ALCATOR C-MOD
FY09-11 WORK PROPOSAL

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Office of Fusion Energy Sciences
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Alcator C-Mod is the only 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.

1. Introduction

Organization of the program is through a combination of topical science areas supporting integrated thrusts. The topics relate to the generic fusion-plasma science, while the thrusts focus this science on integrated scenarios, particularly in support of ITER design and operation. The project is also aggressively investigating important issues on the MFE development path from ITER to DEMO. The program has five topical science areas: core transport; pedestal physics; 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. All aspects of the research are intimately connected to a broad program of theory and modeling. The connections among the topical science areas and the integrated scenarios are illustrated in Figure 1.1.

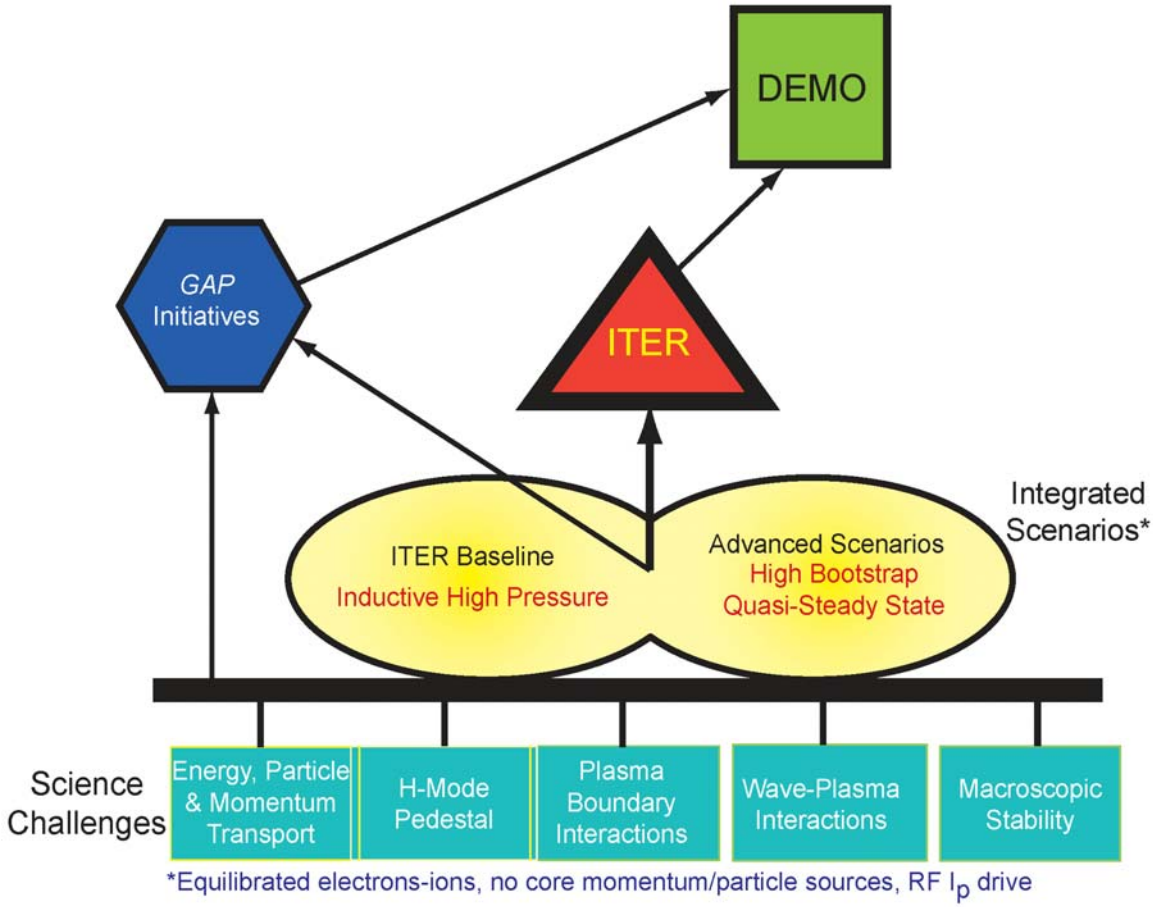

Figure 1.1 Integrated scenarios and topical science areas.

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. Combined with Lower Hybrid Current Drive for current density profile control, 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 atmospheres), 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 plasmas are dimensionally unique, but can be dimensionlessly comparable to those studied in larger tokamaks, which allows us to provide key points on scaling curves for confinement, H-mode threshold, pressure limits, etc. At the same time, joint 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 SOL 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 solid molybdenum and tungsten plasma facing components on C-Mod are unique among the world's major facilities. The use of high Z PFCs 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.

The C-Mod facility already has an impressive set of facility capabilities, control tools and diagnostics. During the next three year period, significant facility upgrades, particularly for ICRF and LHRF systems, and upgraded and new diagnostics will be implemented.

The C-Mod program is fully collaborative. In addition to MIT, which hosts the facility, major collaborations are ongoing with the Princeton Plasma Physics Laboratory and the University of Texas at Austin. Many smaller groups of collaborators at Universities and Laboratories, both domestic and international, are integral participants in the research.

Education is a very important aspect 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. At any time, about 30 graduate students are doing their PhD thesis 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 coming three year period. Many of the experiments are carried out jointly with other tokamak facilities, both in the US and around the world, with coordination through the ITPA. Major C-Mod contributions are as follows.

Integrated Scenarios, Baseline H-mode and Advanced Scenarios:

- Breakdown and current rise in ITER
- Reference set of ITER scenarios for baseline H-mode, steady-state and hybrid operation, for databases and modeling
- ITER hybrid scenarios: experimental development and understanding mechanisms for maintaining $q_0>1$
- Profile control methods: especially j(r) with combined LHCD and bootstrap

Transport

- Core transport regimes with equilibrated electrons/ions, no momentum input, dominant electron heating: regime for majority of C-Mod operation.
- Collisionality dependence of density peaking: addition of C-Mod data breaks the covariance between collisionality and n/n_G seen in other experiments; heating decoupled from core fueling.
- Develop and demonstrate turbulence stabilization mechanisms compatible with reactor conditions, such as magnetic shear stabilization, shear flow generation and q profile; compare these mechanisms to theory.
- Develop common technologies for integrated modeling (frameworks, code interfaces, data structures): MDSplus is a model.

Pedestal Physics

- Understand L-H power threshold at low density: C-Mod provides data at unique (ITER value) magnetic field; effects of neutrals/opacity.
- Role of rotation in the H-mode transition.
- Improve predictive and design capability for small ELM and quiescent H-mode regimes: small ELM regimes for $\beta_N>1.3$; effects of shaping.
- ELM control techniques: stochastic fields with external coils.

Plasma-Boundary Interactions

- Tritium retention and tritium removal: solid high Z PFCs; disruption cleaning; plasma and nuclear damage; erosion.
- Scaling present-day conditioning and operational techniques to future devices: boronization with high Z walls; ICRF induced impurity generation.
- Power handling and impurity control: SOL transport, radiative/detached divertor.

Macrostability

- Disruption database (energy loss, halo current): excellent diagnostics for radiated power, surface heating, halo currents.
- ITER applicable disruption mitigation, validate 2 and 3-D MHD codes with radiation: pioneering studies with NIMROD/NIMRAD of C-Mod experiments; LHCD for controlling seed population of non-thermal electrons to study runaway amplification/suppression.
- Develop reliable disruption prediction methods: work started on robust algorithms; real-time automatic mitigation using Digital Plasma Control System planned.
- NTM physics: effects of rotation; LHCD control/stabilization
- Understand intermediate n Alfven Eigenmodes (AE's); Damping and stability of AE's: active MHD antennas couple to intermediate n modes.
- Redistribution of fast particles from AE's: ICRF ion tails drive AE's unstable, Compact Neutral Particle Analyzers (passive and active with Diagnostic Neutral Beam), plus new scintillator lost ion detector to measure effects of AE's on fast particles.

Fusion Science Priorities

"Gap" issues on the path from ITER to DEMO

In its October, 2007 report, "Priorities, Gaps and Opportunities: Towards A Long-Range Strategic Plan For Magnetic Fusion Energy", FESAC identified 15 science and technology gaps that need to be filled on the path to designing and building a successful MFE DEMO reactor. Complete resolution of most of these issues requires both a successful ITER program, and additional initiatives, including new facilities, primarily studying D-T burning fusion plasmas. However, many of the issues related to the gaps are amenable to research on existing experiments, coupled with advances in theory and modeling. Alcator C-Mod is working on a significant number of these issues, and substantial progress is expected in the next three years, which in turn will help to inform the design of new facilities that will be needed. The areas where C-Mod makes the strongest contributions are:

- Plasma facing components: high Z metals, ultra-high SOL power densities.
- Off-normal events: disruption avoidance, prediction and mitigation.
- Plasma-wall interactions: SOL and divertor transport, erosion and redeposition, hydrogen isotope retention.
- Integrated, high performance burning plasmas: focus of the Integrated Advanced Scenarios thrust.
- Theory and predictive modeling: code benchmarking, discovery of new phenomena, iteration of theory and comparison with experiment.
- Measurements: new and improved diagnostic techniques.

- RF antennas, launchers and other internal components: Advancing the understanding of coupler-edge plasma interactions, improvement of related theories and modeling.
- Plasma modification by auxiliary systems: RF systems (ICRF and LHRF) for current drive, flow drive, instability control; ELM control.
- Control: maintaining high performance advanced scenarios.

Recent scientific achievements and research plans for FY09-FY11 are detailed in subsequent sections of this work proposal, organized by scientific topics and integrated thrusts.

Funding for the MIT portion of the C-Mod program is provided under the umbrella of a Cooperative Agreement with the Department of Energy, Office of Fusion Energy Sciences. The current five year agreement period began November 1, 2008, and covers FY09-FY13. The formal proposal for this five year period was successfully peer reviewed in the spring of 2008, and we are currently implementing those plans, subject to budget constraints.

Budget and Schedule

The baseline (A) budget for the C-Mod project in FY2009 is based on guidance from the Office of Fusion Energy Sciences, with total national project funding of $24.843M, including $20.969M at MIT, and major collaborations totaling $3.874M. These budgets will accommodate 10 weeks of research operations in FY2009. For FY10A, we have taken a flat budget (relative to the FY09 guidance we had prior to a very recent $200k reduction) with no increase for inflation. This results in a plan for 13 research weeks in FY2010. The FY10B budget assumes a 10% increment above the FY10A funding, and results in a plan for 18 research weeks in FY10. For FY11A (guidance), we assume flat from FY10A, with a 2.5% increment for inflation, with a plan for 13 weeks of research operation. A 10% cut relative to FY11A leads to a plan for 6 research weeks in the FY2011D case.

Implications of the different budgets, along with prioritized increments, are listed in Appendix C. Areas of research emphasis are also listed there; more details can be found in the March 2008 proposal for the next 5 years of research on Alcator C-Mod, covering the grant period Nov 1, 2008 through Sept. 30, 2013.

Proposed facility research run time is given in table 1.1. In addition to the guidance cases, we show the 2010B and 2011B incremental cases, and the 2011D decremental case.

Table 1.1: Research operation for guidance (08-10A), incremental (09B-10B) and decremental (10D) budget cases

Fiscal Year	**09**	**10A**	**11A**	**10B**	**11B**	**11D**
National Budget ($M)	24.84	25.04	25.67	27.54	28.23	23.10
Research Operation Weeks	10	13	13	18	18	6
Research Operation Hours	320	416	416	576	576	192

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 assumes an integrated effort involving all of the collaborators.

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.

Self-generated plasma rotation [Sept 2009]
Rotation has been found to improve plasma performance by regulating instabilities on a wide range of spatial scales. However, fusion reactors, like C-Mod but unlike most other current experiments, will not have heating systems that provide significant external torque. C-Mod will carry out experiments aimed at improving predictions of self-generated plasma rotation in ITER.

Hybrid Advanced Scenario investigation [Sept 2009]
With the implementation of Lower Hybrid RF for current profile control, and active cryopumping for density control, C-Mod will investigate advanced scenarios for improved performance of the tokamak. Investigations into the so-called "hybrid" mode of operation, being considered as one possible advanced approach for ITER, will be carried out to evaluate the potential to maintain central safety factor near or slightly above 1 and to assess the effects on plasma transport and confinement.

Testing a model of the fuel retention process in first-wall tiles [September 2010]
Tritium fuel retention is an important issue for ITER and reactors, since the on-site inventory of tritium is restricted by safety considerations. In addition, the high neutron fluence environment of a reactor necessitates the use of tungsten as the first-wall tile material. Initial experiments have revealed a higher-than-expected level of fuel retention in C-Mod's molybdenum and tungsten tiles compared to laboratory studies. It has been proposed that when such tiles are exposed to C-Mod's high plasma particle fluxes, they experience damage deep within the material, forming 'traps' that can enhance fuel retention. Experiments will be performed to quantify the level of fuel retention in C-Mod's molybdenum and tungsten tiles and, with the help of parallel laboratory

experiments, to develop and test a model for the trap formation and fuel migration that can explain the observations.

Study of runaway electron dynamics during disruptions [Sept 2010]
Disruption mitigation is a crucial issue for ITER. Viable techniques for reducing halo current forces and thermal loads to the ITER divertor have been successfully developed and tested on a number of tokamaks. However, avalanche growth of very high-energy (multi-MeV) populations of electrons (potentially carrying as much as 10 mega-amps of current) in ITER is a disruption-related critical issue that has not been experimentally studied in depth, and viable mitigation techniques have not yet been developed. We plan to investigate the use of the Alcator C-Mod lower hybrid current drive system to generate populations of non-thermal electrons as a seed for disruption runaways that can be studied using a number of specialized diagnostics on C-Mod, including an array of hard x-ray energy analyzers and synchrotron radiation detectors, with the goal of understanding runaway electron growth, confinement, and loss mechanisms. While not part of this goal, the results should be applicable to the eventual development of practical runaway electron mitigation techniques.

Characterize accessibility conditions for small edge-localized modes [Sept 2010]
Global tokamak energy confinement is determined largely by the level of transport suppression obtained in a barrier at the plasma edge. However, strong edge transport barriers often reach pressure limits that are manifested as large, intermittent losses of particles and energy from the edge. In ITER and reactors, these edge-localized modes, or ELMs, are expected to result in deleterious transient heat and particle loads on material surfaces, unless the ELMs can be made small or non-existent. C-Mod will explore accessibility conditions for small ELMs, with variations in magnetic geometry, density and input power serving as important experimental knobs. Local measurements of edge conditions, combined with edge magnetohydrodynamic stability calculations, will be used to investigate the physical mechanisms responsible for reducing ELM size, and for suppressing them altogether.

