulties with one ICRF antenna arising from titanium dust generated by the first Lower Hybrid launcher, the ICRF systems delivered more than 2 megawatts of power to 780 plasmas during FY05. This performance allowed C-Mod to successfully meet its Joule milestone of characterizing plasma performance with all-metal (molybdenum) walls and the effects of low-Z wall coatings.

PPPL Engineering Support

- PPPL engineers, physicists and technical staff provided active participation in the technical issues involved in the Lower Hybrid launcher repair activities.

- PPPL engineers, physicists and technical staff are participating in the installation and initial checkout of the Lower Hybrid launcher on C-Mod.

- PPPL RF engineers and technical staff continued to assist MIT with ICRF transmitter operation, retuning, and repairs.

Future Accomplishments: FY2006 (baseline)

The repair of the first Lower Hybrid coupler has been completed and it has been reinstalled on C-Mod. We will continue plasma heating and current drive studies at high ICRF power levels and initial Lower Hybrid power, and make initial measurements of changes in the q-profile.

Wave Particle Studies

- Begin simulations of lower hybrid current drive in C-Mod, using the TORIC-LH code that is being extended to include non-Maxwellian electrons (RF SciDAC and C-Mod collaboration).

- Continue study of mode conversion of a launched fast wave into an ion Bernstein wave, an ion cyclotron wave and the associated poloidal flow generation.

- The study of ICRF-induced core-Alfvèn modes will be continued, with ongoing use of the PPPL NOVA-K code and participation of energetic particle physicists. Also begin exploratory studies of driven global radio frequency eigenmodes in tokamaks, using the TORIC code. (RF SciDAC, NSTX and C-Mod collaboration)

<u>Transport Studies</u>

- Two experiments designed to elucidate the role of micro-turbulence have been proposed, and mini-proposals will be prepared. The first will search during the ITB formation phase for small changes in the ratio of T_e from neighboring ECE channels, which is directly related to the electron temperature gradient scale length. This technique should be capable of reliably detecting changes as small as ~5%. The second experiment will form ITBs in lower density plasmas that permit reflectometers to measure density fluctuations in the ITB region; we will look for a change in the reflectometer signal when central heating is added to stop the density rise. Micro-turbulence codes will simulate the plasma before and after the addition of central heating, and the predicted turbulence will be compared with the turbulence measurements.

- Internal transport barrier modeling will focus on linear GS2 stability analysis of C-Mod off-axis RF-generated ITBs, with possible extension to Ohmic H-mode ITBs.

- Perform initial benchmarks of GYRO code against measured profiles and heat fluxes in Ohmic and RF-generated ITB plasmas, and against available fluctuation measurements.

- Test performance of the PILATUS detector for use in a curved- x-ray crystal diagnostic for measuring Ti and V_ϕ.

- Install a replacement 140 GHz fixed-frequency reflectometer to measure electron density fluctuations at $n_e = 2.4 \times 10^{20}$ m^{-3}, corresponding to typical H-mode densities in Alcator C-Mod and densities at the 'foot' of ITB density profiles.

- Install a 132-140 GHz swept-frequency correlation reflectometer to measure the density fluctuation correlation length.

<u>Plasma Boundary Studies</u>

- Improve the design of the optics of the existing GPI telescope at the outer midplane to either get better spatial resolution and/or increased spatial coverage.

- Design an additional lower-divertor GPI view of edge turbulence using a new in-vessel fiber optic bundle.

- Take additional imaging data from the ‘side view’ to look for possible structures in the direction parallel to B (using the existing PSI-4 and or PSI-5 cameras).

- Analyze existing GPI outer midplane data from scans done during 2005 to look for scalings of `blob’ structure and motion with B field, density, etc

Integrated Scenarios research for ITER

- Low power LH testing will commence and the power will be increased to maximum capability through conditioning and operation into plasma.

- Evaluate and optimize the coupling efficiency and power handling capability of the Lower Hybrid launcher.

- Investigate the physics of coupling Lower Hybrid waves to high density plasmas.

- Initiate current drive in the plasma with the Lower Hybrid launcher.

- Evaluate the capabilities of the MSE/DNB diagnostic system including spatial resolution, accuracy of measured q-profiles, and accuracy of measured locally driven currents as a function of plasma conditions.

- Contribute to the MIT design of the interlock system for the long pulse diagnostic neutral beam to protect the inner wall tiles from overheating.

- Contribute to the MIT design of a variable aperture to reduce the width of the diagnostic neutral beam for improved spatial resolution of MSE and BES.

- Extend studies of the Internal Transport Barrier mode to higher levels of heating power, and continue to investigate its suitability as a target plasma for the Advanced Tokamak LHCD experiments.

- Extend the C-Mod plasma’s parameter space by means of the high levels of ICRF heating power including combined ICRF and LH.

- Use the high heating power to investigate divertor and inner wall power handling capability.

- Studies of the Internal Transport Barrier mode will be extended to higher levels of heating power, and its suitability as a target plasma for the Advanced Tokamak LHCD experiments will continue to be investigated.

PPPL Engineering Support

- PPPL RF engineers and technical staff will continue to assist in all phases of the Lower Hybrid system operation.

- PPPL RF engineers and technical staff will continue to assist MIT with ICRF transmitter operation, retuning, and repairs.

- PPPL physicists, RF engineers and technical staff will assist the C-Mod RF group in the design and preparation of a second 4-strap ICRF antenna.

Future Accomplishments: FY2007 (baseline)

Study the processes relevant to the generation of Advanced Tokamak discharges using high power on- and off-axis ICRF heating and LH off-axis current drive.

Wave-Particle Studies

- Continue and extend the study of mode conversion physics, including current drive and flow drive.

- Simulate a range of conventional ICRF heating and lower hybrid current drive scenarios in C-Mod, using the TORIC and TORIC-LH that were generalized in FY2005/2006 to include non-Maxwellian species.

Transport and Fluctuation Studies

- Based on simulations of C-Mod plasmas with LHCD, new experimental scenarios will be designed to improve confinement and the results will be compared with simulation.

- Internal transport barrier modeling will be extended to nonlinear GS2 and GYRO analysis of the off-axis RF-generated ITBs and Ohmic H-mode ITBs.

- Density fluctuation studies will be carried out with the 140 GHz reflectometer channel over a range of plasma conditions.

- Install a new correlation reflectometer to measure the radial correlation length at 132-140 GHz ($n_e \sim 2.3 \times 10^{20}$ m^{-3}).

- Existing diagnostics for toroidal rotation on C-Mod are limited to three x-ray spectrometers that provide a three-point radial profile with a time resolution of tens of milliseconds. The high densities routinely realized in C-Mod H-mode plasmas also preclude the use of charge-exchange recombination diagnostics because the beam penetration is insufficient.

We will build and install a curved x-ray crystal spectrometer looking at the Ar16+ line using a PILATUS detector which removes the count-rate limitation that previously limited the performance of this diagnostic. This will provide measurement of the ion temperature and toroidal velocity profile with significantly improved spatial and temporal resolution (spatial resolution of a centimeter and temporal resolution of a few milliseconds), over an 8-cm vertical view of the plasma. Alcator C-Mod has agreed to purchase a second PILATUS detector that would allow the total viewed region to be extended to 16 cm.

Plasma Boundary Studies

- Purchase an additional fast camera for use in C-Mod, probably one that can record continuously and so capture intermittent events (L-H transition, ELMs, etc.).
- After installation of the new lower-divertor GPI view, compare edge turbulence near the lower divertor with that at the outer midplane.
- Work with theorists to explain existing data and predict what will be seen near the lower-divertor from simulations of edge turbulence in C-Mod.

Integrated Scenarios research for ITER

- Modify the C-Mod plasma current profile with off-axis LHCD.
- Measure the resulting plasma performance changes with the C-Mod diagnostics.
- Compare the plasma behavior with transport and stability models.
- Continue to extend the C-Mod plasma's parameter space by means of the high levels of ICRF heating power.

PPPL Engineering Support

- PPPL RF engineers and technical staff will continue to assist in operation and maintenance of the Lower Hybrid system.
- PPPL RF engineers and technical staff will continue to assist MIT with ICRF transmitter operation, retuning, and repairs.
- PPPL physicists, RF engineers and technical staff will assist the C-Mod RF group in the design and preparation of a second Lower Hybrid launcher.

Future Accomplishments: FY2008 (baseline)

Wave-Particle Studies

- Extend the physics included in algorithms for generating the non-Maxwellian dielectric tensor elements to the hot ion lower hybrid regime and explore effects of non-Maxwellian species on lower hybrid wave propagation and absorption (RF SciDAC and collaboration with C-Mod).

- Continue integrated full wave simulations of ICRF and lower hybrid regime scenarios in support of C-Mod experiments.

Plasma Boundary Studies

- Evaluate 'blob' formation and direction of radial motion in lower divertor region.

- Evaluate the effect of LH heating or off-axis current drive on the edge turbulence.

- Evaluate theoretical models for SOL turbulence in C-Mod.

- Attempt to actively control edge turbulence (possibly w/ lower hybrid, ICRF, edge pellets, biased divertor plates, impurity seeding, etc.).

- Evaluate SOL turbulence and transport for ITER.

Transport and Fluctuations

- Design, build and install an additional correlation reflectometer using a swept-frequency Gunn diode at about 90-110 GHz to enable correlation length measurements at a density of $1.0 - 1.5 \times 10^{20}$ m^{-3}, which is a typical Ohmic or L-mode density in Alcator C-Mod.

- Obtain measurements of radial correlation length with the swept-frequency correlation reflectometer (132-140GHz) and compare with microturbulence codes.

- Build a curved x-ray crystal spectrometer to measure molybdenum L line emission profiles for determination of T_i and V_ϕ profiles as well as core molybdenum impurity content, using a single PILATUS detector. This development effort parallels plans by R. Barnsley at JET to measure tungsten L lines, as part of the program to diagnose the ITER-like Be/W wall to be installed on JET in 2008.

Integrated Scenarios research for ITER

- Achieve AT parameters in C-Mod plasmas up to the limits of launcher power and source capability.