Investigate ICRF sheaths and impurity generation with an advanced ICRF antenna [Sept 2011]
Coupling high power ICRF with mimimal negative impact on the plasma performance, due to impurities for example, is challenging in tokamaks with metallic plasma facing components. One approach to reduce impurities associated with ICRF antenna operation is to reduce the RF sheaths through antenna design. We will characterize associated RF sheaths and impurity production for a field aligned ICRF antenna and compare it with a standard ICRF antenna. We will also investigate the voltage and power limits of the field aligned antenna, for comparison with a standard ICRF antenna.

Characterize the H-Mode pedestal [Sept 2011]
Currently OFES is in the process of writing a Joint Research Target for FY2011, which will involve an integrated experiment, theory, and computational effort on tokamak pedestal physics. Though the exact objectives of this target have yet to be elucidated, its general goal is to understand the physics mechanisms responsible for the structure of the

pedestal and develop a predictive capability for burning plasma devices. Likely components of this research effort include

- Collection of pedestal data on C-Mod, DIII-D and NSTX
- Comparison of pedestal structure between various H-mode regimes on the different devices
- Testing of theoretical and computational models for the pedestal structure (width, height *etc.*) against experimental data

Performing H-mode experiments on the three devices provides a broad range in dimensionless and dimensional parameters, as well as operational schemes. Important features that distinguish C-Mod in this effort include

- High B_T/R_0
- Edge neutral opacity approaching that of ITER
- Coupled electrons and ions
- Capability to heat with insignificant core particle and momentum source

Thus, C-Mod will fill a critical experimental role in this activity. As described in detail in the section on pedestal physics, we will continue our studies of pedestal transport and edge relaxation mechanisms, in particular with respect to proximity to L-H transition thresholds. We will also engage theoretical colleagues in comparisons of predictions with experimental results. Wherever possible, C-Mod data will be used to validate predictions for pedestal structure and scalings, as well as simulations of time-dependent phenomena.
We will examine the structure of the H-mode pedestal in a number of contrasting regimes (EDA *vs.* ELMy, high *vs.* low density, single null *vs.* double null). High resolution Thomson scattering and CXRS diagnostics will be used to characterize the electron and ion temperature profiles, as well as the radial electric field. All will be critical for comparison to model predictions. In ELM-free, EDA and inter-ELM periods, profile and fluctuation response will be examined in response to changes in power/particle flux in order to better understand the transport-limited profile gradients. These results in turn will be compared, where possible, with the output of pedestal simulation codes, both neoclassical and with turbulent transport included (for example, XGC0, XGC1). Comparisons of detailed pedestal structure in a transport-limited regime will also be compared with other devices. The C-Mod pedestal, without ELMs, has been shown to have clear correlations between collisionality and poloidal beta gradient. Can this trend be reproduced, and is there a relationship between some set of dimensionless parameters that unifies the results on all devices?
Gradient limits imposed by peeling-ballooning modes will be examined as well, with linear stability analysis using the ELITE code more regularly applied to C-Mod H-modes. In this way the peeling-ballooning model, highly successful in predicting the existence of ELMs on a number of devices, can be further refined. C-Mod provides discharges with naturally high diamagnetic stabilization to MHD instabilities, which adds additional information for modeling. One goal of the theoretical component of this milestone will be to generalize the promising pedestal height model of Snyder, EPED1, and obtaining additional benchmark data from C-Mod ELMy H-mode will be very helpful in this

regard. Modeling within this context will improve our understanding as well of the boundaries in operating space between regimes of different edge relaxation mechanisms (EDA, small ELMs, Type I *et al.*).

The role of neutral fueling on pedestal structure, insignificant in most C-Mod H-modes, will be assessed carefully in low-density discharges. In high-density H-modes, experiments attempting to reproduce the edge neutral opacity of ITER will examine the real limits of edge fueling, and look for a pedestal particle pinch. Such a pinch, if significant, would greatly impact models for pedestal height on ITER. This work will be aided by the enhancement of both edge Thomson scattering and diagnostics for measuring line emission of fueling neutrals, providing poloidally separated neutral density profiles. Modeling of the edge neutral sources will also be expanded beyond simple 1D cases previously examined on C-Mod.

A possible factor in determining pedestal width is the level of magnetic shear present in the pedestal region. This is poorly characterized experimentally, but experiments on C-Mod will attempt to vary edge shear systematically with strong variation in both shaping and q. As more LH power becomes available late in this proposal timeline, we will attempt to drive current near the edge to modify dq/dr externally. The LH system can also be used for electron heating, which will allow a critical assessment of pedestal scalings as a function of the fraction of power going into the electron channel.

Using our ICRF heating tools, we have reasonable flexibility for initiating H-modes in a wide variety of regimes, as needed for projecting to ITER. As indicated in the pedestal section of this proposal, H-mode studies will expand to include helium, and perhaps hydrogen, plasmas, as well as discharges with significant dI_P/dt.

Goals Accomplished in FY2008

Achieve research operating time of 15 weeks (±10%) [September 08] (JOULE milestone).

Alcator C-Mod operates on an 8 hours/day, 4 days per week schedule. One research week corresponds to 32 hours of facility operation.

The FY08 campaign concluded on May 23, 2008, with 15.7 research weeks accomplished. Quarter by quarter run statistics can be found at http://www.psfc.mit.edu/research/alcator/facility/Operations/FY08_research_table.html and links to details about each run day can be found at http://www.psfc.mit.edu/research/alcator/program/cmod_runs.php.

Confinement at High Plasma Current [September 08]

The operational space of C-Mod in the plasma current range at and above 1.3 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.

This goal was completed in May 2008. Discharges with plasma current in the range 1.2 MA to 1.4 MA were studied, with high power ICRF (P>3 MW). These included lower single null (LSN), near double null (DN), and upper single null (USN) divertor configurations, all with downward **B** X ∇**B** drift direction. The LSN and DN discharges transitioned to H-mode, and generally exhibited transiently good confinement; however, at these high currents, the H-modes were ELM-free, and impurity build-up (primarily molybdenum), led to large radiated power increases, decreases in confinement, and eventually back transitions to L-mode. It will be interesting to revisit these regimes with the new advanced field aligned ICRF antenna (FY10), combined with thick boron coated tiles (installed in FY09), to see if the impurity effects can be ameliorated. During the course of these experiments, in the USN configuration (the so-called unfavorable direction for the H-mode threshold), we explored a new regime for C-Mod, enhanced L-mode. In many of these discharges, a strong edge temperature pedestal develops, but the edge density pedestal remains relatively weak, core density does not increase, and impurities do not accumulate. Global energy confinement is excellent ($t_{ITER\text{-}98\text{-}Y2} \sim 1$), and there are no ELMs. Figure 1.2 shows the time histories of representative plasma parameters for an example of an enhanced L-mode discharge. The input powers in these plasmas can be as much as 3 times the conventional scaling for the H-mode threshold, while the transition to a conventional H-mode, with the development of a strong density pedestal, can be delayed by up to 10 confinement times. If this transition does occur, the subsequent discharge develops along the same lines as LSN H-modes, with impurity accumulation. By reducing the input power somewhat, we have found regimes where the particle transport barrier does not develop for the entire duration of the heating pulse,

(~20 t_E, ~2 $t_{resistive\text{-}skin}$). In future experiments, it will be interesting to examine edge fluctuation characteristics during the enhanced L-mode phase, to try to elucidate the key mechanism(s) which lead to a strong energy edge barrier, while particle transport remains L-mode like. If this regime can be better characterized and understood, it might be feasible to apply it to future experiments, including ITER.

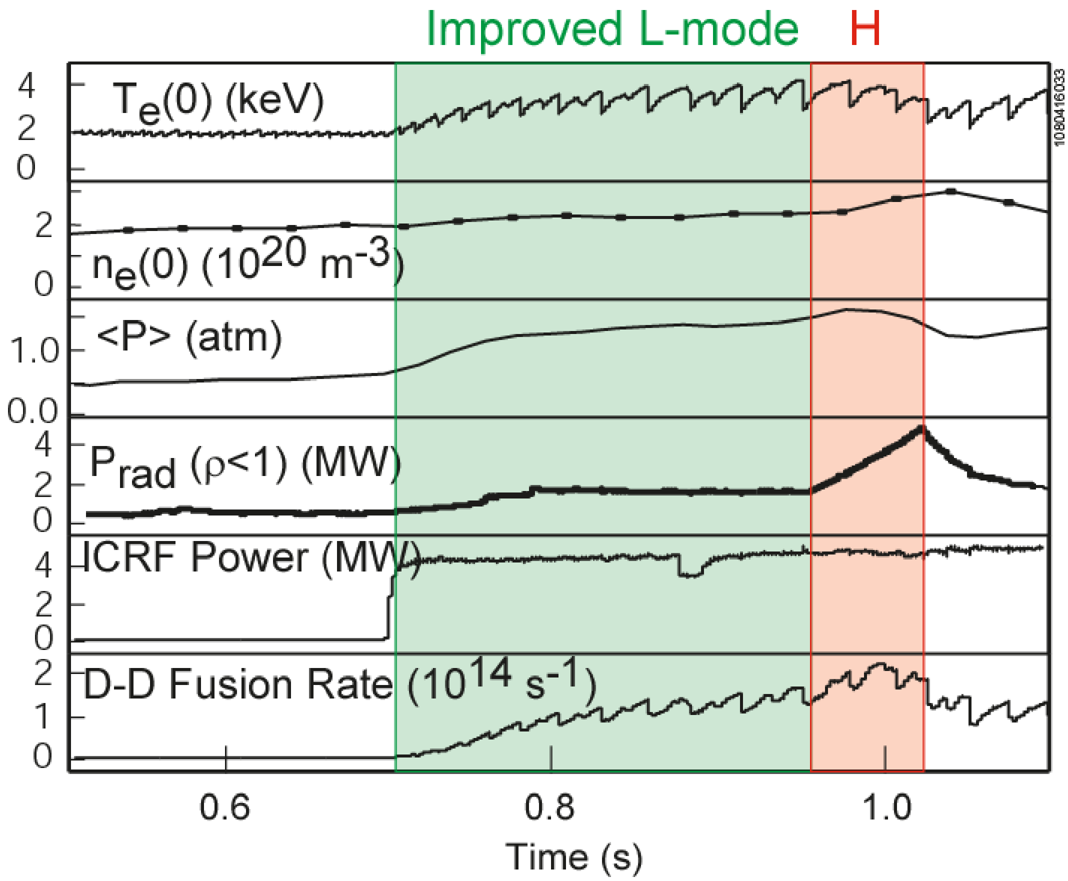

Figure 1.2. Time histories of typical plasma parameters during a typical I_p = 1.4 MA improved L-mode discharge. Note that plasma density and radiated power are well controlled during the improved L-mode phase.

Active control of ICRF antenna [Sept 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.

This goal was completed in March, 2008. A real-time ICRF antenna matching system has been successfully implemented in the E-port antenna matching network. The network is a triple-stub tuning system working where one stub is fixed and a pre-matching stub and the other two stubs use fast ferrite tuners (FFTs) to accomplish real-time matching. The system utilizes a digital controller for feedback control (200µs per iteration) using real-time antenna loading measurements as inputs and the coil currents to the FFT as outputs. The system has achieved and maintained matching, 1% power reflection, for a large range of plasma parameters, including L-mode, H-mode, and plasmas with edge localized modes (maximum mismatch of <15% in power). An example of the system's capability to maintain match during for an ELMing discharge is shown in Figure 1.3. For this prototype system, the speed is limited by the digital controller speed of 200 µsec. The system has succeeded in delivering up to 1.85 MW into H-mode plasmas at maximum voltage of 37 kV (see Figure 1.4) on the unmatched side of the matching system. A much more detailed write-up of the results of these experiments can be found in [Y. Lin, A. Binus, and S.J.Wukitch, Fusion Engineering and Design 84 (2009) 33–37].

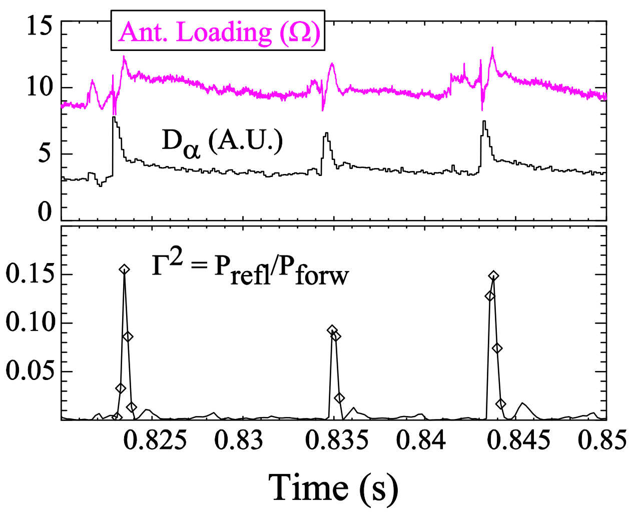

Figure 1.3: *Typical FFT load following during an ELMing discharge where the reflected power is maintained <15%.*

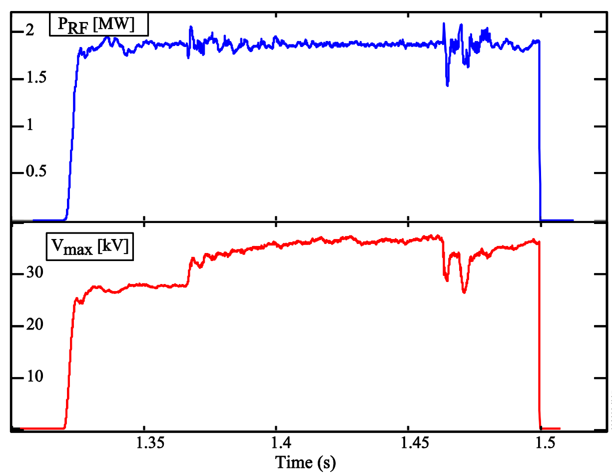

Figure 1,4: *Demonstration of high power operation with FFT where 1.85 MW was injected and the maximum voltage on the antenna is ~37 kV.*

2. H-mode Integrated Scenarios – ITER Baseline

This research activity includes experiments and modeling aimed at supporting and optimizing the baseline ITER operating scenario, generally cutting across multiple science topics and often involving interaction and compatibility issues between different plasma processes or regions. It *integrates* work described in the 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. The overall theme of the program is to support development of the ITER H-mode (baseline) Scenario, by demonstrating operating regimes with relevant plasma parameters and control tools. The goal is to establish the physics basis required to extrapolate from present-day experiments to ITER.