- Study the modifications in plasma performance resulting from the changes in current and pressure profile.

Future Accomplishments: FY2007 (incremental)

Enhanced Lower Hybrid Physics, Engineering, and Modeling ($350k)

As an incremental level of effort in FY07, we propose to roughly double the PPPL research staff time devoted to Lower Hybrid current drive and its effects on plasma confinement.

The recent successful installation of a stainless steel LH launcher in C-Mod opens up significant new and exciting research opportunities in FY07 for both Lower Hybrid physics studies (coupling, heating, current drive, phase control) and the effect of modified current profiles on plasma confinement and stability. The proposed incremental level of effort will strengthen the overall Lower Hybrid research program on C-Mod and ensure that the fusion program realizes its full return on the investment in LH hardware. The incremental effort is also motivated by the need for a decision, sooner rather than later, regarding the possible inclusion of Lower Hybrid current drive capability in ITER. Data from JET and C-Mod are likely to play a key role in characterizing the feasibility and attractiveness of LHCD on ITER.

The incremental FY07 effort on Lower Hybrid would be directed at three activities:

- Increased participation in LHCD experiments;

- Enhanced numerical modeling of LHCD; and

- Additional engineering support for design of the second Lower Hybrid launcher.

Increased participation in experiments: the incremental effort would provide support for a PPPL lower hybrid physicist to be on-site for all of the C-mod experiments that involve the LH system. This would increase direct participation in LH experiments and improve communication with the remainder of the PPPL-LH team members particularly with regard to modeling and numerical simulation activities.

Enhanced numerical modeling of LHCD: Integrated discharge simulations for C-Mod will be developed using the Tokamak Simulations Code (TSC) and TRANSP. This combination utilizes the free-boundary evolution, plasma control, and predictive transport evolution from TSC, with the sophisticated source modeling and fast particle treatments in TRANSP. This analysis is intended to:

- Reproduce experimental discharge behavior, testing particle and energy transport models and source heating and current drive models.

- Provide projections to new discharges to optimize their programming with particular focus on advanced tokamak scenarios involving internal transport barriers, lower hybrid off-axis CD, and high non-inductive current fraction,

- Provide plasma descriptions that can be analyzed by off-line computations including ideal MHD, fast particle MHD, non-ideal MHD, gyro-kinetic simulations, sophisticated RF calculations, and SOL/divertor analysis. In addition, various computational developments will be required as the work progresses.

The lower hybrid packages in the TSC and TRANSP simulation codes are based on ray tracing modules coupled to simplified Fokker-Planck treatments of the driven current. Though these models have agreed qualitatively with previous experiments on PBX-M, more sophisticated models are being developed within the RF SciDAC and in collaboration with C-Mod and PPPL. In particular, the TORIC code will be coupled with the CQL3D code to provide self-consistent full wave simulations of the lower hybrid driven current at particular single time slices of an experimental discharge. With incremental funding in FY2007 and FY2008, this integrated package would be used to benchmark the accuracy of the lower hybrid driven current used in the time-dependent models described above. This package could also be used to support modeling of conventional ICRF heating scenarios.

Enhanced modeling of fast particle physics and global eigenmodes

The physics of ICRF-induced core-Alfvén modes as well as driven global eigenmodes are currently being studied with the PPPL NOVA-K code. This code directly calculates the eigenmodes frequency and structure, in full geometry, using a linear, perturbative treatment of the continuum damping. To complement these simulations, the TORIC linear full wave code will be utilized to study the structure and damping of driven eigenmodes, using a full kinetic treatment. This effort, which would be leveraged off of development work funded by the RF SciDAC project, would be directed towards simulation of these modes in C-Mod, in support of the experimental program. Depending on the success of the initial exploratory studies in FY2006, the TORIC package would be utilized in subsequent years to:

FY2007: Compare the linearized damping rates for driven global radio frequency eigenmodes obtained with the TORIC code with those obtained with local kinetic models.

FY2008: Compare the structure and damping of driven radio frequency global eigenmodes with and without energetic particles present in the tokamak plasma (RF SciDAC and collaboration with C-Mod and NSTX.

Engineering support: The baseline budget for Alcator C-Mod supports fabrication of a second Lower Hybrid launcher in FY07 and installation in FY08. Engineering design will be carried out by the C-Mod engineering team in FY06 and FY07. The incremental PPPL FY07 effort proposed here would include additional engineering support of these design activities.

CQL3D Integration into TRANSP ($100k)

We propose a two-year incremental effort, supported jointly by the PPPL-CMod collaboration and NSTX, to increase the rate of progress integrating the CQL3D/GENRAY code into TRANSP.
CQL3D is an all-frequencies, multispecies (ions and/or electrons), 2D-in-velocity, 1D-generalized radius, relativistic, bounced-averaged Fokker-Planck code that models LHCD, ICRF, EBW, and HHFW heating and current drive. The funding would primarily support additional efforts by the TRANSP computer scientists for integration activities.

Future Accomplishments: FY2008 (incremental)

Enhanced Lower Hybrid Physics, Engineering, and Modeling ($400k)

Continue the additional research and engineering effort devoted to Lower Hybrid current drive that was begun in FY07.

CQL3D Integration into TRANSP ($175k)

This incremental effort is a continuation of the incremental project to integrate CQL3D/GENRAY into TRANSP described above.

ITER prototype reflectometer design ($125k)

The United States is responsible for delivering a low-field-side reflectometer to ITER. We are fortunate that the most important parameters which affect the performance of the reflectometer on ITER – namely toroidal magnetic field, density, and density gradient scale length – are matched in existing C-Mod plasmas. Therefore, there is an unprecedented opportunity to build and operate a prototype reflectometer diagnostic for ITER on C-Mod. Such a system would both increase confidence in the performance of the ITER reflectometer and would contribute significantly to turbulence studies on C-Mod.

We envision a two-year incremental program (FY2008-09) with a funding profile of $120k in FY08 for design and $750k in FY09 for fabrication.

The objective is a single-channel system operating at a frequency of 180 GHz measuring radial correlation lengths, poloidal correlation lengths, and the group delay (equivalent to measuring the local density gradient). Measuring the group delay will require two sources operating with a stable frequency difference of 50-100MHz. The frequency difference will be maintained with a phase-locked loop mechanism. Measuring the radial correlation length requires either the development of a swept-frequency source ($\Delta\nu$ = 5-10 GHz) with sufficient power at 180 GHz (swept-frequency sources are currently available at ~140 GHz) or else the use of two fixed-frequency sources operating at frequencies separated by several GHz. The poloidal correlation length will be measured with a second poloidally-displaced antenna and receiver.

4.2 The Texas Collaboration

Collaboration Between The Fusion Research Center, University of Texas at Austin and The MIT Plasma Science and Fusion Center

Overview

The Fusion Research Center participates in the operation and maintenance of and data analysis for three diagnostics: charge exchange recombination spectroscopy (CXRS), beam emission spectroscopy (BES), and a high spatial and temporal resolution electron-cyclotron-emission (ECE) system (FRCECE). Turbulence analysis and simulations are provided for some experiments. We also contribute to operation of and diagnostics for the diagnostic neutral beam (DNB).

Our current physics program includes investigation of the relation between turbulence and transport in EDA and ELMing H-mode discharges and in internal transport barriers (ITB's). These will exploit our contributions to high resolution ECE measurements and DNB operation and diagnostics (CXRS and BES) as well as our expertise in gyrokinetic simulation for comparison with real-world diagnostics.

Planned experiments

Electron temperature scale-length measurements and analysis with gyrokinetic microstability computations

In previous campaigns, we developed a tool for precise measurement of electron temperature-gradient scale lengths $L_{Te}(r)$ using the high resolution electron cyclotron emission (ECE) diagnostic, FRCECE. We will exploit this tool for evaluation of analytical predictions of transport and as input into gyrokinetic microstability computations by the continuum codes GKS,[4] GS2,[5] and GYRO.[6] Scale length measurements are important in other areas of the C-Mod program. As noted in the transport section, a critical area for upcoming research is the interplay of R/L_T and R/L_n in the creation and control of C-Mod internal transport barriers.

During previous campaigns, we measured L_{Te} using small changes in the magnetic field. Small, nonperturbing ramps (~1%) in the toroidal field move the ECE viewing volumes along the profile a distance of the order of the channel spacing. The change in signal

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[4] M. Kotschenreuther, G. Rewoldt, and W.M. Tang, Comput. Phys. Commun. **88**, 128 (1995).

[5] W. Dorland, F. Jenko, M. Kotschenreuther, and B.N. Rogers, Phys. Rev. Lett. **85**, 5579 (2000).

[6] J. Candy, R. E. Waltz, J. Comput. Phys. **186**, 545 (2003).

during the ramp provides the gradient and the average signal provides the electron temperature T_e. The ratio of these two quantities is L_{Te}:

$$L_{T_e} = T_e \Big/ \nabla T_e = \overline{ECE(t)} \Big/ \left. \frac{\Delta ECE(t)}{\Delta R} \right]_{\substack{during \\ ramp}} \qquad (1)$$

Thus, the technique provides a direct measurement of the temperature scale length that is independent of the channel-to-channel calibration. The close channel separation (or, high spatial resolution) of the diagnostic is important since only very small changes (< 1%) in the toroidal field are required to achieve high spatial resolution in the scale length measurements. Note that only small toroidal field changes can be allowed if the experiment is to be used for study of internal transport barriers in C-Mod. This is because significant changes of RF deposition region can adversely impact the formation of internal transport barriers. We are confident that these measurements will be generally interesting and can be readily available to others provided they specify the "small TF sweeps" during their experiments.