This effort is complementary to the "Integrated Scenarios – Advanced Regimes" task described in the next chapter. 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 *Conventional H-Mode "Baseline" Scenario* (ITER Scenario 2) is relied on to provide the target Q=10 fusion performance. The operating point 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, β_N=1.8 and density ~ 10^{20} m^{-3}, all similar to current C-Mod values.

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 useful goal to focus attention on the challenging combination of conditions which must be simultaneously met on burning plasmas.

2.1 Recent Research Highlights

Integrated H-mode scenario studies in C-Mod have resulted in improved plasma performance, enabling operation in regimes highly relevant to ITER. Increases in stored energy, 250 kJ, and volume averaged pressure, <P> = 1.8 atm, were achieved with 5 MW of ICRF, shown in Figure 2.1. 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.

Experimental work in the Integrated Scenarios on C-Mod includes support for ITPA/IEA Joint Experiments, in conjunction with U.S. and international collaborators at NSTX,

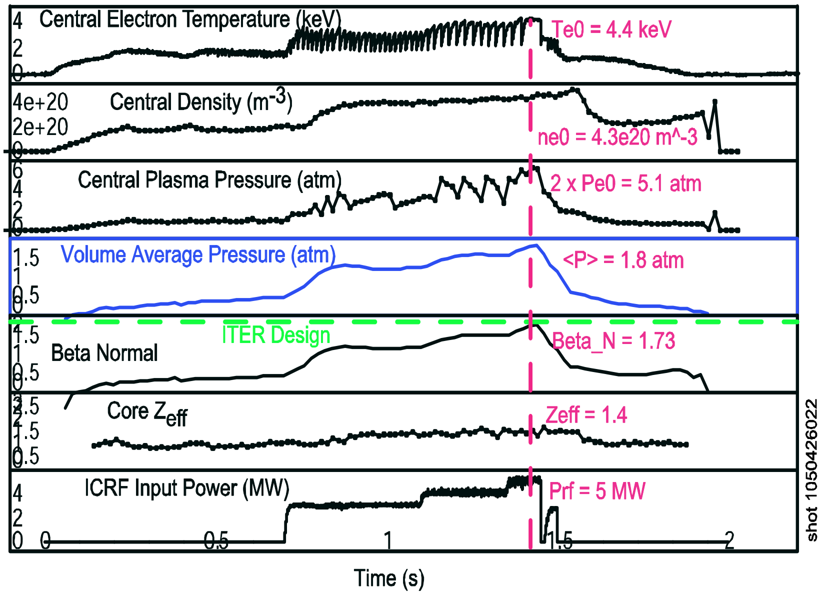

Figure 2.1 *High performance C-Mod H-mode plasma with world record plasma pressure <P> = 1.8 atm, at values of toroidal field, 5.4 T, and normalized beta, β_N=1.73, very close to those on ITER.*

DIII-D, JET, MAST, and ASDEX-UG. These experiments exploit the high leverage provided by the unique C-Mod parameters for non-dimensional scaling studies.

The C-Mod program is also addressing issues raised during the ITER Design Review. An example is an experiment concerning the scaling of the L-H threshold power at low density (Issue cards LH-2 and AUX-11). As noted on C-Mod several years ago, and subsequently on other facilities as well, the power required to access H-mode increases rapidly below some critical density, deviating from the general trend toward lower power as the target density is reduced. The ITER H-mode scenario is predicated on accessing H-mode at relatively low density, around 2×10^{19} m^{-3}. However, the scaling of the low density bound is uncertain, and if the minimum in the power versus density relation is in fact near 5×10^{19} m^{-3} then the planned heating power in ITER could be insufficient to achieve H-mode operation. This issue is of critical near-term importance, since it impacts the specification of the ITER heating systems. Experiments on C-Mod investigated the question of whether the low density limit scales with plasma current, *e.g.* as n/n_G. Such a scaling would be favorable for ITER, which proposes to access H-mode at a value of n/n_G above that which corresponds to the low density limit in C-Mod, at the same toroidal field. However, these experiments appear to rule out a scaling of the low density bound with current or Greenwald parameter, leaving the relevant scaling still to be determined. An active ITPA Joint Experiment (TC-3, formerly CDB-11) has been initiated on this subject.

ELM studies undertaken during 2008 included investigation of the use of the C-Mod external non-axisymmetric field coils to modify or suppress type I ELMs using n=1 resonant magnetic perturbations. This work was motivated by results from JET[1] which reported modification of the ELM frequency by use of the error field correction coils (EFCC). The hope was that this technique might enable the use of the external error field correction set on ITER to also serve the purpose of ELM suppression. The C-Mod experiments were carried out with field amplitudes corresponding to a Chirikov (island overlap) parameter just over unity in the outer 10% of the profile, based on vacuum field superpositions. No clear signature of ELM suppression or modification was observed in the C-Mod experiment, which may indicate that the available perturbation amplitude was insufficient or the mode spectrum was not optimized. Subsequent developments have led to the adoption of internal ELM control coils, with capability for higher n perturbations, as part of the ITER baseline design. Additional recent ELM studies on C-Mod have included continued investigation of small ELM regimes[2] in conjunction with the NSTX and MAST, under the auspices of the ITPA Joint Experiment PEP-16.

C-Mod experiments begun in 2008 as part of ITPA Joint Experiment SSO-5 addressed issues related to the ramp-up and ramp-down experiments in ITER. ITER requires routine operation at 15 MA within the operational constraints of the device. The original proposed poloidal coil-set was specified only for rather low plasma inductance (l_i=0.7-1.0). In C-Mod, the current rise and current decay phase of the ITER discharge scenario have been studied, trying to keep l_i low. The experiments used early X-point formation during the current ramp-up, and remained diverted during the current ramp-down. Ohmic ramp up discharges achieved li(3) as low as 0.9 at the start of the flat top, and ICRF heating during the rise achieved only modest reduction in l_i, despite approximately doubling T_{e0}. Analysis of the ramp-up phase using TSC/TRANSP was carried out using the experimental PF coil currents and density evolution; the simulations also used the experimental radiated power profiles. Comparison of two transport models, Coppi-Tang and GLF-23, indicated that each provided satisfactory overall agreement with the experiment, with the GLF-23 (with a suitable boundary condition) giving a better representation of the li(3) evolution and sawtooth onset times.

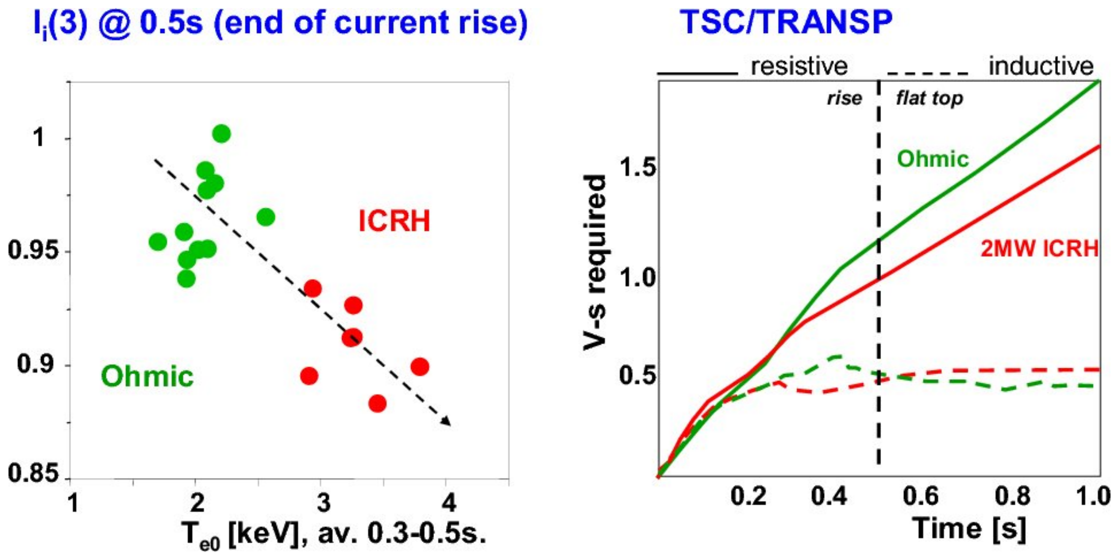

Figure 2 *Result of heating during diverted phase of ITER-like ramp-up scenario. Reduction in l_i (left) is modest. TSC analysis (right) indicates main saving is in resistive flux consumption.*

Initial experiments modeling the ITER ramp-down phase were also carried out, using ramp-down rates of 1, 2, and 4 MA/sec, starting from ITER shape single null discharges with $q_{95}\approx3.2$. The plasma remained diverted through the rampdown. Reduction of the elongation from 1.8 to 1.4 was used to maintain vertical stability. For the slowest rampdown rate, $l_i(3)$ could be kept below 1.2 during the first half of the current decay, at the cost of some additional flux consumption from the transformer. Further experiments to help validate the ITER ramp-down scenario are planned for 2009.

2.2 H-Mode Scenarios Research Plans

A major emphasis of the C-Mod H-mode Integrated Scenarios research program will be to address important issues related to ITER construction and operation. These include:

- Compatibility of core and boundary, extending beyond the last closed flux surface to open field lines
- Interaction with plasma-facing materials, including heat-flux and particle control
- Control of the operating point, and also of the startup and approach sequence and shut-down phase

The program builds on previous results, as well as exploiting newly developed C-Mod capabilities, including the cryopump for particle control as well as enhanced diagnostics.

The C-Mod research program in this area is embedded in a world-wide effort in support of ITER. Continued participation in ITPA Joint Experiments and High Priority Tasks will leverage C-Mod's unique parameters and provide valuable input toward the development of improved physics understanding and predictive capability for ITER. Coordination and collaboration activities are being pursued among the major U.S. facilities, as well as within the world program. The scope of these activities ranges from closely coordinated Joint Experiments to complementary or related experiments on multiple facilities, and includes shared development and exploitation of tools and methods.

Modeling, Analysis and Simulation tools

Modeling and analysis codes are key to Integrated Scenarios research, and development, benchmarking, and validation of these tools is both a goal and a requirement for this effort. In simulations supporting this research, a code such as TSC, which contains a model of the tokamak actuator systems and geometry, is used to advance the tokamak discharge subject to transport estimates from transport estimates of varying degrees of sophistication, ranging from empirical transport coefficients to physics-based models embodied in codes such as TGLF or GLF23. Transp runs are used to compute source terms due to ICRF (using TORIC-FPP module) and LHCD (presently employing the LSC code module, to be augmented by more advanced modules incorporating CQL3D-GENRAY). The resulting source terms are then employed to time advance the TSC simulation. Such simulation procedures are employed both for experimental design and for post-experiment interpretation and model validation.

Design of individual experimental discharges studied in the Integrated Scenarios program benefits from the use of the Alcasim code[3], a MATLAB-Simulink application (developed by an MIT Nuclear Science and Engineering Department graduate student) which incorporates details of the C-Mod magnetics diagnostics, power supply characteristics, and digital plasma control system (DPCS). The time evolution of plasma parameters are taken from experimental data from actual or simulated discharges, and can be modified by the user through a graphical interface. This code is coupled directly to the C-Mod Plasma Control System Operator interface, so that discharge programming can be evaluated off-line (software-in-the-loop), or even between plasma shots during an experimental run. This tool is also useful for developing and debugging new plasma control algorithms prior to on-line testing in C-Mod experiments.

In addition, sophisticated simulation models in standalone codes such as NIMROD, ELITE, and M3D in the MHD area, AORSA, TORIC, CQL3D, GENRAY in RF, XGC, GS2, and GYRO for Transport, are used for analysis of experimental results in these areas, and to provide guidance for experiments. Going forward, the Integrated Scenarios program will both benefit from and inform the development of the integration of such modules planned for the IPS (Integrated Plasma Simulator) being developed through the SWIM Prototype Fusion Simulation Project, as described elsewhere.

ITER H-mode Operational Scenarios

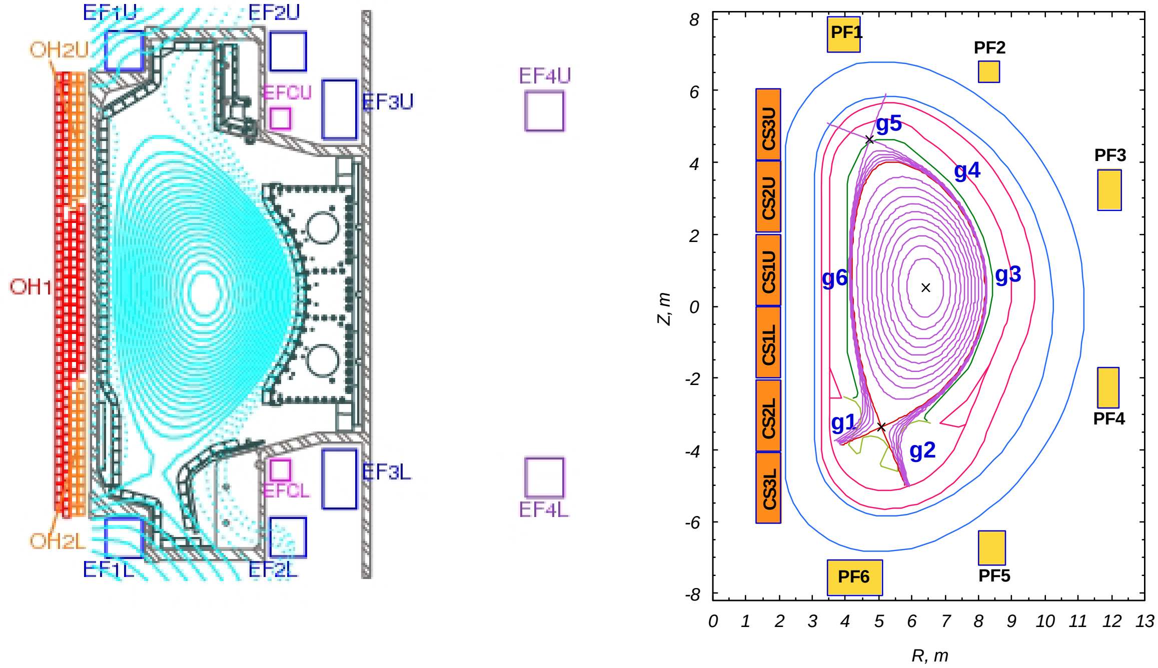

Figure 2.3. *C-Mod (left) and ITER (right) poloidal field coil sets have similar arrangement and functionality. Shape control depends on combinations of coils to maintain a small number of gaps, with minimal null space.*

A major focus of the H-mode Integrated Scenarios research program will be aimed at development, demonstration, and validation of operational scenarios for ITER operation. The C-Mod physics regime, machine capabilities and control tools are highly relevant to these tasks. In addition to operation at the ITER field of 5.3 T, the C-Mod experiments feature high ohmic power and ITER-relevant auxiliary heating (ICRF, LH), metallic walls and conducting structures, and similar PF coilset and control issues. The goal of these experiments is to provide input for benchmarking the ITER simulations, particularly the transport assumptions, as well as evolution of impurities, heat loads, and electron density.

During the next two years we will be continuing the exploration of robust ramp-down and burn termination sequence for ITER. These experiments will in part be carried out as part of the new ITPA Joint Experiment IOS-2.2. The experiments will explore a range of ramp-down scenarios starting from an ITER base-line demo plasma (q_{95}=3, ITER shape). The aim is to qualify strategies to and stay within (k, l_i,q_{95}) stability envelope of ITER, while maintaining adequate control of particle and power exhaust. The requirements for shape control during ramp-down required to maintain vertical stability will be documented. The merits/demerits of staying in H mode or L mode during the ramp-down will be investigated, including especially the issue of avoiding the density limit during ramp-down in H-mode. Heating requirements during ramp down and compatibility with exhaust power control will also be evaluated.

In the longer term, C-Mod experiments will address aspects of the approach to and control of the nominal operating point. As noted previously, the layout of the C-Mod poloidal field coilset is rather similar to that of ITER. The central solenoid is used not only for inductive drive but is composed of separate coils which play an important role in plasma shaping. The PF ring coils are distributed around the poloidal crossection, utilizing essentially all the available space not required for port access. There is no strict one-to-one correspondence between individual PF coils and the plasma shape parameters of interest, resulting in shape (and current) control algorithms that are inherently MIMO (Multiple Input, Multiple Output) in character. C-Mod is therefore an appropriate test-bed for development and testing of control approaches and algorithms in a realistic tokamak environment, including the effects of noise, parasitic currents in conducting structures, effects of power supply nonlinearities, transient events, *etc.* These experiments also serve to benchmark the simulation codes used to design and validate control approaches for the ITER system.

Other H-mode experiments with direct relevance to ITER burning plasmas include burn control simulations. The digital plasma control system gives the capability to vary input RF power as a function of plasma temperature or neutron production, mimicking the alpha heating in a burning plasma. This will allow us to study the evolution and stationary states of a self-heated plasma, and to develop and test burn control techniques. These experiments will also provide a good test of the ability of control algorithms to deal with stable as well as potentially unstable operating points. The goal would be to demonstrate the ability to maintain constant "fusion power" in the presence of perturbations such as ELMs, sawteeth, MHD instabilities, density excursions, *etc.*

In addition to maintaining a potentially unstable operating point, burn simulation experiments would also address the issue of access to the operating point and burn termination. Of particular interest is to simulate the entry to the H-mode phase including (simulated) alpha-heating, in view of the small margin of the ITER auxiliary heating system with respect to the H-mode threshold. The safe termination of the burn and subsequent ramp-down and plasma termination is also an important topic for these simulation studies.

Support for ITER Pre-nuclear Phase

The first two and a half to three years of operation in ITER will be restricted to operation in hydrogen and helium, prior to the issuance of a regulatory for nuclear operation. During this period the initial complement of heating and current drive sources, presently scoped to comprise a total of 73 MW including NBI (negative ion source, operating in H), ICRF (40-55MHz) and ECRF(170GHz). Plasma operation would be brought up to "full parameters" in a staged manner, including TF operation initially near half-field (2.6-3T) progressing to full 5.3T operation. Plasma current would be raised in prudent steps to the design value of of 15MA (at 5.3T). The objectives of this plasma commissioning phase are to demonstrate facility operation; commission heating and diagnostic systems with plasma; commission control and safety systems; **demonstrate plasma operation to full technical performance;** demonstrate critical system performance, *e.g*. divertor loads; validate licensing assumptions concerning disruptions; characterize operational boundaries, and off-normal events; demonstrate, *to the extent possible,* plasma performance and scenarios envisaged for DD and DT operation; characterize hydrogenic retention and demonstrate techniques applicable to tritium recovery. In order to qualify operational techniques, including ELM control, fueling, and power exhaust control, it is considered highly desirable to have access to H-mode operation during the pre-activation phase. In addition, the present schedule calls for initial operation to employ the CFC divertor hardware, with a changeout to W divertor to take place prior to first DT operation. The earlier data and operational experience can be obtained to support the divertor change, the more efficient the transition to DT operation will be.

The major issue confronting the pre-nuclear operation is the question of access to H-mode. Based on presently accepted scaling, the initial power complement will be insufficient to achieve and sustain H-mode in hydrogen. It is believed that operation in helium may provide this access. Supporting research to improve the extrapolation of the H-mode threshold in helium, and to characterize the relevance of the helium H-mode behavior, particularly with respect to ELM phenomena, is requested by the IO. Additional issues include validation of plasma control, including vertical stability, at higher q values and in L-mode plasmas, which may be encountered during the plasma commissioning and discharge development sequence, and the use of atypical rf heating scenarios, including second harmonic minority (He^3) heating with either H or He^4 majority, which are required for half-field operation.

C-Mod research will address these issues by carrying out experiments on the isotopic scaling of the L-H transition threshold, particularly in He as compared to D plasmas, using the same heating technique (ICRF) for each. This activity is undertaken as part of ITPA Joint Experiment TC-4. Further work on C-Mod will endeavor to characterize the He H-mode discharges, including assessment of confinement quality, edge relaxation phenomena, SOL and divertor properties, and compatibility with RF heating. Experiments will be carried out in ITER-like shapes and parameter ranges, and also in shape conducive to observation of type I ELMs in C-Mod, since a key question is the suitability of helium H-modes in ITER to test ELM control techniques.

High performance Demonstration Discharges

Demonstration of high performance discharges with the ITER baseline non-dimensional parameters (excluding ρ^*) is in itself a challenging task which serves to increase confidence and provide benchmarking for models and codes used to develop and refine operational scenarios for ITER. Moreover, these demo discharges provide a platform for other ITER-relevant physics studies, and a reference comparison for more advanced scenarios. Issues to be addressed include optimization of pedestal characteristics for best core reactivity, while maintaining satisfactory divertor parameters and particle and energy exhaust. This work will apply and *integrate* the results from research in the Pedestal Physics, Core Transport, MHD, and Boundary Science programs.

Building on the successful high-power H-modes already demonstrated , with ITER field and β_N, H-mode high-performance scenario experiments will be extended toward conditions which are closer to those on ITER in other respects, especially collisionality and normalized current (I/aB). We have demonstrated ITER-shaped ohmic and ICRF heated discharges with $\kappa>1.8$ and $q\leq 3.2$ at the nominal ITER field of 5.3 T. This discharge shape is well-matched to the cryopump configuration. We will operate in this configuration at high ICRF power (P>5MW) using D(H) minority heating with f≈80MHz, which provides high single pass absorption, comparable to the ITER ICRF heating scenario. This should provide an expanded parameter space for databases and extrapolation to ITER, as well as demonstrating operation at the ITER field, q_{95}, β, and absolute pressure. Using the cryopump to reduce density below that set by natural evolution from the L-mode target, the resulting increase in T_e should cause a substantial reduction of ν^*. These experiments will provide integrated tests of confinement, heating and power handling in a highly ITER-relevant regime. In addition to enhanced $D_{\alpha,}$ pedestal and confinement studies will focus more on the regime of small ELMs , 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} ~3.2.

Because ρ^* is not matched in these "demonstration discharges", no single measure of collisionality is adequate to characterize all the relevant physical processes. It is therefore not possible in general for a single set of parameters to serve as the basis for extrapolation to ITER. Typically different phenomena will be best addressed at different absolute parameters in a given device, and a range of "collisionalities" is required for ITER H-mode validation experiments. Some transport effects may be well-characterized by the usual neoclassical $\nu_*^{neo} = (\varepsilon^{-3/2}\nu_{ii}\ q\ R/v_{thi})$. Others, along with NTM physics and other MHD processes, will depend on ν/ω_*, which, for fixed q and geometry, is larger than ν_*^{neo} by a factor of ρ^*. Electron-ion equilibration depends on $\nu^{e/i}\ \tau_E$, which in turn depends on the ρ^* scaling of the transport. For gyro-Bohm scaling, $(\nu^{e/i}\ \tau_E) \sim \nu_*^{neo}\ (\rho^*)^2$, *i.e.* two powers of ρ^*. Figure 2.4 shows example parameters of C-Mod discharges which match the ITER H-mode operating point in $\beta_{N,}$ q, shape, and each of these three measures of collisionality, as well as having the same field and therefore the same absolute pressure. An additional measure, not strictly related to collisionality (and not strictly conforming to the paradigm of plasma physics non-dimensionality) is the ratio of density to the empirical (Greenwald) density limit, which is also shown in the figure, This parameter serves to delimit the accessible operational range, and may also be relevant to the physics of the SOL/divertor interaction and questions of exhaust power handling. The nominal ITER H-mode operating point is at the upper end of the range in this parameter, n/n_G=0.85.

As shown, under the given transport assumption (H89=2), between 5 and 6 MW of heating power should be sufficient to access the ITER β value over the range of required density; this power is consistent with C-Mod's installed ICRF power. Typical density in high power H-mode discharges in C-Mod are in the range of 3 to 5×10^{20}m^{-3}, corresponding to the upper two collisionality matching points. High power H-mode operation in C-Mod at q_{95}~3 and a density of 2×10^{20}m^{-3}, as required in order to match the projected ITER neoclassical ν_*, is a challenge, both in terms of accessing the low-density condition and for maintaining a quasi-steady discharge at low Z_{eff} and P_{rad}. Furthermore, these low density conditions provide the most severe challenge to the divertor and plasma-facing components. The other end of the normalized density range, to n/n_G=.85, is also a challenging target in C-Mod, both because of the difficulty of gas fueling to such high density and due to technical issues with high power ICRF operation associated with the attendant high neutral pressure at the antenna. Extension of the target density for ITER demonstration discharges to this values of n/n_G is part of the prescription for ITPA Joint Experiment IOS-1.1. The C-Mod research plan in this area is to carry out a systematic program of experiments covering the accessible range of density, documenting all the relevant aspects of plasma and divertor performance.

In addition to operation at the ITER field of 5.3 T, some high performance H-mode experiments will be carried out at 8 T, using the $D(He^3)$ heating scenario at f~80 MHz. These experiments, while at higher than the ITER field, extend ρ^* scalings, and facilitate important ITPA intermachine experiments with larger, lower field tokamaks including JET, DIII-D and Asdex Upgrade

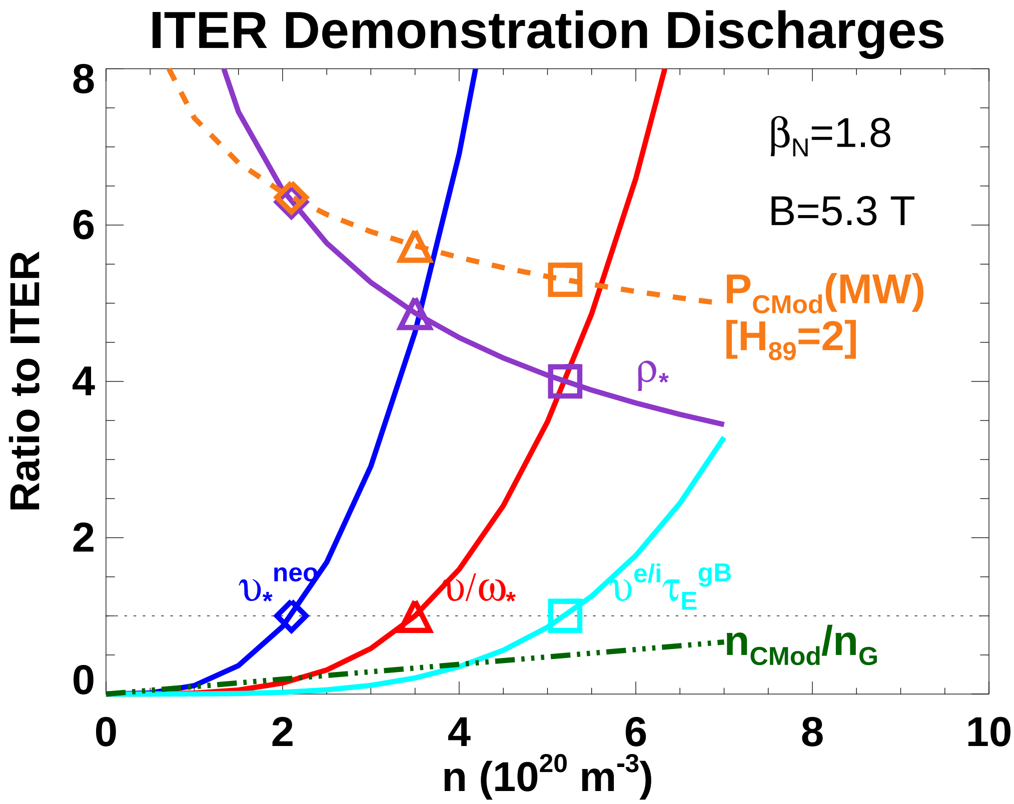

Fig. 2.4 Parameters of ITER H-mode demonstration discharges in C-Mod which match ITER in terms of three dimensionless measures of collisionality. Vertical axis corresponds to the ratio of the C-Mod parameters to the ITER nominal values for ρ^* (purple) and each collisionality parameter. Also shown are the power (MW) required for C-Mod (yellow) and the Greenwald density parameter of the C-Mod discharge.

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/ω_*, C-Mod will be positioned to provide important tests of NTM thresholds and RF stabilization techniques. Investigation of the use of LHCD for NTM stabilization by means of Δ' modification, initially being carried out by the MHD Physics topical area, will, if successful, be developed as a component of an integrated scenario for application to the ITER H-mode baseline case. Accessibility constraints will likely require these experiments to be carried out at densities close to those of ITER, around $1\text{x}10^{20}$ m^{-3}.