Accurate measurements of L_{Te} enable meaningful evaluation of theoretical predictions of transport which depend on the gradient of T_e. For example, Fig. 4.2. shows calculations of the electron thermal diffusivity χ_e with the standard large-machine transport empirical formula (Taroni-Bohm)[7] $\chi_e = T_e q^2 \Big/ B_T * R \Big/ L_{T_e}$ using the measured electron temperature scale lengths. The data is compared with the TRANSP calculation of χ_e for a similar discharge. Accurate measurements of L_{Te} are also required for comparisons to the critical gradients for electron temperature-gradient (ETG) computed by gyrokinetic microstability codes. For example, Fig. 4.2. shows the critical T_e gradient profile computed using GKS along with the measured gradient for a representative C-Mod EDA H-mode plasma. These results would imply the plasma is stable to ETG modes outside of *r/a* ~ 0.65 and perhaps unstable inside. (The region inside *r/a* ~ 0.4 is within the sawtooth mixing radius.)

One continuing area of interest to us is to search for evidence of critical gradients for microturbulence. One experiment that we plan to pursue "in background" is measurement of the variation in local L_{Te} as the plasma density is changed to vary the collisionality and the coupling between ion and electron temperatures and thus the critical gradient. In this continuing study, data will be built up over the period of the campaign.

We also propose the measurement of L_{Te} during the ITB discharges. Sawtooth heat-pulse experiments have suggested that there may possibly be a thermal barrier in these C-Mod

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[7] M. Erba, V. Parail, E. Springmann, A Taroni, Plasma Phys. Control. Fusion **37** (1995) 1249.

ITB's. While there is direct evidence for a particle barrier, strong direct evidence for a thermal barrier would add considerably to our understanding. Many of the ITB discharges have small or no sawteeth with steep density gradients. These discharges are ideally suited to local measurement of L_{Te} using field ramps. Any change in the local L_{Te} would be very important in understanding any thermal transport barriers.

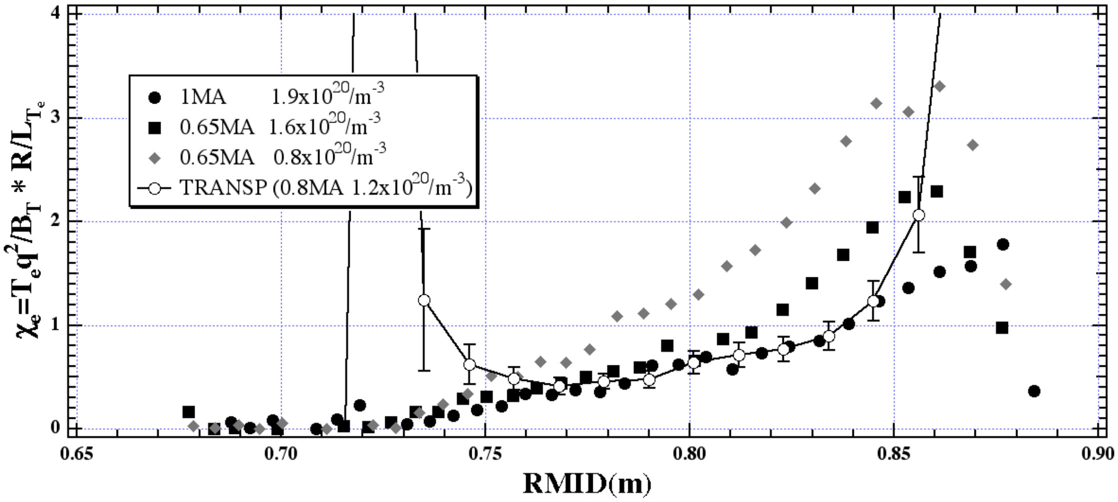

Fig. 4.2.1 Detailed measurements of the local L_{Te} using the FRCECE system are used to calculate χ_e from the Taroni-Bohm empirical formula for three discharge conditions. These are compared to the calculation of χ_e from a TRANSP run for similar conditions.

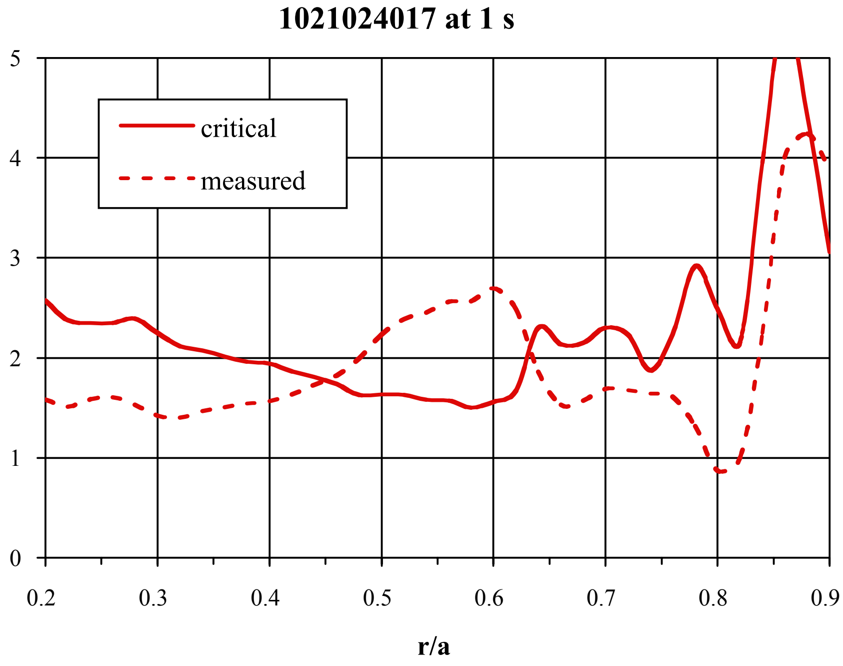

Fig. 4.2.2. Normalized critical electron temperature gradient for ETG modes computed by GKS (solid) and measured gradient (dashed) for a C-Mod EDA H mode plasma.

The L_{Te} in RF-induced internal transport barriers will be contrasted with that in C-Mod ITB's generated in Ohmic discharges. The ITB's induced by off-axis RF can be explained as suppression of specific turbulence modes.[8] Those induced in Ohmic plasmas remain unexplained. Using existing discharges, linear GKS, GS2, or GYRO analysis will be performed near $r/a \sim 0.5$ for L-mode, ELMy H-mode, and Ohmic (both normal and H mode) plasmas to determine the differences in critical gradients. Presumably, the Ohmic H-mode plasma will be the closest to critical. We should be able to infer the profile(s) most responsible for this by direct comparison of the different plasmas, or by perturbations of profiles one at a time in a microstability analysis. This scale length work may be accompanied by measurements of core fluctuations which are discussed below.

Pedestal physics

A yet unresolved puzzle concerning the H-mode pedestal is what determines the width of the steep-gradient region. The transport physics that we know is complex and embodied in gyrokinetic microstability codes such as GKS, GS2, and GYRO. Unfortunately, these codes are not valid in the H-mode pedestal because of violation of the ballooning approximation or of the ordering $\rho_i \ll L_{\text{ped}}$, where ρ_i is the ion gyroradius and L_{ped} is the scale-length of the plasma profiles there. However, a paradigm recently put forward is that the steep-gradient region extends into the core only as far as the $\boldsymbol{E}\times\boldsymbol{B}$ shearing rate can overwhelm the maximum linear growth rate of the turbulence. We will test this paradigm by analyzing the region just at the top of the pedestal, i.e., where the gradients are shallow enough such that the codes are valid, and glean insight into the character of the turbulence.

The first steps to develop this comparison have been taken. An example of linear growth-rate analysis from GKS is shown in Fig. 4.2.3. In the future, we will employ the GS2 code, which can use actual numerical equilibria. We will then compare the maximum linear growth rates to the $\boldsymbol{E}\times\boldsymbol{B}$ shearing rates as determined from measurements. The ion temperature, plasma rotation, and E_r measurements will be extracted from charge-exchange recombination spectroscopy (CXRS) spectra. This spectroscopic measurement may possibly be supplemented by results from spectroscopy of ambient impurity emission. The latter was developed during the last campaign while the long pulse diagnostic neutral beam (required for CXRS) was being commissioned. With the long-pulse beam, ion temperature, rotation, and B^{+5} density measurements are now available. Sample data is shown in Fig. 4.2.4. An example of shear suppression

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[8] D. R. Ernst, P. T. Bonoli, P. J. Catto, W. Dorland, C. L. Fiore, R. S. Granetz, M. Greenwald, A. E. Hubbard, M. Porkolab, M. H. Redi, J. E. Rice, K. Zhurovich, and the Alcator C-Mod Group, Phys. Plasmas **11**, 2637 (2004).

measurements using spectroscopy of ambient impurity emission is shown in Fig. 4.2.5. In this example, the radial electric field is inferred from the momentum balance equation using measurement of the profiles of ion temperature, plasma impurity rotation, and impurity density. In this case, the toroidal rotation velocity did not affect the inference of E_r. This data is limited to the region at the top of the pedestal, and does not extend as far into the plasma as does the CXRS data.

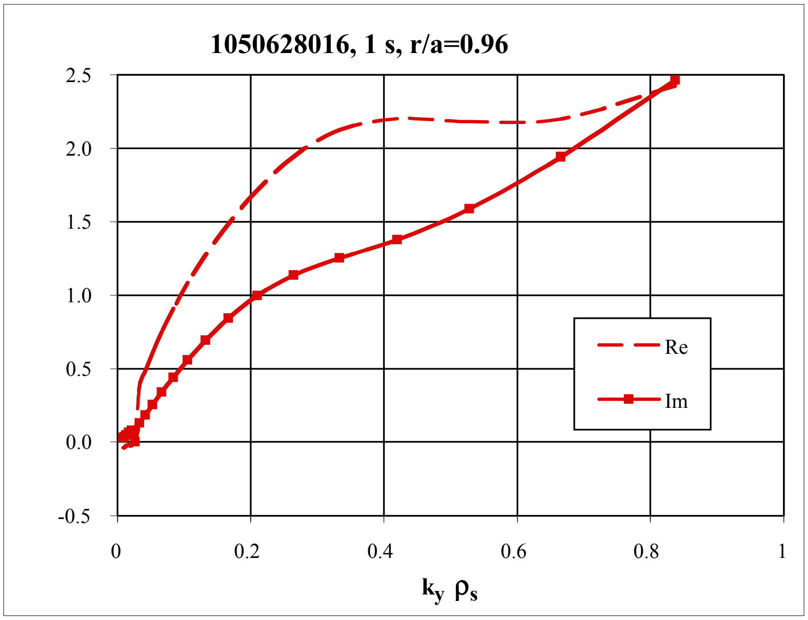

Fig. 4.2.3. Real and imaginary parts of the mode frequencies computed by GKS for a C-Mod EDA H-mode plasma. Positive real frequency corresponds to propagation in the electron diamagnetic direction.