It is perhaps worth noting that the issues and challenges that must be met for C-Mod to access this regime are similar to those faced by ITER. In both cases, the divertor and plasma facing components must deal with heat fluxes near the limits of the materials, and the impurity control must be compatible with maintaining low Z_{eff} and low radiated power from the core. The issue of hydrogenic retention in plasma facing components must be met in each case, in ITER because of tritium issues and in C-Mod because the

ICRF proton minority heating depends on a low hydrogen fraction in the plasma. C-Mod activities in response to this challenge are reported in the Boundary Physics chapter. In C-Mod, ICRF minority heating is employed as the primary auxiliary heating source to access and sustain the high performance plasma regimes of interest. For ITER ICRF is also employed as part of the complement of bulk auxiliary heating required to access the burning plasma regime. The devices face similar challenges in terms of RF power density at the antennas, compatibility of wave accessibility with the H-mode edge pedestal and ELM perturbations. These loading issues are being addressed at C-Mod through design of the ICRF antennas, and by utilization of a combination of active and passive load matching components, as described in the ICRF chapter. C-Mod operation has also identified an issue concerning interaction of the ICRF waves with high-Z metallic plasma facing components, as well as erosion of low-Z films used for wall conditioning.

Pedestal and ELM Control

For the H-mode baseline scenario, as for all scenarios with an H-mode edge transport barrier, control of the pedestal parameters and edge relaxation phenomena is crucial, both for setting the boundary conditions for the (stiff) core transport and for particle and impurity control. A particular challenge for C-Mod in this respect is in accessing regimes with sufficiently low collisionality. We propose to utilize the cryopump to extend our exploration of pedestal structure and edge relaxation mechanisms to lower collisionality regimes. The H-mode pedestal height is critical for determining the core profiles and confinement in ITER. Recent work, described in the Pedestal Physics section, extended C-Mod H-modes to lower ν^* under specific circumstances, including high field (~8 T), unfavorable grad-B drift direction, and strong shaping. These regimes were characterized by higher pedestal T_e and lower pedestal density than our typical H-modes. Some of these cases also exhibited very steep pressure gradients. We will be continuing and extending our studies of these regimes, making use of the cryopump for additional particle control.

Experiments on C-Mod in the last few years have identified a regime of low-density H-modes with large (type I) ELMs. These plasmas were characterized by strong shaping, in particular by large values of lower triangularity. These experiments provided an opportunity to contribute to studies of ELM structure, dynamics and energetics. Using the cryopump, we will extend these studies to investigate ELM stability as a function of shape and collisionality. In particular, we should be able to pursue these studies in the ITER shape. Specific topics of interest include the further investigation, with improved diagnostics, of the high frequency magnetic oscillations observed at filament ejection, and the non-thermal electron generation associated with the ELM crash.

Power and Particle Exhaust – Plasma-Wall interaction

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. Research will extend to include testing of new tungsten tile designs over a greater portion of the PFCs. As described in the Boundary Physics section, a complete toroidal band of tungsten lamellae tiles is presently installed in the outer divertor. 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. The outer divertor modules are scheduled to be replaced in 2011 with a new structure featuring tungsten plasma facing components based on the presently installed lamella design. The new structure will be fully axisymmetric, with no leading edges, further increasing the power handling capabilities. In addition this divertor structure will be capable of DEMO-relevant operation at elevated temperature, up to 600C, providing improved hydrogenic retention properties.

We will continue to explore the prospects for radiative divertor H-mode scenarios on C-Mod at higher power than was available in past investigations. Moderately 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 in conjunction with this edge and divertor radiation. Recent experiments have demonstrated the ability of the cryopump to pump injected radiating gases, allowing for additional control of the radiation fraction. Making use of this capability, along with the improvements in the plasma control system made possible by DPCS, we propose to revisit and extend our previous detached divertor control experiments[4] to ITER-like conditions. Puffing of impurity gases which radiate primarily in the divertor and SOL regions will be employed to increase the divertor radiation fraction and decrease the heat flux conducted to the divertor plates. These experiments will contribute to ITPA Joint Experiment IOS-2.2. Feedback on the puff rate together with pumping should enable us to maintain a constant ratio of edge radiation to total power. Upgrades to the divertor bolometry and thermography will facilitate these studies, as will the improved symmetry provided by the new divertor structure once it is installed.

Control Algorithm Development and Validation

Another aspect of Integrated Scenarios research which will be receiving increasing attention at C-Mod and other facilities as ITER moves into the construction phase is the development and demonstration of robust machine protection algorithms and techniques. While development of these methods at C-Mod will be carried out initially under the auspices of the Operations and Control System tasks, incorporation of such techniques into routine operation at relevant parameters will be required. Robust fault sensing algorithms capable of identifying off-normal or unplanned conditions near operating boundaries must be developed, validated, and demonstrated. The range of off-normal conditions which must be identified include failure of control system sensors, proximity to actuator limits, such as power supply voltage or current saturation, actuator failure, unanticipated variation from expected plasma behavior, and loss of plasma stability tending toward disruption. For each such condition reliable detection algorithms must be identified, with satisfactory look-ahead to enable appropriate remedial action and extremely low susceptibility to “false positives”.

One example of such machine protection actions is disruption mitigation using massive gas jet injection to ameliorate the effects of VDE’s, as described in the MHD section. Experiments on C-Mod and elsewhere[5] are demonstrating the efficacy of the technique, and optimizing the amount and mixture of injected gas. A trigger algorithm based on detecting an incipient VDE by observing the amplitude of the error in the vertical position has been tested successfully[6]. Adequate time for actuation of the gas jet was demonstrated with a trigger threshold that should not lead to termination of controllable discharges, based on a database survey. However, implementation of the mitigation system using this algorithm has not been attempted on a routine basis, since on C-Mod the deleterious effects of unmitigated VDE's have not been considered sufficiently serious to warrant this step. In order to develop the database of experience required to qualify this technique for application on ITER, we propose to incorporate this algorithm into standard operation on C-Mod. Routine use of disruption mitigation will provide valuable operational experience leading to further optimization of the technique, as well as offering the potential for improved disruption recovery on C-Mod.

Another aspect of machine protection algorithms currently under development is a transition to an alternate, "safe" equilibrium trajectory, followed by graceful discharge termination, in response to power supply saturation. Such solutions have been proposed, and tested in simulations[7,8], but routine application of such nonlinear control methods in actual experiments is lacking. C-Mod is a suitable test-bed for such a scheme, since current saturation is frequently encountered in operation, particularly near the end of flattop or in the ramp-down phase of discharges. The MIMO linear control scheme employed at C-Mod (7 shape parameters, plus plasma current; 9 independent power supplies) allows for only a rather small null space, rendering alternate solutions to the saturation problem without abandoning the original target equilibrium problematic. Similar considerations may be expected for the ITER shape control. Important issues for the design of such an adaptive response include the identification of appropriate alternative "safe" fallback equilibria, and the development of a smooth interpolation procedure between the original targets and the fallback. Successful implementation of such an adaptive algorithm for dealing with impending actuator saturation was demonstrated[9] in C-Mod for a specific equilibrium in 2008. In the general case, the stability of the intermediate equilibria during the transition is not guaranteed, and must be evaluated. Finally, the determination of the criteria used to instigate such an adaptive sequence leading to a graceful termination must be based on a trade-off between performance and safety margins: an early transition strategy based on proximity to the limit would avoid any non-linearity in response due to actuator saturation, but gives up some of the design range; delaying the response until the limit is actually reached risks disruption. In the case of ITER, which has very small margin in terms of coil currents in the nominal scenario, these trade-offs are especially critical.

While disruption mitigation is a necessary component of the ITER strategy, disruption avoidance is clearly a more desirable goal. Of necessity, the ITER scenarios require operation close to stability boundaries. Real-time estimation of the proximity of the operating point to instability, coupled with effective avoidance measures, would have significant benefits for robust operation. The ITER baseline H-mode scenario operates at modest β_N~1.8, so the disruptive limit of primary concern is probably the n=0 vertical instability. Two approaches to estimation of the stability boundary seem feasible: evaluation of the operating point equilibrium elongation, l_i, *etc*., for comparison to pre-computed closed loop stability margins; and direct observation of the plasma response to the control system drive. The latter method could be a relatively straightforward extension of the observer employed in the VDE mitigation system, while the former would require a more sophisticated real-time equilibrium calculation but could have the advantage of longer look-ahead capability, allowing more response time for modifying the discharge trajectory. An obvious adaptive response to detection of reduced stability margin would be to reduce the elongation, although such an approach must be applied in such a way as to reduce q_{95} below 3, so the current may need to be reduced as well.

Contributions to ITPA/ITER

Experimental work carried out under Integrated Scenarios – H-mode thrust includes support for ITPA/IEA Joint Experiments. Currently open experiments and those completed during the previous five year period are summarized in the following table. In some cases the C-Mod experiments are conducted jointly between the Integrated Scenarios and one or more of the Topical Science Groups, so there will be some overlap between this table and similar ones found in other chapters.

Summary of Integrated Scenario (H-mode) Work for ITER/ITPA

Description	JOINT Experiments	Notes on C-Mod Contributions
L-H threshold power at low density	TC-3 (CDB-11)	Initial C-Mod experiment completed
H-mode transition and confinement dependence on ionic species	TC-4 (CDB-12)	For ITER pre-nuclear phase
Scaling of spontaneous rotation with no external momentum input	TC-9 (TP-6.1)	(2008)
C-MOD/NSTX/MAST Small ELM regime comparison	PEP-16	(2008)
Role of Lyman absorption in the divertor	DSOL-5	See also Boundary physics
Non-resonant magnetic braking	MDC-12	(2007-08)
Vertical stability physics and performance limits in tokamaks with highly elongated plasmas	MDC-16	Report submitted, additional work pending
Simulation and validation of ITER startup to achieve advanced scenarios	SSO-5	Addressing both AT and H-mode baseline scenarios; see also Integrated Scenarios-AT

ITER demo at q_{95}=3, β_N=1.8, n_e->0.85 n_G	ISO-1.1	New
Study seeding effects on ITER demo discharges	ISO-1.2	New
Ramp-down from q_{95}=3	ISO-2.2	New

H-mode Scenarios research on C-Mod directly supports ITER short term needs[10]

- Transport and confinement during transient phases; Ohmic, L-mode and H-mode.
- Access to high confinement regimes during steady-state and ramp-up/ramp-down phases, including power thresholds, ELM regimes and isotope scaling.
- Characterisation of ... ELM control, compatibility with scenario requirements
- Particle transport and fueling in ITER reference scenario plasmas
- Activities in plasma control
- Evaluation of impurity seeding in high-Z (Mo, W) divertor
- Development of non-active phase scenarios

2.4 References

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3. Advanced Integrated Scenarios

This integration thrust, as with the H-mode scenarios, includes research which integrates aspects of multiple Topical Groups – in some cases all of them – to investigate cross-cutting issues and to demonstrate scenarios which could be prototypical of ITER or other future devices. Chief among these "advanced scenarios" are the "hybrid" scenario, characterized by weak core shear and q_{min} ~1, and confinement improvement over standard H-mode, and non-inductive scenarios with higher bootstrap fraction. Experimental and modeling efforts are largely motivated by, and focused on, the needs of ITER since as discussed in the Introduction to this proposal the parameters and tools of C-Mod are uniquely relevant to ITER in many important respects and can inform the plans for ITER scenarios.

3.1 Research Highlights from FY08

The 2008 campaign was a highly productive period for experiments on advanced scenarios, largely enabled by the impressive progress in the LHCD system and physics understanding described in section 7.2 on Lower Hybrid research. LHCD operation, at the 1 MW power level, was even more routine than in past campaigns, and its effects on the plasma have been assessed over an increasingly wide range of plasma parameters. The upgraded MSE diagnostics confirms substantial j(r) modification. This has meant that the integration program can begin to move from development of current drive control tools to *using* these tools to explore their effects on plasma performance. Similarly, the new cryopump, in combination with a range of density control techniques developed as part of the Pedestal Physics program (see Pedestal Physics, section 5), has been used to develop reduced H-mode plasmas accessible to LH waves.

Several experiments during the FY08 campaign explored techniques to modify j(r) evolution during the current ramp. These included strong ICRF heating, triggering L-H mode transitions, and LHCD. Results were summarized in an APS presentation by Chuck Kesssel [C. Kessel, A.E. Hubbard, P. T. Bonoli et al,, PP6:74, *Progress using LHCD and ICRF on Advanced Tokamak discharges on Alcator C-Mod*, 50th Annual Meeting of the Div. Plasma Physics, Dallas, TX, November 2008, Bull. APS **53** (14) 221 (2008)]. As has been found in other C-Mod experiments, including some ITER demonstration discharges in the H-mode Scenarios program, ICRH, even up to 3 MW, does not generally decrease q(0) or significantly delay sawteeth, and can often even induce earlier sawteeth. Early L-H transitions do reduce l_i, but the effect is modest and transitions are difficult to reproducibly trigger and maintain, given their strong dependence on the low density limit for H-modes which is itself poorly understood. Impurity and radiation levels are a key factor in both techniques. Much more effective at maintaining q0 > 1 and substantially delaying sawteeth is LHCD during the rampup. Even relatively modest power levels (0.5 MW) have succeeded in reducing li and in delaying sawtooth onset for hundreds of ms – greater than a current relaxation time. This effect increases with decreasing density, as expected due to higher LH driven current. In the best cases, sawteeth are suppressed until 700 ms, and return only at the end of the LH pulse. (See Figure 3.1)

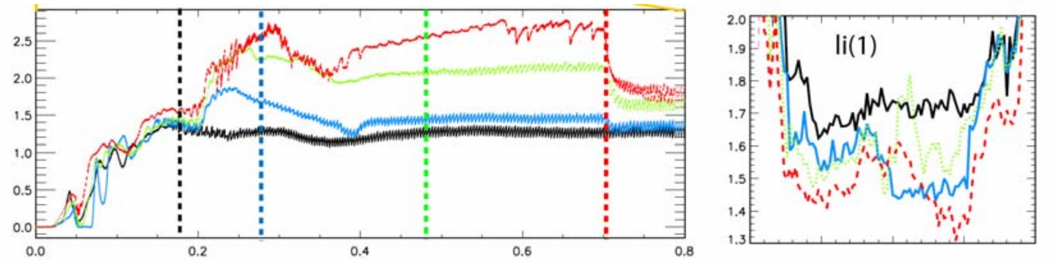

Figure 3.1 Demonstration of sawtooth suppression using Lower Hybrid current drive.