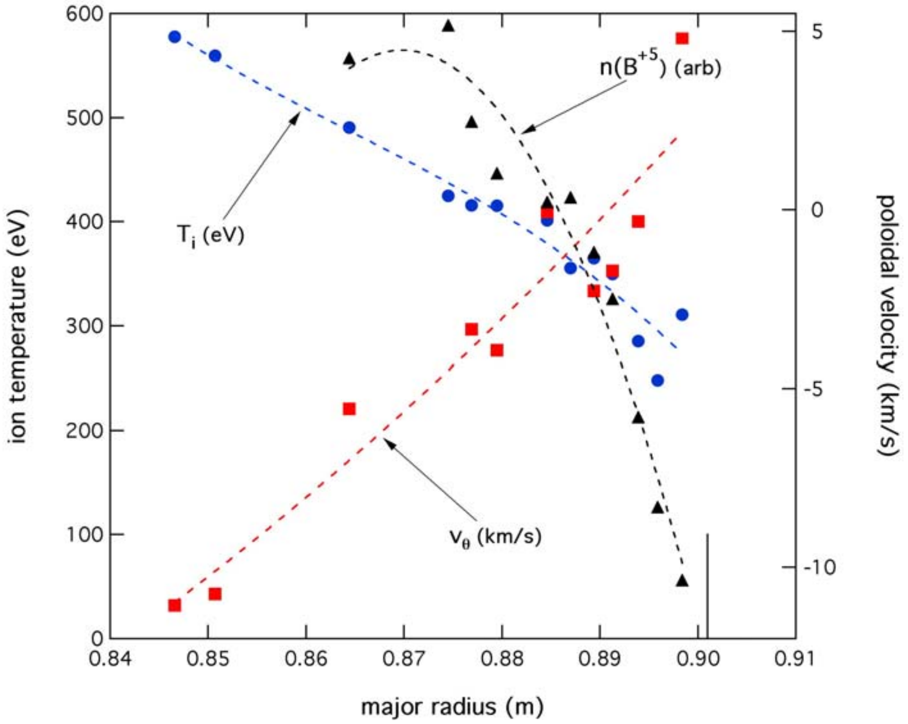

Fig. 4.2.4. Ion temperature, plasma rotation and relative B^{+5} density are shown as a function of radius for a single discharge. The CXRS diagnostic was operated with a temporal resolution of 0.02 s for this example.

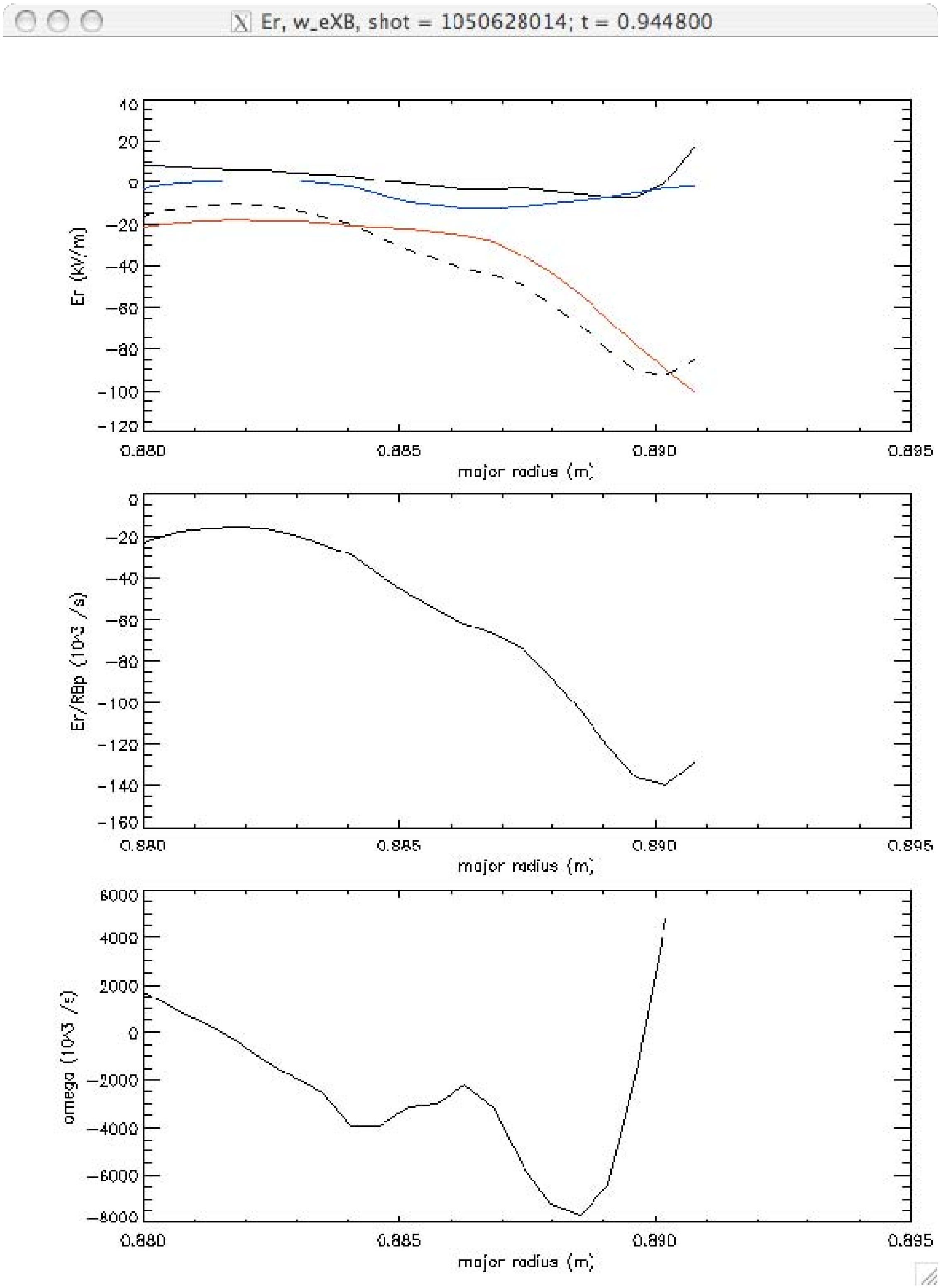

Fig. 4.2.5. Top: Terms in the momentum balance equation contributing to E_r (*dashed*): $(kT_□/e_□)\partial \ln n_□/\partial r$ (*solid*), $(1/e_□)\partial kT_□/\partial r$ (*blue*), $\upsilon_{□\mathrm{tor}}B_{\mathrm{tor}}$ (*red*). Middle: the quantity E_r/RB_{pol}. Bottom: the $E\times B$ shearing rate.

Comparison of Fig. 4.2.3 and Fig. 4.2.5 reveals that at $r/a = 0.96$ (corresponds to $R_{\mathrm{mid}} = 0.884$), the $\boldsymbol{E}\times\boldsymbol{B}$ shearing rate exceeds the maximum linear growth rate from Fig. 4.2.. However, the growth rate clearly needs to be investigated at higher values of $k_y\rho_s$. As mentioned earlier, we also need to use GS2 because of its more accurate rendering of the outer flux surfaces, as well as measured profiles of ion temperature T_i. (The example shown here used a T_i profile calculated by TRANSP.)

Core fluctuations

Measurement of fluctuations with FRCECE was developed during the last two campaigns. With this diagnostic, density fluctuations can be detected via refractive effects in some cases. Electron temperature fluctuations can be detected using cross-correlation techniques. As noted in the transport section, these techniques can be used for undemanding but important applications such as localization of PCI fluctuation measurements, thus greatly increasing the value of the PCI diagnostic. Of course, the fluctuation measurements have utility beyond acting as a supplement for other diagnostics.

Density and temperature fluctuations have been measured in some ITB discharges. This will be continued with measurements in low-field ITB's. The fluctuation measurements will supplement the measurements of L_{Te} which have been discussed above.

Of immediate interest is continuation of investigations of core fluctuations in ITB discharges to determine whether observed fluctuations at ~80 kHz are actually the cause for the saturation of the density in steady-state ITB discharges. This is described in Fig. 4.2.6. In the discharge described by this figure, an ITB is formed by off-axis RF. The conventional wisdom is that the density peaking observed in the ITB can be quenched and a strong density peaking maintained by applying central RF. In this case shown in Fig. 4.2.6, the FRCECE detects a low level of core fluctuations which may be induced by the density gradient formed at the beginning of the ITB. Perhaps in the case of this discharge, that fluctuation is suppressing the density peaking. This data would imply that. When the central RF is turned on, the core fluctuations increase significantly. From past experience, we know that the central RF will quench the density peaking. The conclusion here would seem to be that there is a naturally-occurring core fluctuation that may or may not suppress the density peaking. If it is too weak in a particular case to suppress the density peaking, then it can be enhanced through the use of RF heating in the core. To construct a reasonable explanation, we need to know whether the naturally-occurring fluctuation is the same as the RF induced fluctuation but with a smaller amplitude. Is it always true that the naturally-occurring fluctuation appears and is simply enhanced by the central RF? Is the central RF driving a pre-existing mode or generating a new one?