Using this technique to modify q(r), initial experiments were performed to assess the impact on H-mode evolution and confinement. This was proposed by and performed in collaboration with A.C.C. Sips (IPP Garching) as part of an ITPA joint experiment to explore the "Hybrid" scenario under ITER-relevant conditions. Sawteeth in the ohmic target plasmas were delayed until typically 600 ms as described above. ICRH was then applied at varying times, in each case promptly triggering an L-H transition, and also the onset of sawteeth. The first order result, which was quite interesting and encouraging, was that the stored energy of these H-modes did systematically vary depending on the timing, indicating that the target q(0) profile was indeed affecting confinement (Figure 3.2). The maximum stored energy was obtained with L-H transitions at 0.6 s, close to the sawtooth onset time, a result reminiscent of hybrid or "improved H-modes" on ASDEX Upgrade.

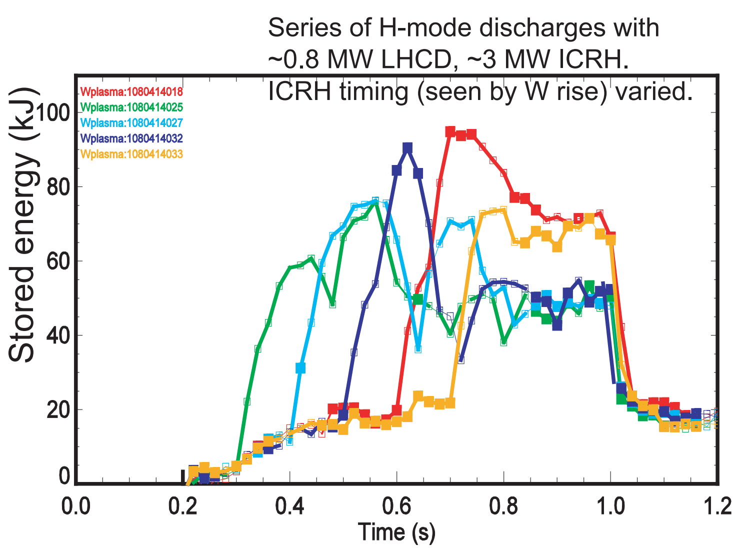

Figure 3.2 Combined LHCD and ICRF illustrates improved confinement as q(0) is increased.

Closer examination of the L-H transition dynamics and pedestal evolution revealed interesting differences. In the discharges with LHCD-modified j(r), the L-H transition was delayed, density rise was slower, and the peak pedestal temperature higher (Figure 3.3). T_{ped} correlated well with the peak stored energy. The higher performance was, however, not sustained. This, and the reappearance of sawteeth, may indirectly indicate that current drive reduces in the H-mode. As discussed in the LHCD section, exploration of LH efficiency at high density, in both L and H-modes, and comparison with improved LH models, is an issue of strong importance. It is one which C-Mod is

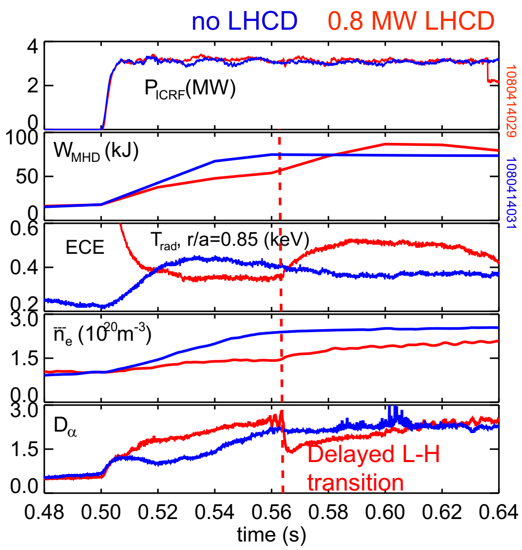

Figure 3.3 Comparison of H-mode transition dynamics with and without LHCD.

uniquely capable of addressing, and could have implications for scenarios on ITER as well as C-Mod.

3.2 Research Plans for FY09-11

The planned research program will build directly on the results outlined above, as well on advances in other topical areas, and in integrated modeling. It will in particular leverage the expected increases in LHCD power, due to the new and improved launcher to be installed in FY09, the second such launcher later, and the additional klystrons to fully power these launchers. As such, the pace of progress will be largely determined by the budgets for such upgrades.

A top priority in the LH physics area, which as mentioned will directly impact the development of advanced scenarios, is investigating the current drive efficiency at high density (> 10 20 m-3). Initial indications are that LHCD effects decrease at a lower density than had been expected. However, this is based on sparse experimental data and on indirect measurements such as x-ray emissivity. Many quantitative questions remain unanswered, such as: *How does n_{CD} depend on density? Is the dependence different in L and H mode? How does it depend on other plasma parameters (I_p, B_T) and on LH parameters ($N_{//}$).* A systematic study early in the FY09 campaign, in concern with improvements in LHCD models (more realistic SOL profiles in CQL3D/GENRAY, COMSOL, full-wave LH calculations) promises to shed light on the mechanisms for this unexpected dependence. Once this is understood, appropriate means of modifying it can be explored, and the possible impact on ITER and C-Mod scenarios assessed.

Experiments to assess the "hybrid" scenario will receive high priority in the advanced scenarios area in FY09, given a Milestone on this topic. The Research Goal here is :

"With the implementation of Lower Hybrid RF for current profile control, and active cryopumping for density control, C-Mod will investigate advanced scenarios for improved performance of the tokamak. Investigations into the so-called "hybrid" mode of operation, being considered as one possible advanced approach for ITER, will be carried out to evaluate the potential to maintain central safety factor near or slightly above 1 and to assess the effects on plasma transport and confinement."

This C-Mod goal dovetails very well with stated priorities of the ITER project and ITPA, in particular a new Integrated Operating Scenarios Joint Experiment IOS 4.1 *"Access conditions for hybrid with ITER-relevant restrictions"* which aims to document the conditions for improved performance in physics variables (eg, j(r)), as opposed to operational recipes. The background to this is that various tokamaks have used different means of producing improved confinement, for example by applying early NBI, leading to tearing modes in the core, and have adopted different definitions of the regime. One point of controversy is that the AUG team regards reduced shear in the core, as being crucial and has shown significant differences with quite small variations in j(r). The DIII-D team tends to regard such differences as being more an effect than a cause, and emphasizes improved stability. By varying j(r) through external means – LHCD - the effect should be much more clearly observable.

4. Core Transport

Progress in understanding transport is essential for the extrapolation to next-generation experiments and for exploitation of fusion for practical energy production. The long-term goals of transport research on C-Mod are to:

1) Provide support for ITER operations through targeted experiments.
2) Contribute toward the development of first-principles understanding of transport in confined toroidal plasmas.
3) Validate the evolving set of nonlinear turbulence codes.
4) Discover and exploit new and unexpected results.

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 C-Mod program seeks to leverage the unique characteristics of the experiment as part of the coordinated national and international efforts in which it is embedded. And while the plan summarized here does describe a large number of critical and unique contributions to that effort, these clearly gain greater value from the context of the world program.

Recent Research Highlights

Development of predictive modeling is a major programmatic goal for the U.S. fusion program. As noted above, a key element in our transport research is the validation of nonlinear gyrokinetic codes via comparisons between theory/simulation and our experiments. The C-Mod team has active collaborations with the gs2 and GYRO simulation teams. Quantitative comparisons are possible for more mature topics, for example ion thermal transport, while research into electron heat transport, particle and momentum transport is still more qualitative in nature. C-Mod researchers and collaborators are actively involved in the development of synthetic diagnostics which calculate the predicted signals from turbulence models. Analysis is carried out by C-Mod staff and students, members of the PSFC theory division and with collaborators.

In recent studies, the dynamics of internal transport barrier (ITB) formation by off-axis ICRF were investigated through observation of fluctuations seen by phase contrast imaging (PCI) [a]. This diagnostic is capable of measuring density fluctuations in a wide range of frequencies (2 kHz-5 MHz) and wavenumber (0.5-55 cm-1). Recent upgrades have enabled the PCI diagnostic to localize short wavelength turbulence and resolve the direction of propagation (i.e., electron vs. ion diamagnetic direction) of longer wavelength turbulence. Nonlinear simulations using the GYRO code [b] were performed and compared with absolutely calibrated experimental measurements through a synthetic PCI diagnostic [c]. The simulated fluctuation spectrum from GYRO agree with experimental measurements in the ITG dominated H-mode transport regime. The magnitude of fluctuations and transport shows good agreement with experimental measurements if the ion temperature gradient is reduced by about 15% and/or by adding E×B shear suppression within the experimental uncertainty. For weak ITBs with only modest density peaking, we find that the TEMs do not play any significant role in transport as compared to earlier results which had much steeper density gradients and more robust ITBs [d].

In contrast, studies of electron transport in low density Ohmic plasmas revealed a discrepancy between experimental results and turbulence models [e]. In this regime, without auxiliary heating, all power is coupled into the electrons and the electron-ion energy exchange is small. Confinement deteriorates rapidly at low density in the "neo-Alcator" regime. Electron transport is poorly understood, but will be important in future devices where $\tau_E > \tau_{ei}$ and neither transport channel is ignorable. In recent experiments, densities were scanned from the linear (Alcator) regime to the saturated (L-mode) regime. The key role played by the ion temperature gradient (ITG) turbulence in the saturated regime was verified by measurements of turbulent wave propagation, which was dominantly in the ion diamagnetic direction. It was found that the intensity of fluctuations increased with density, in agreement with simulations (fig 4.1b). The measured fluctuation intensity agreed with simulation within experimental error (+/-60%). In the saturated Ohmic regime, the simulated fluctuation k spectra (fig 4.1a) and ion and electron thermal diffusivities also agreed with experiments within experimental uncertainty. However, in the linear Ohmic regime, GYRO predicted significantly larger ion thermal transport and smaller electron thermal transport than the experimentally measured values. Nonlinear simulations show that a significant thermal transport contribution from the trapped electron mode or electron temperature gradient (ETG) turbulence is not likely in the low density Ohmic regime in C-Mod.

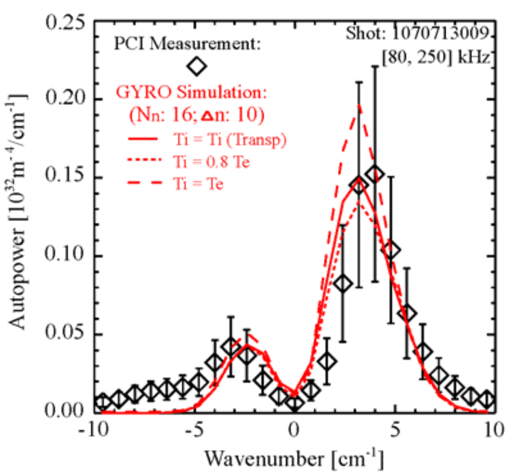

Fig 4.1a. The experimental measurements of the fluctuation k spectra taken with PCI agree well with nonlinear gyrokinetic simulations using the GYRO code.

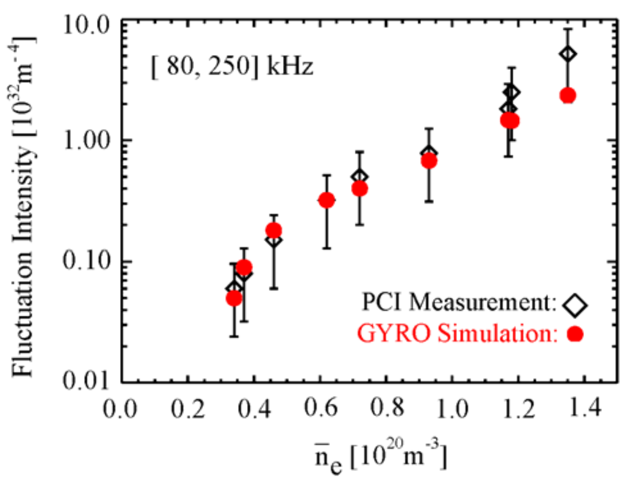

Fig 4.1b. The measured fluctuation intensity, in the band associated with ITG turbulence, agrees with gyrokinetic simulations.

GYRO simulations have provided the first understanding of density peaking in low-collisionality H-mode plasmas in Alcator C-Mod, and motivated revised analysis of similar plasmas in JET and AUG. Previously reported findings [f] showed moderate peaking as collisionality was lowered to the values expected for ITER. The C-Mod results confirmed earlier experiments on JET and AUG [g,h] but broke the covariance between collisionality and n/n_G (density normalized to the density limit) which was found in that analysis. The combined data of the three machines allowed for an unambiguous prediction that baseline H-mode operation in ITER would have a modestly peaked density profile, with $n(0)/\langle n_e \rangle \sim 1.5$. More recent experiments on C-Mod showed a significant dependence on safety factor as well (see fig 4.7). Simulations were based on measured profiles of ion temperature, and the electron temperature in a series of plasmas with varying density peakedness. By varying plasma parameters used in the simulations, it was shown that the particle pinch responsible for the density peaking is caused by kinetic electron effects that 'amplify' turbulence that is driven primarily by the ion temperature gradient. These effects are diminished by high electron collisionality, and this reduces the density peaking in higher density C-Mod plasmas. In higher density plasmas the ion temperature gradient is larger and this further diminishes the role of the kinetic electron instability drive. The pinch is generated by the shorter wavelength modes in the simulations, with $k_\theta\rho_i > 0.4$, and this explains why previous quasi-linear analyses at AUG failed to find an experimentally relevant particle pinch. Motivated by the C-Mod simulations, recent re-analysis for both AUG and JET conditions based on higher $k_\theta\rho_i$ modes now finds good qualitative agreement with the experimental data.