We also plan to further investigate a puzzling observation: We have observed correlations between core fluctuations and the quasi-coherent (QC) mode that is localized in the edge pedestal of an enhanced-D_α H-mode plasma. Fluctuations believed to be core toroidal Alfvén eigenmodes (TAE modes), measured using external magnetic coils, appear to be coupled to the QC mode, as evidenced by sidebands in the spectra spaced by the QC-mode frequency. There are may be explanations for these core/edge correlations: fast, non-diffusive transport of the QC-mode fluctuations into the core, modulation of the core RF heating by the QC mode, etc. We and relevant C-Mod staff plan to investigate this phenomenon further in a number of ways, including cross-correlating individual FRCECE channels with the magnetics signals as was done on JET[9] and employing cross-bispectral analysis to verify actual coupling.

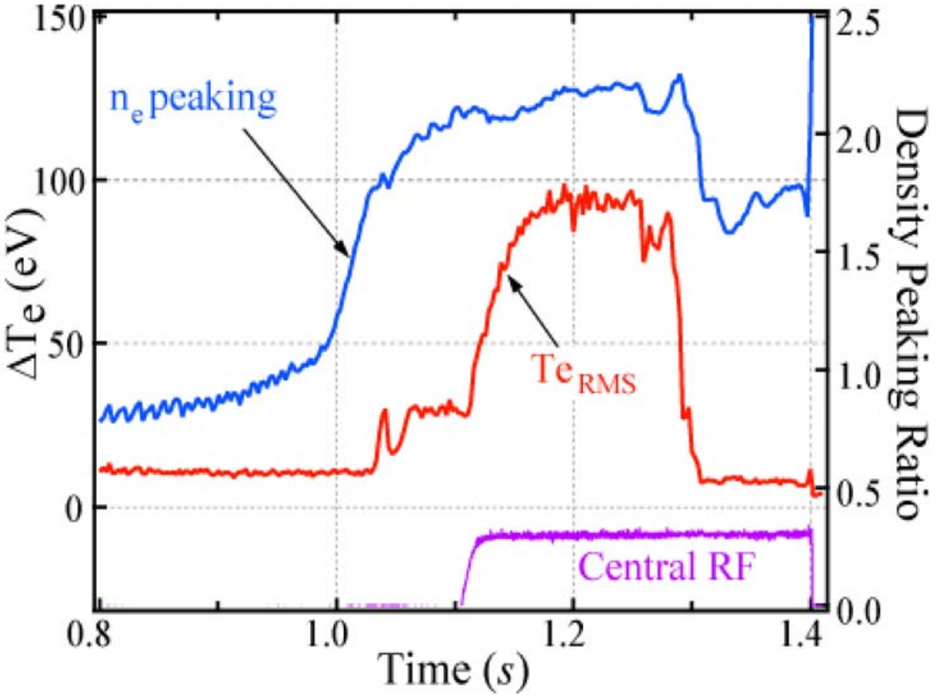

Fig. 4.2.6 Time traces of the fluctuation amplitude measured by a central FRCECE channel (Te_{RMS}), the ratio of the central density to the pedestal density measured by the visible bremsstrahlung array, and the central RF power for an ITB discharge. Although indicated as Te_{RMS} in the figure, this is actually the ECE signal which may contain a density component due to refractivity.

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[9] S. E. Sharapov *et al.*, JET Report No. EFDA-JET-PR(01)87, 1987.

Gyrokinetic Microstability Analysis and Nonlinear Simulations

The interpretation of data using gyrokinetic microstability codes (in the examples here, GKS) tacitly assumes that the codes are “correct.” However, they have not yet been shown to unambiguously and consistently reproduce reality, represented by experiment. We therefore also intend to pursue verification and validation of the codes, including development of synthetic diagnostics for nonlinear simulations (also a component of validation). This will be done in the course of analyzing the plasmas used for experiments. Detailed comparison with data (validation) will naturally arise in the experimental analysis. Perusing the physics which appears in the codes and which is relevant to our experiments via test cases and intercomparison of codes (verification) will be an important part of the analysis. Meaningful comparison of the codes with real-world diagnostics (for example, FRCECE) requires the construction of synthetic diagnostics[10] to allow for the unique characteristics of the diagnostics.

Administrative

Personnel

The Fusion Research Center participates in the operation and maintenance of and data analysis for three diagnostics, Charge Exchange Recombination Spectroscopy (CXRS), Beam Emission Spectroscopy (BES), and the FRCECE Radiometer, a high spatial and temporal resolution ECE system. Turbulence simulations are provided for some experiments. We also contribute to operation of and diagnostics for the diagnostic neutral beam.

We provide 3.25 full-time-equivalents (FTE's) which are distributed as follows:

CXRS: 1.25 FTE (staff member and full time student)
BES and turbulence computations: 0.5 FTE (staff member)
ECE: 0.76 (staff member and full time student)
Technical support: 0.3 FTE
Administrative: 01.5 FTE

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[10] R. V. Bravenec and W. M. Nevins, Rev. Sci. Instrum. **77**, 015101 (2006).

4.3 MDSplus

Recent Highlights

MDSplus software maintenance, bug fixes and ongoing support for off-site installations continues to be a major activity for the MDSplus development group. New support for data acquisition hardware was added, with emphasis on CPCI devices which have become increasingly popular. Enhancements of the support for firewire digital camera systems have been added. Site specific work has been done for experimental groups at Columbia University, KSTAR (KBSI-Korea), CHS (NIFS- Japan), LDX (MIT), DIII-D. A list of active fusion sites is appended at the end of this document. An MDSplus web service was prototyped which enables the retrieval of MDSplus data using standard Service Oriented Architecture protocols. Additional content has been added to the MDSplus web site (www.mdsplus.org). An MDSplus workshop was conducted at the Fifth IAEA Technical Meeting on Control, Data Acquisition, and Remote Participation for Fusion Research meeting in Budapest in July 2005. Discussions of long pulse extensions to MDSplus were discussed.

Plans

Support for remote MDSplus sites will be increasing as the number of sites and the number of users increases. An ongoing effort to improve online documentation and to train local support staff at each of the major sites where the code is used will be made. The hope is to hold a MDSplus users meetings on an biennial basis. The next meeting is scheduled in conjunction with the IAEA technical meeting in Japan in April 2007. MDSplus software maintenance will continue to be a principle activity

In response to requests from several new experiments and the needs of advanced simulations, extensions to MDSplus to support long-pulse operation will begin with conceptual design work. Both applications will require the ability to store data incrementally and will need a conceptual framework for describing and browsing data sets which are too large to be displayed by conventional means. (To complete this work in a timely manner would require additional funding which is being requested in a new proposal under the 2006 SciDAC call.) Upgrades to the Scope utility are planned, with the highest priority being the ability to display multiple traces in each panel. Capabilities for color plotting would be added at the same time. Support for additional data acquisition devices – particularly CPCI will be provided as useful modules are identified.

Partial List of MDSplus sites.

US:

1. PSFC - MIT
2. PPPL
3. GA
4. U. Wisconsin
5. U. Texas
6. UCLA
7. Columbia
8. U. Washington
9. Auburn University
10. Los Alamos
11. University of Maryland
12. University of Utah
13. U.C. Irvine

International:

1. IGI- Padua, Italy (RFX)
2. EPFL – Lausanne, Switzerland (TCV)
3. EFDA-JET – Culham, UK (JET)
4. UKAEA – Culham, UK
5. IPP-Garching, Germany
6. CEA – Cadarache, France (TORE-SUPRA)
7. Kurchatov Institute of Nuclear Fusion – Moscow, Russia
8. IPP – Hefei, China (HT-7)
9. Korean Basic Science Institute, Taejon, S. Korea (KSTAR)
10. NIFS – Toki, Japan
11. Australia National University, Canberra (HELIAC)
12. ENEA - Frascatti, Italy
13. University of Quebec

4.4 International Collaborations

4.4.1 JET

Collaborations between C-Mod and JET are ongoing in a number of topical areas, including transport, macroscopic stability, RF physics and plasma-boundary interactions. Subjects being emphasized are edge barrier stability and dynamics, core transport and confinement, error fields and locked modes, fast particle driven Alfvén modes, massive gas jet disruption mitigation, ICRF and Lower Hybrid physics, scrape-off-layer transport, recycling and isotope retention, as described earlier in this work proposal. Joint experiments are coordinated through the ITPA process. A portion of this effort is supported by funding from the International Collaborations budget, and the rest from the base C-Mod program. The explicit international collaboration funding supports a small number of trips to JET, as well as small fractions of the salaries for some of the scientists involved in the research. In addition to the topics already listed, we are working with the JET ITER-like wall project on solid lamellae tungsten tiles. Both C-Mod and Jülich groups are developing designs for this in parallel. We are sharing information, designs, and test results. The Jülich group also kindly agreed to test our lamellae design tiles. We plan to continue the collaboration as we manufacture the lamellae tiles for C-Mod and continue development in this area.

4.4.2 KSTAR

Accomplishments (FY2005):
MIT provided support and consultation for the MDSplus software as required. MDSplus was modified to utilize the GLOBUS BIDI XIO driver for network communications. This driver will utilize parallel network I/O to provide high data transfer rates when communicating over fast networks between geographically dispersed sites. Additional work on the BIDI XIO driver by the GLOBUS development team is required to fully reap the benefits of the parallel I/O. This work should be completed at Argonne National Laboratory later this calendar year.

Conceptional designs of enhanced features of MDSplus to support long pulse experiments have been completed. The current implementation of MDSplus is designed for short pulses and each signal is stored as a single record after the pulse has occurred. The enhancements to MDSplus will enable applications to append data records during the course of the pulse. Applications will then be able to retrieve "time windows" of a signal without needing to read in all the data acquired for the signal during the full length of the pulse.

Plans:

The KSTAR is planning to use MDSplus software for data acquisition and data management. MIT will provide support and consultation for this implementation including help for the KSTAR team as it develops drivers for data acquisition equipment not currently supported. MIT will continue to work on high speed network data throughput enhancements to MDSplus to enable efficient transfer of large quantities of experiment data to remote sites. MIT will also be developing prototypes of the long pulse extensions to MDSplus which will enable the appending of data segments to signals during the course of the pulse. The prototypes will be used to evaluate the performance and functionality of this data appending capability. It is envisioned that a series of design, prototype and evaluation cycles will be required to implement a fully functional long pulse MDSplus system while retaining compatibility with the existing MDSplus data stores.