Recent work on self-generated rotation has revealed a complex dependence on plasma parameters, particularly density, safety factor and magnetic topology [i]. We had previously reported a strong dependence on magnetic topology which was correlated with scrape-off layer (SOL) flows [j]. A strong decrement in rotation is observed when the plasma is scanned from lower single null (ion B drift toward the x-point) to upper single null. This dependence is found to be strongest at densities near 1.5×10^{20} and diminishes at lower or higher densities. Rotation direction can be inverted by density ramping, with a sharp reversal observed near 1×10^{20}. The effect is seen in both diverted and limited plasmas and shows a dependence on safety factor as well. Similar results were reported by the TCV group [k]. Comparisons between C-Mod and TCV data are ongoing. Momentum transport was investigated by measuring the rotation profile evolution following fast sweeps of the magnetic topology (fig. 4.2). These experiments found a momentum confinement time, in L-mode, of about 0.04 seconds, comparable to the energy confinement time. The changes in rotation profile

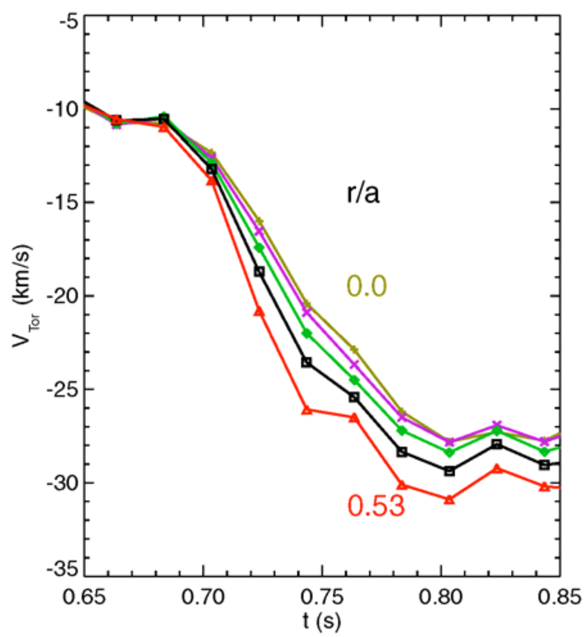

Fig 2. Toroidal velocity rotation evolution following a shift in magnetic topology which causes the SOL flows to decrement in the counter current direction.

were seen to propagate inward from the edge with an effective diffusivity of about 0.2 m^2/s and no evidence of a pinch, for these discharges. Scaling studies for intrinsic rotation were extended to include the regime of improved L-mode confinement [m]. This regime has a strong temperature pedestal, but no density pedestal. Energy confinement is H-mode-like, but particle transport is L-mode-like. Intrinsic rotation in these discharges matched the scaling found previously for H-mode plasmas.

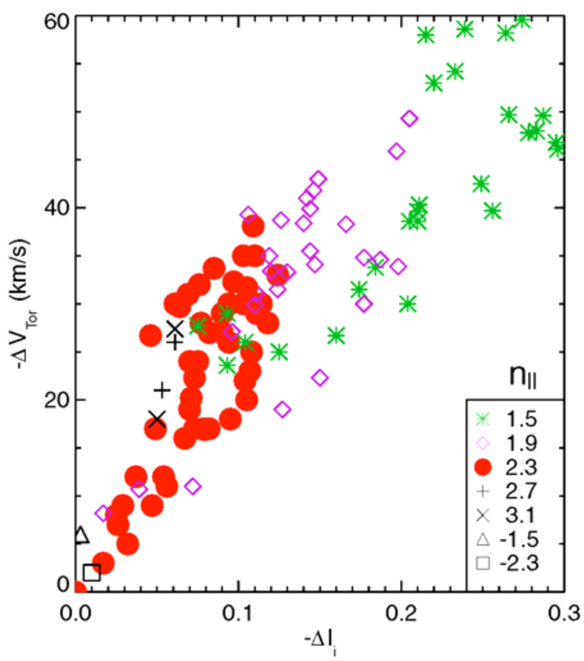

Fig 3. The change in toroidal rotation observed during LHCD is tightly correlated to the change in plasma current profile as parameterized by the internal inductance.

Important new observations concern plasma flows driven by both ICRF and LHCD systems on C-Mod. The ICRF results compared the standard D(H) minority heating regime to a mode conversion regime using a mixture of D and He_3. Very strong core rotation was observed, up to 100 km/s, which was proportional to RF power/particle and exceeded the magnitude of intrinsic rotation by more than a factor of two [n]. Unlike intrinsic rotation, the ICRF driven flow is core localized and appears immediately in the plasma core rather than propagating inward from the edge. Poloidal rotation was also observed at velocities approaching 2 km/s in a narrow region which corresponded to the mode conversion location. The radial electric field shear was dominated by the toroidal velocity gradients with **E**x**B** shearing rates as high as 2×10^5/s close to the computed ITG linear growth rates [o]. In contrast, LHCD was seen to drive counter-current rotation. The magnitude observed was up to -60 km/s and was found to be proportional to the amount of driven current or equivalently, the measured change in internal inductance (see fig 4.3). The time history of the driven rotation followed the change in current profile, as measured by the internal inductance with a time constant much longer than the energy, particle or momentum confinement times. This rotation was also localized in the plasma core and resulted in a significant E_r well and **E**x**B** shearing [o2].

Investigations of internal transport barrier (ITB) physics included those formed by off-axis ICRF and by Ohmically heated H-modes. Previous comparisons with simulations showed that these barriers were formed by reduction of the ITG drive term, R/L_T and the action of the neoclassical pinch. High frequency fluctuations measured before the onset of the barriers are consistent with gyrokinetic simulations of ITG turbulence. The barrier is most obvious in the particle channel and leads to highly peaked density profiles with a strong inflection point at the barrier foot. The density peaking helps sustain the barrier by decreasing $\eta = L_n/L_T$ and thus further reducing the ITG drive. The processes and phenomenology for ICRF and Ohmic ITBs are quite similar, suggesting no important role for RF specific mechanisms. Transport analysis shows sharply reduced energy transport inside the barrier region (with ions approaching neoclassical levels), but with only small changes in the electron temperature profiles. New measurements with the high-resolution soft x-ray spectrometer, show an inflection in the ion temperature profile with stronger gradient just inside the barrier foot (see fig 4.4). The working hypothesis is that the barrier is in the ion channel and that the unimproved electrons, which are strongly coupled to the ions in C-Mod, prevent stronger peaking of the temperature profiles. Measurements of boron profiles, using charge exchange recombination, show peaking, but at levels well below neoclassical. Even though these plasmas are relatively collisional, the pressure gradient is sufficient to drive up to 12% of the plasma current via the bootstrap mechanism.

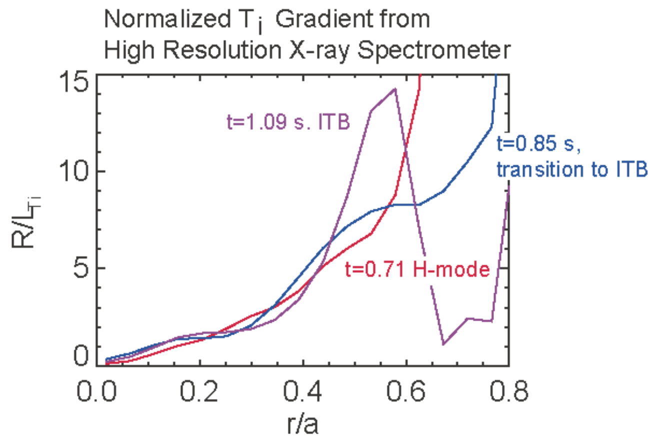

Fig 4. Following a transition to an ITB regime, the normalized ion temperature gradient is found to increase sharply just inside the barrier foot.

Core Transport Research Plans

We will continue careful testing of theoretical and computational transport models. New and improved diagnostics will be available, broadening the scope of comparisons between experiments and simulations. Efforts will be undertaken for continued development of synthetic diagnostics for the turbulence codes and appropriate metrics for quantitative evaluation of the comparisons. Work on ion energy transport will continue through improved documentation of T_i, T_e and J profiles and their gradient scale lengths along with comparison with the GYRO and gs2 codes. We expect that in most C-Mod plasmas the experimentally achievable temperature gradient scale lengths will be larger than or equal to the effective critical gradient scale lengths set by ITG/TEM and/or ETG modes. It should also be possible to measure density fluctuation amplitudes and radial correlation lengths to enable a broader comparison with theory. Studies in a variety of plasma regimes will experimentally determine important parametric dependences and enable a broad test of theoretical expectations. Other proposed diagnostics may contribute to these studies. A measurement of zonal flows may be possible with faster version of the high-resolution x-ray spectrometer used to measure ion temperature and velocity profiles. These fluctuating flows, which are believed to provide the primary saturation mechanism for drift-wave turbulence, may also be studied with a proposed Doppler reflectometry system.

Electron transport remains poorly understood. As noted above, we still have no answer for what is causing electron transport in the low-density OH regime. The most advanced gyrokinetic codes do not predict the increase in electron energy confinement that is universally observed at low densities (fig 4.5). Research will proceed along three parallel tracks. First, the measurements of background profiles, including the current profile, will be improved in order to provide the best data for the simulations. At the same time, the measurements of density fluctuations will be expanded to the highest range of k values.

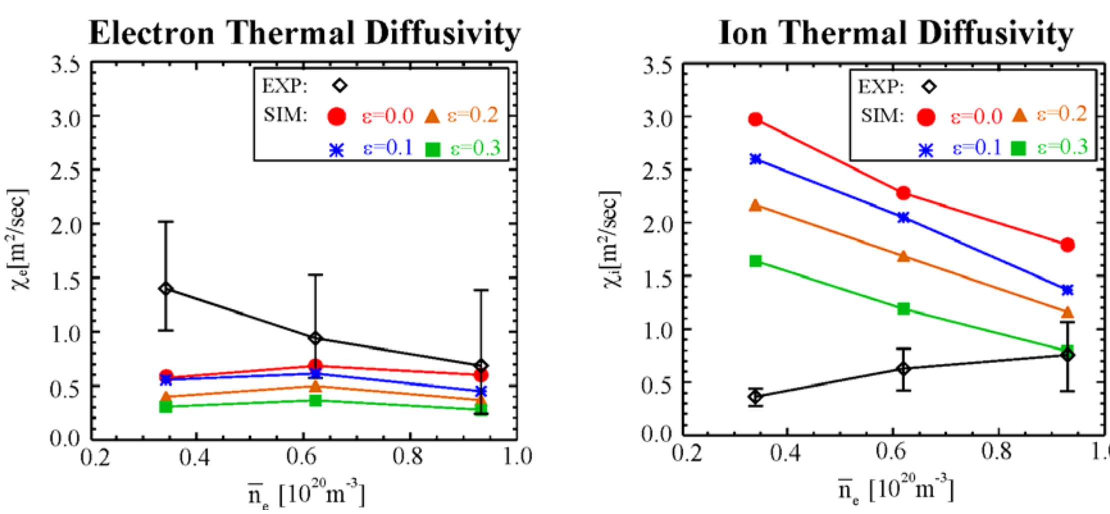

Fig 5. Predictions of thermal diffusivity from the GYRO code are compared with experimental measurements of transport in low-density Ohmic plasmas. While the experiments show that the decrease in confinement is due to losses in the electron channel, the codes find that the trend is due to deterioration in the ion channel. In the plots, the normalized ion temperature gradient is scanned - reduced by a factor 1-ε to match the absolute level of overall transport.

Secondly, we will attempt to measure magnetic fluctuations using the far-infrared polarimeter, currently being installed. Finally, in collaboration with the code developers, the discrepancy with simulations will be studied. Mechanisms, such as micro-tearing, which are currently not simulated by the gyrokinetic codes, will be investigated. Discussions with other experimental groups has begun with an aim of repeating these studies on DIII-D and/or NSTX.

The complexity of intrinsic rotation in L-mode provides a unique laboratory for the study of momentum transport. Very little is known about the mechanism for this phenomenon, though several theories have been proposed. It should be possible to test aspects of many of these theories. It has been proposed that the rotation is due to Reynold's stress [p] or symmetry breaking by sheared **E**x**B** flows [q]. It will be desirable, but difficult, to address the question of self-generated flows at the level of fluctuations and turbulence. Other theories rely on the Coriolis drift effect [r], turbulent equipartition [s] or the thermoelectric pinch. These would predict specific relations between profiles or transport in different channels – that is a fixed relation between momentum, particle or thermal transport. Gyrokinetic codes are beginning to incorporate enough physics to address this problem through simulation. It should be possible to compares measured flow profiles and cross-field momentum fluxes with these emerging models. Transient transport experiments via modification of the magnetic topology have already yielded important results. These will be extended, including flow modulation via MC-ICRF and LHCD providing, perhaps, the most stringent tests of theories and codes. Improved measurements of the edge flows and electric field may provide information on the boundary conditions or sources for intrinsic core rotation.

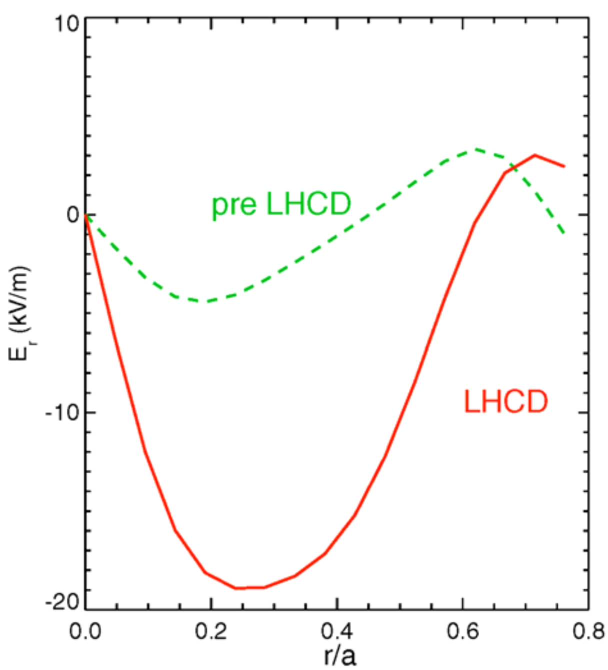

Fig 6. The profile of the radial electric field and the ExB shearing rate can be strongly modified by LHCD.

Detailed plans for investigating RF flow drive are covered in the RF section of this proposal. However, these driven flows are already providing flow shear at levels that are interesting for transport studies. Experiments are planned to exploit these tools by looking for the associated changes in transport (Fig 4.6.). Because rotation from LHCD is in the counter-current direction, while MC-ICRF and intrinsic flows, H-mode, are co-current, it may be possible to tailor the E_r profile in a way which is particularly favorable for modifying transport and controlling the pressure profile. Future studies will focus on optimizing the driven flow and investigate the effects of localizing the RF-plasma interaction in controlling the rotation profile produced. An important goal is to understand the driven rotation sufficiently to extrapolate into the reactor regime where external torque from neutral beams would be small or non-existent and to determine whether the resulting rotation will be sufficient to affect micro- or macro-instabilities. Multi-machine scalings already suggest that resistive wall modes may be stabilized by intrinsic rotation alone, without resort to more complicated feedback schemes.