Appendix A.

Alcator C-Mod Publications – FY2005 to present

Papers Published in Refereed Journals

Basse N.P., Zoletnik S., Michelsen P.K., “Study of intermittent small-scale turbulence in Wendelstein 7-AS plasmas during controlled confinement transitions ,” Phys. Plasmas **12** (1): Art. No. 012507 Jan 2005

Basse, N.P. et al., “Small-angle scattering theory revisited: Photocurrent and spatial localization”, Physica Scripta **71** (2005) 280-292

Basse, N.P., Edlund, E.M., Ernst, D.R et al., “Characterization of core and edge turbulence in L- and enhanced D-alpha H-mode Alcator C-Mod plasmas”, Physics of Plasmas **12** (2005) 052512

Basse, N.P. “Density fluctuations on mm and Mpc scales”, Physics Letts A 340 (2005) 456-460

Batchelor, D. B., Berry, L. A., Bonoli, P. T., Carter, M. D., Choi, M., D’Azevedo, E., D’Ippolito, D. A., Gorelenkov, N., Harvey, R. W., Jaeger, E. F., Myra, J. R., Okuda, H., Phillips, C. K., Smithe, D. N., and Wright, J. C., “Electromagnetic mode conversion: understanding waves that suddenly change their nature” Journal of Physics: Conference Series **16** (2005) 35–39

Bonnin, X., Coster, D., Schneider, R., Reiter, D., Rozhansky V., Voskoboynikov, S., ‘Modelling and consequences of drift effects in the edge plasma of Alcator C-Mod,” J. Nucl. Matter. **337-339**, 301-304 (2005).

Chung, T., Hutchinson, I.H., Lipschultz, B., LaBombard, B., and Lisgo, S., "DIVIMP modeling of impurity flows and screening in Alcator C-Mod," J. Nucl. Mater. **337-339** (2005) 109.

Cordey, J.G., Snipes, J.A., Greenwald, M. Sugiyama, ,L.,et al, " Scaling of the energy confinement time with β and collisionality approaching ITER conditions ", Nucl. Fusion **45** (2005) 1078–1084

Graves, T., LaBombard, B., Wukitch, S.J. and Hutchinson, I.H. “The coaxial multipactor experiment (CMX): A facility for investigating multipactor discharges” Rev. Sci. Instrum. **77,** 014701 (2006).

Greenwald, M. et al, “Overview of Alcator C-Mod Program” Nucl. Fusion **45** (2005) S109–S117

Grulke, O., Terry, J. L., LaBombard, B., and Zweben S. J., “Radially propagating fluctuation structures in the scrape-off layer of Alcator C-Mod” Phys. Plasmas **13**, 012306 (2006)

Houlberg, W.A., Gormezano, C., Artaud, J.F., Barbato, E., Basiuk, V., Becoulet, A., Bonoli, P., Budny, R.V., Eriksson, L.G., Farina, D., Gribov, Yu., Harvey, R.W., Hobirk, J., Imbeaux, F., Kessel, C.E., Leonov, V., Murakami, M., Polevoi, A.; Poli, E.; Prater, R., St. John, H., Volpe, F., Westerhof, E., Zvonkov, A., ITPA Steady State Operation Topical Group; ITPA Confinement Database and Modeling Topical Group “Integrated modelling of the current profile in steady-state and hybrid ITER scenarios”

Hutchinson, I.H. “Ion Collection by a Sphere in a Flowing Plasma: 3. Floating Potential and Drag Force,” Plasma Phys. Control. Fusion **47** No 1 (2005) 71-87.

Hutchinson, I. H., “Collisionless ion drag force on a spherical grain” Plasma Physics and Controlled Fusion, v **48**, n 2, Feb 1, 2006, p 185-202

LaBombard, B., Rice, J.E., Hubbard, A.E., Hughes, J.W., Greenwald, M., Granetz, R.S., Irby, J.H., Lin, Y., Lipschultz, B., Marmar, E.S., Marr, K., Mossessian, D., Parker, R., Rowan, W., Smick, N., Snipes, J.A., Terry, J.L., Wolfe, S.M., Wukitch, S.J., and the Alcator C-Mod Team., "Transport-driven scrape-off layer flows and the x-point dependence of the L-H power threshold in Alcator C-Mod," Phys. Plasmas **12** (2005) 056111.

LaBombard, B. Hughes, J.W.; Mossessian, D.; Greenwald, M.; Lipschultz, B.; Terry, J.L. “Evidence for electromagnetic fluid drift turbulence controlling the edge plasma state in the Alcator C-Mod tokamak” NuclFusion, **v 45**, n 12, Dec 1, 2005, p 1658-1675

Leggate, H., Cordey, J.G., Lomas, P.J., McDonald, D.C., Maddison, G., Petty, C.C., Snipes, J., Voitsekhovitch, I. and JET EFDA contributors **“**The significance of the dimensionless collisionality and the Greenwald fraction in the scaling of confinement” contributed EPS paper for Tarragona 2005

Lipschultz, B., Whyte, D., LaBombard, B., “Comparison of particle transport in the Scrapeoff Layer plasmas of Alcator C-Mod and DIII-D”, Plasma Phys. Control. Fusion **47** (2005) 1559–1578

Lipschultz B, Haasz AA, LaBombard B, et al., “Proceedings of the 16th International Conference on Plasma Surface Interactions in Controlled Fusion Devices – Preface” J. of Nucl. Materials **337-39** (1-3): VII-VIII Mar 1 2005

Lin, Y., S. Wukitch et al., “Observation and modeling of ion cyclotron range of frequencies waves in the mode conversion region of Alcator C-Mod”, Plasma Physics and Controlled Fusion **47**, 1207 (2005).

Lisgo, S., Borner, P., Boswell, C., Elder, D., LaBombard, B., Lipschultz, B., Pitcher, C.S., Reiter, D., Stangeby, P.C., Terry, J.L., and Wiesen, S., "OSM-EIRENE modeling of neutral pressures in the Alcator C-Mod divertor," J. Nucl. Mater. **337-339** (2005) 139.

Lynch, V.E., Carreras, B.A., Sanchez, R., LaBombard, B., van Milligen, B.P., and Newman, D.E., "Determination of long-range correlations by quiet-time statistics," Phys. Plasmas **12** (2005) 052304.

Marr K, Lipschultz B, LaBombard B, et al. Spectroscopic measurements of plasma flow in the SOL in C-Mod J. of Nucl. Materials **337-39** (1-3): 286-290 Mar 1 2005

Pigarov, A.Y., Krasheninnikov, S.I., Brooks, N., Hollmann, E., Maingi, R., Labombard, B., Lipschultz, et al "Multi-ion fluid simulation of tokamak edge plasmas including non-diffusive anomalous cross-field transport," J. of Nucl. Materials **337–339** (2005) 371

Redi, M. H., Dorland, W., Fiore, C. L., Baumgaertel, J. A.,et al,"Microturbulent drift mode stability before internal transport barrier formation in the Alcator C-Mod radio frequency heated H-mode" Phys. Plasmas **12**, 072519 (2005)

Rice, J.E., Hubbard, A.E.; Hughes, J.W.; Greenwald, M.J.; LaBombard, B.; Irby, J.H.; Lin, Y.; Marmar, E.S.; Mossessian, D.; Wolfe, S.M.; Wukitch, S.J. "The dependence of core rotation on magnetic configuration and the relation to the H-mode power threshold in alcator C-mod plasmas with no momentum input," *Nuclear Fusion*, v **45**, n 4, April 2005, p 251-

Smick, N., LaBombard, B., and Pitcher, C.S., "Plasma profiles and flows in the high-field side scrape-off layer in Alcator C-Mod," J. Nucl. Mater. **337-339** (2005) 281.

Snipes, J. A., et al., "Active and fast particle driven Alfvén eigenmodes in Alcator C-Mod", Physics of Plasmas **12** (2005) 056102.

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Rice, J.E.,, Hubbard, A.E., Hughes, J.W., Greenwald, M.J., LaBombard, B., Irby, J.H., Lin, Y., Marmar, E.S., Wolfe, S.M.and Wukitch, S.J. "The Dependence of Core Rotation on Magnetic Configuration and the Relation to the H-mode Power Threshold in Alcator C-Mod Plasmas with No Momentum Input"

Fiore , C. L., Basse, N.P., P. T., Bonoli, P. T., Ernst, D.R., Greenwald, M.J., Hubbard, A.E., Marmar, E.S., Rice J.E., Wukitch, S.J.and Zhurovich, K. "Prospects for High Performance Internal Transport Barriers in Alcator C-Mod"

Granetz, R., Bader, A., Terry, J., Hughes, J.W., "Dust Observations and Development of a Real-time Dust Diagnostic on Alcator C-Mod"

Snipes, J. A., Edlund, E., Gorelenkov, N., Porkolab, M., Sears, J., and Wukitch S., "Alfvén Cascades in Alcator C-Mod"

Terry, J.L., Basse, N.P., Cziegler, I., Greenwald, M.J., LaBombard, B., Porkolab, M., Veto, B. and S.J. Zweben "Characteristics of the Quasi-Coherent Fluctuation in the Edge of Alcator C-Mod Plasmas"

Zhurovich , K., Fiore, C. L., Bonoli, P. T., Ernst, D.R., Hubbard, A.E., Greenwald, M.J., Marmar, E.S., and Rice J.E. "Investigation of triggering mechanism for internal transport barriers in Alcator C-Mod"

16th Topical Conference on Radio Frequency Power in Plasmas and US-Japan RF Physics Workshop: Physics of Plasmas Produced by Large and Steady State RF Power
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Invited papers

Jaeger, E.F., Berry, L.A., Harvey, R.W., Myra, J.R., Dumont, R.J., Phillips, C.K., Smithe, D.N., Batchelor, D.B., Bonoli, P.T., Cater, M.D., D'Azevedo, E., D'Ippolito, D.A., Wright, J.C., “Self-consistent full-wave / Fokker Planck calculations for ion cyclotron heating in non-Maxwellian plasmas”,

Wright, J.C., Bonoli, **P.T.**, Brambilla, M. *et al*., “Full-wave electromagnetic field simulations of lower hybrid waves in tokamks”,

Posters

Choi, M., Chan, V.S., Tang, V., Bonoli, **P.**, Pinsker, R.I., and Wright, J.C., “Monte Carlo Orbit / Full-wave simulation of fast Alfven Wave (FW) damping on resonant ions in tokamaks”, p. 31.

Harvey, R.W., Jaeger, F., Berry, L.A., Ershov, N.M., Smirnov, A.P., **Bonoli, P.,** Wright, J.C., Batchelor, D.B., D’Azevedo, E., Carter, M.D., and Smithe, D.N., “Velocity-space diffusion coefficients due to full-wave ICRF fields in toroidal geometry”, p. 46.

Parker, R. “Implementation of LHCD Experiments on Alcator C-Mod”

Wallace, G., “Microstrip Directional Coupler Design for a Reduced Height Waveguide”

Schilling, G, “Initial Operation of the Alcator C-Mod ICRF Antennas with High-Z Metal Antenna Guards”

Zweben, S., “Edge Minority Heating Experiment in Alcator C-Mod”

Porkolab, M., “ICRF Heated Current Ramp Discharges and Alfvén Cascades in Alcator C-Mod”

32nd European Physical Society Conference on Plasma Physics.
Tarragona, Spain
27 June - 1 July 2005

Invited Orals

LaBombard, B. “Transport-driven scrape-off layer flows, the role of the X-point and connections to the L-H power threshold in Alcator C-Mod”

Contributed Orals

Grulke, O., “Turbulence Imaging of Spatiotemporal Fluctuation Structures in the Scrape-Off Layer of Alcator C-Mod”

10th IAEA Technical Meeting on H–mode Physics and Transport Barriers, St. Petersburg, Russia, Sept 28-30th 2005.

Biewer, T.M., Hughes, J.W., Chang*, C.S., Hubbard, A.E., Ku*, S.H., Marmar, E.,"Extensions to Pedestal Scaling Studies on the Alcator C-Mod Tokamak"

Hubbard, A.E., Kamiya, K., Oyama, N., Basse, N., Biewer, T., Edlund, E., Hughes, J.W., Lin, L.M., Porkolab, M., Rowan, W., Snipes, J., Terry, J., and Wolfe, S.M. ,"Comparisons of small ELM H-Mode regimes on the Alcator C-Mod and JFT-2M tokamaks"

Oyama N, Gohil P, Horton L D, Hubbard A E , Hughes J W , Kamada Y, Kamiya K, Leonard A W, Loarte A, Maingi R, Saibene G, Sartori R, Stober J K, Suttrop W, Urano H, West W P and the ITPA Pedestal Topical Group, "Pedestal conditions for small ELM regimes in tokamaks",

47th Annual Meeting of the APS Division of Plasma Physics, Denver, October 2005

Invited Orals

R., Granetz, "Gas Jet Disruption Mitigation Studies on AlcatorC-MOD"

J., Hughes "Advances in measurement and modeling of the H-mode pedestal on the AlcatorC-Mod tokamak"

B., Lipschultz "Operation of Alcator C-Mod with high-Z plasma facing components and implications"
S., Zweben "Structure and Motion of Edge Turbulence in NSTX and Alcator C-Mod"

Contributed Orals

Biewer, T., "Extension of Pedestal Scaling Studies on the Alcator C-Mod Tokamak"

Granetz, R., "B-field scaling of locked mode error field threshold in Alcator C-Mod"

Graves, T., "Electron Multipactor Discharges in Vacuum Transmission Lines for VHF"

Hutchinson, I., "Radiation loss and confinement performance of Alcator C-Mod"

LaBombard, B., "Operational Phase Space of the Edge Plasma in Alcator C-Mod"

Marmar, E., "Results of between-shot boronization on Alcator C-Mod"

Porkolab, M., "Alfvén Cascade experiment and modeling"

S., Scott "Overview of Recent Alcator C-Mod Results"

Terry, J., "Investigation of ELMs on Alcator C-Mod"

Whyte, D., “Disruptions: Mitigation and D/H recovery”

Posters

Bespamyatnov, I., “Ion temperature in Alcator C-Mod H-mode discharges with ELMs and with the quasi-coherent mode”

Bonoli, P., “Self-Consistent Studies of Lower Hybrid Current Drive in ITER Relevant Regimes”

Bose, B., “Observations of Pellet Ablation in Alcator C-Mod”

Bravenec, R., “Characterization of the New Diagnostic Neutral Beam on Alcator C-Mod”
Dominguez, A., “Reflectometry Upgrade on Alcator C-Mod”

Edlund, E., “Alfvén cascades: NOVA-K modeling of recent experiments”

Ernst, D., “New Developments in Trapped Electron Mode Turbulence”

Fiore, C., “High Performance Internal Transport Barriers in Alcator C-Mod”

Graf, A., “Visible Spectroscopy from a Transmission Grating Spectrometer for Doppler Measurements on Alcator C-Mod”

Ince Cushman, A., “Spatially resolved Neon and Fluorine soft X-ray spectra on Alcator C-MOD”

Izzo, V. “Simulations of disruption mitigation by high-pressure gas jet on Alcator C-Mod”

Ko, J., “Study on the Characteristics of Wire Grid Polarizer for Alcator C-Mod Motional Stark Effect Diagnostic”

Lin L., “An upgraded PCI diagnostic to detect and localize high-k waves and fluctuations in Alcator C-Mod”

Lin, Y., “Plasma Performance Study of an All-metal Tokamak -- Alcator C-Mod”

Liptac, J., “Modeling of Lower Hybrid Experiments on Alcator C-Mod”

Marr, K., “Edge toroidal flow measurements”

McDermott, R., “Uncertainties in CXS measurements”

Mikkelson, D., “The role of kinetic electron effects in gyrokinetic turbulence simulations at high and low collisionality”

Parisot, A., “Initial results from Mode Conversion Current Experiments on Alcator C-Mod”

Podpaly, Y., “A Statistical Analysis of Momentum Transport Scaling in Alcator C-Mod Plasmas with No Momentum Input”

Philllips, P., “Density and Temperature Fluctuations in on Alcator C-Mod Plasmas with Peaked Density Profiles”

Parker, R., “First Results from the Alcator C-Mod Lower Hybrid Experiment”

Reinke, M., “Molybdenum density profiles on C-Mod using FAC generated cooling curves”

Rice, J., “The H-mode Power Threshold and Plasma Rotation in Alcator C-Mod”

Rowan, W., “Poloidal rotation and edge impurity behavior in Alcator C-Mod”

Schmidt, A.,, “Modeling of non-thermal Electron Cyclotron emission due to Lower Hybrid” Current Drive on Alcator C-Mod”

Sears, J., “Results from Real-time Open-loop Alfvén Eigenmode excitation in Alcator C-Mod”

Smick, N., “Design Verification Results for New Inner Wall Probe Diagnostic”

Smith, K., “Initial Results from the C-Mod Prototype Polarimeter/Interferometer”

Snipes, J., “A Comparison of Measured and Calculated TAE Damping Rates in Alcator C-Mod”

Tang, V., “Experimental studies of C-Mod ICRF minority tails via a multi-channel compact neutral particle analyzer”

Wolfe, S., “The C-Mod Digital Plasma Control System”

Wukitch, S., “Comparison of D(3He) and D(H) minority heated discharges”

Zhurovich, K., “Investigation of the triggering mechanism for internal transport barriers in Alcator C-Mod”

10th International Conference on Accelerators and Large Physics Control Systems”, Geneva, Switzerland, October 10 -14, 2005.

Greenwald, M., Schissel, D., Burruss, J., Fredian, T., Stillerman, J., Lister, J., “Visions for data management and remote participation on ITER”,

U.S. Burning Plasma Workshop
Oak Ridge National Laboratory
Oak Ridge, Tennessee
December 7-9, 2005
Marmar, E.S., Chair, Program Committee

Seminars

Greenwald, M., “Visions for data management and remote participation on ITER”, Plasma Seminar, CRPP-EPFL, October 2005

Greenwald, M., “Core and Edge Transport in Alcator C-Mod: Connections Across the Separatrix", Applied Physics Colloquium, Columbia, March 2005

Appendix B.

Summary National Budgets, Run Time and Staffing

		FY06 Approp.	FY07A Request	FY08D reduced	FY08A level	FY08B full
Funding ($ Thousands)						
Research		8,510	8,890	8,110	8,890	11,280
Facility Operations		13,207	13,939	12,450	13,939	18,220
Research Capital Equipment		293	302	250	302	360
Operations Capital Equipment		99	100	90	100	120
PPPL Collaborations		1,993	2,047	1,860	2,047	2,747
UTx Collaborations		410	415	390	415	475
LANL Collaborations		99	100	90	100	120
MDSplus		147	151	140	151	200
International Activities		75	80	60	80	100
Total (inc. International)		21,792	22,909	20,620	22,909	29,600
Staff Levels (FTEs)						
Scientists & Engineers		55.2	54.76	49.4	53.9	65.5
Technicians		25.5	25.6	24.3	25.6	30.3
Admin/Support/Clerical/OH		12.2	12.61	11.9	13.4	14.6
Professors		0.2	0.2	0.2	0.2	0.2
Postdocs		2.0	2.0	0	2.0	3.0
Graduate Students		29	28	26	28	31
Industrial Subcontractors		1.4	1.2	1.0	1.0	1.4
Total		125.5	124.4	112.8	124.1	146.0
	FY05 Actual	FY06 Approp.	FY07A Request	FY08D reduced	FY08A level	FY08B full
Facility Run Schedule						
Research Run Weeks	18.4	14	15	8	13	25
Users (Annual)						
Host	38	39	39	35	38	52
Non-host (US)	63	65	66	59	63	93
Non-host (foreign)	46	*48*	48	40	45	55
Graduate students	29	29	29	26	29	31
Undergraduate students	5	7	7	4	6	10
Total Users	181	188	189	164	181	241
Operations Staff (Annual)						
Host	70	67	68	63	67	76
Non-host	*4*	4	4	3	4	5
Total	*74*	71	72	66	71	81

Appendix C.

Alcator C-Mod Program Detail in Bullet Form

FY06 Appropriation: 14 weeks total research operations (1 week = 4 days, 8 hrs/day)

Areas of Emphasis

- Inductive H-Mode scenarios (ITER baseline)
    - First wall, materials studies
    - Integrated H-mode scenario development, performance enhancement
    - Pedestal and edge relaxation studies
        - Edge stability; ELM transport
        - Pedestal width and height physics
- Steady state operation (ITER-AT)
    - Internal transport barrier dynamics and control
    - Development of real-time j-profile control using current-drive and heating actuators
        - Main thrust of LDCD program; also MCCD, FWCD
        - State of the art modeling tools being developed and applied
- Pedestal and core transport studies
    - Self-generated flows and momentum transport
    - Comparisons of turbulence measurements with simulations
    - ITB access and control mechanisms
    - Electron thermal transport and fluctuations
        - Upgraded diagnostics
    - Construct physics-based and empirical scalings of pedestal parameters
    - Improve predictive capability for ELM size and frequency and assess accessibility to high performance regimes with small or no ELMs
    - Improve predictive capability of pedestal structure through profile modeling
- Plasma Boundary Physics and Technology
    - Improve understanding of SOL plasma interaction with main chamber
    - Improve understanding of tritium retention and the processes that determine it
        - Understand deuterium retention dynamics on tiles (including sides) for B and Mo
        - Understand hydrogen removal at low tile temperature
    - Develop improved prescription of inter-ELM transport and perpendicular SOL transport
    - Dimensionless cross-machine comparisons for SOL physics
- Wave-Plasma Interactions
    - Minority ICRF ^{3}He heating (50 MHz @ 5.3 T first, then 80 MHz @ 8 T)
    - Mode conversion flow and current drive (ICRF)
    - Lower Hybrid RF
        - LH coupling physics

- Phase studies (current drive, heating, accessibility, radial deposition)
- Increased power (~500 kW)
  - Fast electron studies

- Macroscopic Stability
  - Disruption mitigation
    - Extend to higher absolute pressures (~ITER)
    - Close coupling to 3-d MHD simulations
  - Intermediate toroidal mode number Alfven Eigenmodes
    - Active antennas, passive magnetics, PCI
    - Fast particle drive (ICRF minority tails)
  - MHD stability analysis of H-Mode edge transport barriers under type I and tolerable ELM conditions
  - Error fields and locked modes
    - Resolve differences between single-machine and multi-machine scaling studies
      - Extrapolation to ITER
  - Seed island control (NTMs)
    - Sawtooth stabilization
      - Mainly ICRF tools
  - New disruption database, including conventional and advanced scenarios
    - Contribute data at high absolute power and energy density

Plain English Goal

- Disruption mitigation of high absolute pressure plasmas

FY07D *10% below FY2007A Guidance (*10 weeks research operations)

Plans

- Commission divertor cryopump density control
  - Lower collisionality plasmas for all studies – especially important for:
    - LHCD efficiency
    - Edge pedestal stability and ELM studies
- Exploitation of Lower Hybrid system at powers exceeding 1 MW
  - Far off-axis current drive for AT regimes
  - Phase control – variation of radial deposition and driven current
- Progress for highest priority science topics, but at significantly reduced pace

Implications

- Layoff 2 engineers, 1 technician, 1 scientist, 1 post-doc; do not replace 2 graduate students
- Reduced runtime (4 weeks)
  - Even fewer of high priority runs can be completed (to less than ¼)
- Defer 2’nd Lower Hybrid launcher

    - Power handling capability of single LH launcher expected to limit total power to plasma
        - Less driven current
        - Less flexibility in launched spectrum, leading to poorer localization of driven current
- Delay completion of polarimetry diagnostic to measure j(r)
    - At least 1 year delay
    - Adjunct to MSE; only current profile measurement for highest density plasmas; overlaps with MSE in optimal density range for LHCD
    - Geometry, density and field the same as on ITER
        - Valuable prototyping lost
- Defer data acquisition and computing infrastructure upgrades
    - Hardware becomes obsolete on time scale of about 3 years

FY07A Guidance: 15 weeks research operations

Prioritized increments:

- Add 5 weeks research operation (to 15 total), and restore personnel cuts
    - Increased productivity across all topical areas
        - Particularly important to take advantage of new tools (LH, cryopump, diagnostics) developed in FY06
    - Student training maintained
- 2nd LHCD launcher on schedule
    - optimal use of available source power
    - compound spectrum for increased flexibility of deposition (and thus current profile control)
- Polarimetry development on schedule
    - Current profile measurements are critical to understanding of LH experiments
    - Enhanced understanding of all regimes, particularly for high density conventional H-mode (ITER baseline)
- Data acquisition upgrade pace maintained
    - Improving reliability and productivity

Research Plan Highlights (see body of this Work Proposal for details)

- Current profile control with Lower Hybrid
    - Maximize power through single antenna
    - Investigate phase control of deposition and current profile
- First experiments with active cryopumping for wall conditioning and density control
    - Efficient far-off axis current drive at optimal densities in H-mode
    - ITB discharges with density control
    - Lower collisionality H-modes (ITER baseline)
- Hybrid and steady-state scenarios
    - Investigations with partial non-inductive, and approaching fully non-inductive (bootstrap + LH)

- Core and edge turbulence measurements with upgraded diagnostics; comparisons with modeling
    - — Particle and momentum transport
    - — High k fluctuations (PCI) and electron transport
- Fast particle collective modes in low and reversed magnetic shear configurations
- Accessibility to regimes with small or no ELMs
- NTM studies
    - o threshold at increased β
    - o sawtooth seed stabilization (ICRF and LH)
- Tungsten lamella divertor tile physics and technology
    - — Isotope retention (application to tritium retention in ITER)
    - — Power handling for long-pulse, relatively low density AT regimes.

Plain English Goals

- Current profile control with microwaves
- Sustaining plasma current without a transformer (50% non-inductive)
- Active density control

FY07B Program planning budget: 25 weeks research operation

Prioritized increments:

- 4 weeks additional research operation, to 19 weeks
- add 4th MW to LH system
    - o optimal use of two couplers
    - o fully non-inductive regimes at higher current
    - o approach no-wall β limits
- Real-time matching for 2 ICRF transmitters for improved utilization
- Outer divertor upgrade
    - — Power handling for >8MW, 5 seconds
- Upgrades to data acquisition and computation infrastructure
- 3 weeks additional research operation, to 22 weeks, with required additional personnel (science, engineering, technical)
- Spare 4.6 GHz, 250 kW klystron
    - — Currently no spares for 16 tube system
- 3 weeks additional research operations, to 25 weeks (full utilization), with required additional personnel

Research Highlights (See body of this Work Proposal for details)

- Substantial increased progress across all topical science areas and integrated thrusts, with particular emphasis on high priority ITER R&D and ITPA joint research

FY08D: 10% Decrement from FY07A Guidance (8 weeks research operation)

Plans

- Continue highest priority studies
    - LHCD for off-axis current profile control
        - Weak and strong reversed shear
        - Optimization of cryopumping for density control
    - Conventional H-mode studies
        - Higher current, pressure
    - Disruption mitigation
        - Some development of real-time triggering
- Progress for highest priority science topics, but at significantly reduced pace

Implications

- Layoff 2 engineers, 1 technician, 1 scientist, 1 post-doc; do not replace 2 graduate students
- Reduced runtime (5 weeks)
    - Even fewer of priority runs can be completed (to less than ¼)
- Delay 2'nd Lower Hybrid launcher (1 year)
    - Power handling capability of single LH launcher expected to limit total power to plasma
        - Less driven current
        - Harder to quantify results accurately, compare with models
- Delay completion of polarimetry diagnostic to measure j(r)
    - At least 6 month delay
- Defer data acquisition and computing infrastructure upgrades

FY08A Level: 13 weeks research operation

Prioritized increments:

- Add 5 weeks research operation and restore personnel cuts
- Complete design of outer divertor upgrade (power handling)
- Complete construction of second LH launcher (reduced power density/increased power, compound spectrum)

Research Plan Highlights (Details as outlined in the body of this Work Proposal)

- Current profile control with Lower Hybrid
    - Pulse length approaching 3 seconds (multiple current rearrangement times)
    - Power to 2 MW coupled
- Density control with divertor cryopump
- Hybrid and steady-state scenarios
- Core and edge turbulence measurements with upgraded diagnostics; comparisons with modeling
    - Electron transport will be continued area of emphasis
    - Particle and momentum transport

- Fast particle collective modes in low and reversed magnetic shear configurations
- MHD stability analysis of H-mode edge barrier under type I and tolerable ELM conditions
- Tungsten lamella divertor tile physics and technology
    - Evaluation of high power handling with total input power approaching 8 MW

Plain English Goals

- Confinement at high plasma current
- Active control of ICRF antenna

FY08B Program planning budget: 25 weeks research operation (assuming guidance budget in FY07)

Prioritized increments:

- 5 weeks additional research operation, to 18 weeks
- add 4th MW to LH system, to make optimal use of two couplers
- Real-time matching for 2 ICRF transmitters for improved utilization
- 4 weeks additional research operation, to 22 weeks, with required additional personnel (science, engineering, technical)
- Completion of outer divertor upgrade
    - Power handling for >8MW, 5 seconds
- Upgrades to data acquisition and computation infrastructure (replace CAMAC, upgrade MPP cluster)
- Spare 4.6 GHz, 250 kW klystron
    - Currently no spares for 16 tube system
- 3 weeks additional research operations, to 25 weeks (full utilization), with required additional personnel

Research Highlights (See body of this Work Proposal for details)

- Substantial increased progress across all topical science areas and integrated thrusts, with particular emphasis on high priority ITER R&D and ITPA joint research