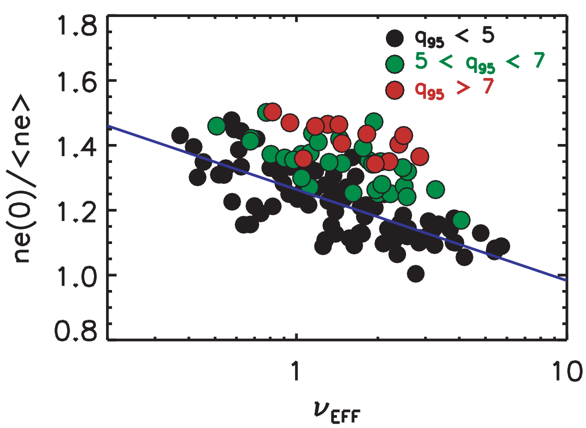

Fig 7. Density peaking experiments demonstrate the existence of a significant inward pinch at low collisionality. The pinch is also affect by the safety factor presumably through the magnetic shear.

Understanding particle transport is crucial for predicting density profiles, fueling requirements and impurity radiation. The experimental work on density peaking at low collisionality will continue. The dominant effects seem to be collisionality and the safety factor (fig 4.7.) Improved data sets will be collected with emphasis on getting the best quality ion temperature, rotation and current profile information – the dependence on safety factor suggesting an important role for magnetic shear. Data on fluctuations will also be collected. Comparison with simulation will focus on the interplay between various forms of drift-wave turbulence – ITG and TEM turbulence are sensitive to L_T, L_n and L_S. Recent simulation work has suggested an important role for turbulence around $k\rho_s \sim 0.4$, somewhat above the range where the ITG spectrum usually peaks, however analysis suggests that the crucial mechanism is the suppression of the kinetic electron response at high collisionality. The pinch is driven by the electron response and the implication is that it exists when the ITG drive is close to marginal stability. We would like to understand the relation between ion-energy, particle and momentum transport at the level of turbulence and fluctuations and to predict the plasma conditions which lead to a significant inward pinch and density peaking. Comparative studies with DIII-D are under discussion. LHCD may allow experiments with $E\phi = 0$ and thus no neoclassical pinch. Experiments should cover both H-mode and improved L-mode and should include helium plasmas in support of stated ITER needs.

A powerful tool for the study of particle transport will be available for upcoming campaigns. A new version of the laser blow-off impurity injector is under construction and will be installed on C-Mod. This device is capable of injecting trace amounts of non-perturbing, non-recycling, non-intrinsic impurities. These impurities act as passive scalars, convected by turbulence and by collisions with the main ions. The source is essentially a delta function in time and radius - localized at the plasma edge. Since these elements do not recycle and are not present in the plasma if not injected, they are ideal for studying the transient behavior of particle transport. The approach will be to calculate impurity profiles, using spectroscopy and broad-band x-ray arrays combined with modeling of the relevant atomic physics. From these data, impurity fluxes can be calculated and dependences on relevant plasma parameters tabulated (n, T, q, L_n, L_T, L_S). The injected impurities may also serve as a tracer in a search for non-diffusive transport (flights or sub-diffusive transport) which are predicted by some models. The injector will also answer several practical questions, mapping out conditions where overly good particle confinement leads to impurity accumulation and unacceptable levels of radiation or fuel dilution.

Both gs2 and GYRO, experimentally-oriented 'first principles' turbulence simulation codes are capable of calculating impurity transport Planned validation efforts include searching for significant pinches in the code results, comparing the experimental and predicted particle diffusivities for a variety of impurity elements and varying q profiles. Since the impurity injection is non-perturbing it is possible to do the simulations before the experiments, and thereby to optimize the experiments.

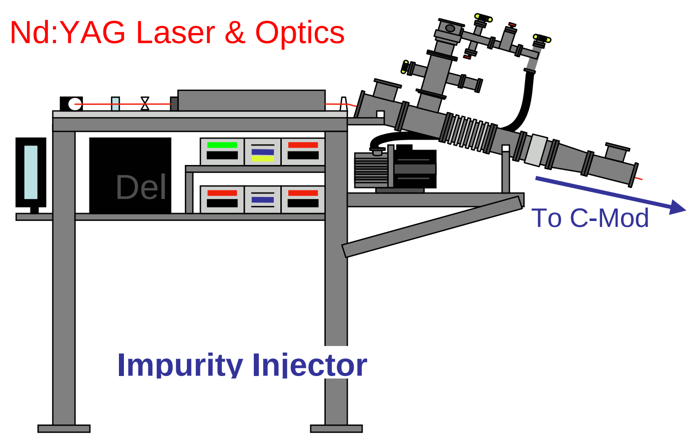

Fig 8. A schematic of the laser blow-off impurity injector.

Preliminary simulations with multiple impurities show that interactions between species is not common (the presence or absence of different impurities does not affect the results much), and that the commonly employed D,V ansatz for particle flux does represent the simulated impurity transport. Studies of the role of magnetic shear will be enabled by the LHCD system, which can directly manipulate the q profile, and by improvements in the measurements of the current profile by the upgraded MSE system.

Studies of internal transport barrier physics will be enabled by improved tools – lower hybrid current drive and mode-converted ICRF – and by improved diagnostics. Particularly relevant are upgrades to the MSE system and the new polarimeter for measuring the plasma current profile and the installation of a scanning reflectometer which should allow measurements of density fluctuations at the barrier foot location, where transport is most strongly reduced. These experiments will also benefit from the comprehensive measurements of ion temperature and velocity profiles that are now available. In general, for barriers, we're interested in: 1) What are the dominant instabilities which are present before the barrier forms? 2) What is the stabilization mechanism? 3)What mechanisms governs evolution and saturation? 4) What is the transport within the barrier? 5) Do we have mechanisms for controlling the strength or position of the barriers? We will study access conditions for barrier formation in terms of local variables, particularly those predicted to be important by theory. Detailed comparisons with the gyrokinetic codes will continue. An important programmatic goal will be the creation of ITBs by directly manipulating the magnetic shear with the LHCD. It is predicted (and found on other devices) that barriers can form when the shear is driven close to or below zero, even in the absence of strong **E**x**B** shear. On C-Mod, the

ICRF flow drive may provide an additional tool by allowing control of the flow shear. Barrier particle transport may be probed using impurities from the laser blow-off system. It will be critical to find barrier conditions where the energy confinement is sufficient to peak up the pressure profile (and thus the bootstrap current), but where the impurities are not so well confined that they accumulate.

Support for ITER and Connection to ITPA Activities

Research on C-Mod directly supports ITER short term needs as defined by the ITER STAC in their 5th meeting and reported as: *"ITER Physics Work Program: Short Term Activities (2009-2011)"*

- Transport and confinement during transient phases; Ohmic, L-mode and H-mode.
- Develop of models and scalings for confinement transients during stationary phases of discharges including L-H and H-L transitions, growth and collapse of ITBs and collapse of other peaked density regimes.
- Access to high confinement regimes during steady-state and ramp-up/ramp-down phases, including power thresholds, ELM regimes and isotope scaling.
- Particle transport and fueling, including density peaking control.
- Momentum transport and prediction of plasma rotation in ITER.
- Evaluation of requirements and confinement performance in hybrid scenario.

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5. Pedestal Physics

Research on the H-mode edge is driven by the desire to understand the detailed physics processes which give rise to edge transport barriers (ETBs), and how these various mechanisms combine to determine the boundary values of density and temperature for the core plasma (*i.e.* the edge *pedestal*). ITER and other future devices present new challenges in this area, as they will require substantial pedestal pressure in order to obtain high energy confinement, yet the edge-localized modes (ELMs) that often characterize high performance discharges must be suppressed or constrained to small amplitude, in order to prolong the lifetime of divertor targets. In addition there are open questions surrounding the mechanisms governing the transition from L-mode to H-mode plasmas, and the capability of existing power threshold scalings to predict the required input power needed for ITER to obtain H-mode.

The C-Mod program is actively engaged in resolving these crucial questions with an aggressive experimental pedestal program. We achieve progress toward these goals by
Commiting resources to improving diagnostics in the scrape-off layer and pedestal regions
Designing dedicated experiments to study pedestal transport, edge relaxation mechanisms and L-H thresholds
Engaging theoretical colleagues in collaborations intended to enhance the interpretation of experimental results
These studies are done in regimes that have particular ITER relevance, typically having coupled ions and electrons and high opacity to neutrals. In this way, the program is complementary to those of other lower-density machines.

5.1 Recent pedestal physics highlights

H-mode studies on C-Mod have enjoyed access to a wide range in toroidal field and plasma current ($2.7<B_T[T]<8.0$, $0.4<I_P[MA]<1.7$), and with varying density and ICRF input power [[11]], allowing empirical scalings in pedestal parameters to be examined with a significant variation in dimensional and non-dimensional parameters. As discussed later, both variation in plasma shaping and active pumping have also been used to extend our H-mode data set in interesting ways. Throughout, we have taken advantage of edge diagnostics with millimeter-scale radial resolution in order to characterize the structure of H-mode pedestals, as well as their scalings with discharge parameters. Figure 5.1 illustrates examples of high-resolution profiles of electron temperature from edge Thomson scattering (ETS) and B+5 temperature and poloidal velocity from a recently implemented edge charge exchange recombination spectroscopy (CXRS) diagnostic. Over a wide range of collisionality, strong equilibration of electron and impurity ion temperature is observed. Well-resolved measurements from CXRS now allow direct inference of the radial electric field E_r via the radial force balance equation, and subsequent characterization of its structure [[12]]. The ratio of the E_r well width to machine size matches published results from DIII-D, ASDEX Upgrade and JET, and a clear correlation between E_r well depth and quality of confinement is demonstrated, as in Figure 5.2.

Because of its substantial leverage on performance in ITER scenarios, there is a substantial effort on all major tokamaks to identify the important physical mechanisms setting pedestal width α. For typical C-Mod equilibria, pedestal width is approximately 3-4% in normalized flux space, and exhibits little systematic variation with engineering parameters. Using both data base analysis and dedicated experiments [13], it has been found that C-Mod pedestal width does not scale explicitly with either the toroidal or the poloidal ion gyroradius. Also, enhanced D_α (EDA) H-modes without edge-localized modes (ELMs), the commonest H-mode regime on C-Mod, show no obvious scaling of pedestal width with poloidal beta. However, recent ELM-synchronized analysis of Type I ELMy discharges on C-Mod did reproduce a scaling of $\Delta \propto \beta^{1/2}$, very similar to that observed on DIII-D (see Figure 5.3), and suggesting a promising new line of collaborative pedestal width research.

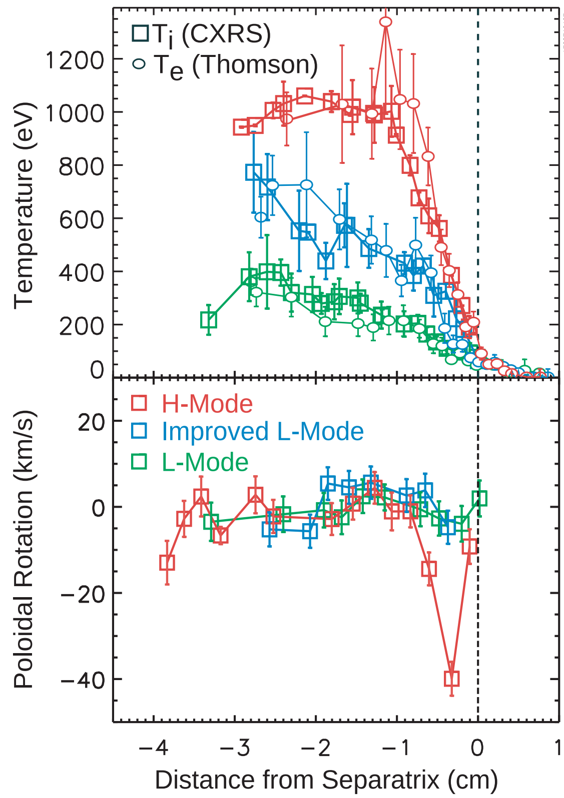

Figure 5.1: Examples of edge profiles from edge Thomson scattering (circles) and CXRS (squares) for three distinct confinement regimes.

The C-Mod pedestal is much less susceptible to details of edge neutral sources than larger, lower-field tokamaks, which for similar Greenwald fraction and safety factor, must run at considerably lower absolute density ($n \sim B_T/R_0$) [14,15]. This is made starkly apparent when attempting to match pedestal dimensionless parameters (*e.g.* β, ν*, ρ*) between C-Mod and JET, devices with a four-fold difference in linear size. In such a dimensionless match, the relative opacity to neutrals is inversely proportional to machine size, and an impact of neutral penetration on the density pedestal is in fact suggested by the JET data, while the C-Mod pedestals assume their usual widths [16].

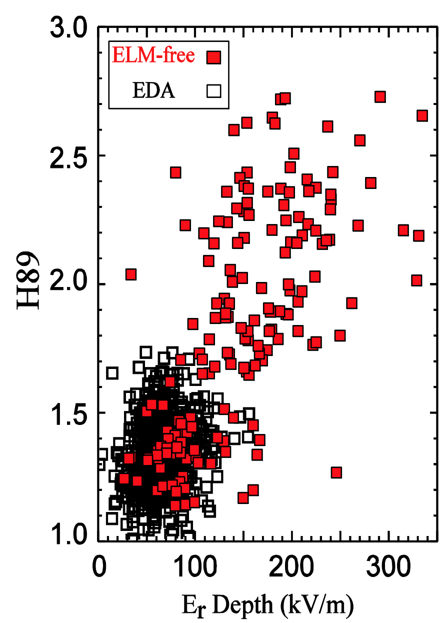

Figure 5.2: H_{89} confinement factor correlates with depth of the pedestal radial electric field well, as inferred from midplane CXRS measurements.

Throughout the bulk of C-Mod H-mode operation, the edge exists in some regime without large Type I ELMs to serve as a pedestal relaxation mechanism. In EDA and pure ELM-free regimes, we find that efforts to modify the edge pedestal by adjusting sinks and sources of particles, as well as efforts to adjust heat flux, reveal relatively (though not perfectly) stiff pressure and density profiles. The emerging picture from these studies is that the H-mode pedestal structure is regulated by transport such that a "critical gradient" is maintained. This transport paradigm contrasts strongly with simple diffusive transport, and is more akin to models for core transport which assume operation near marginal stability. Transport-governed stiff pedestals can complicate H-mode particle control, since the pedestal density is so tightly linked to plasma current. Recently, the application of lower hybrid (LH) power into EDA H-mode was shown to substantially reduce the "natural" pedestal density at a roughly constant level of energy confinement, raising the temperature pedestal, and thus lowering the collisionality [17]. An example of the modification to the electron temperature and density profiles is shown in Figure 5.4. Increases in the edge particle transport are observed to be quite prompt and occur without large amounts of LH power, suggesting the pedestal modification does not require core current drive. The particle pump-out, while not well-understood, suggests that LH could provide a useful tool for H-mode density control.

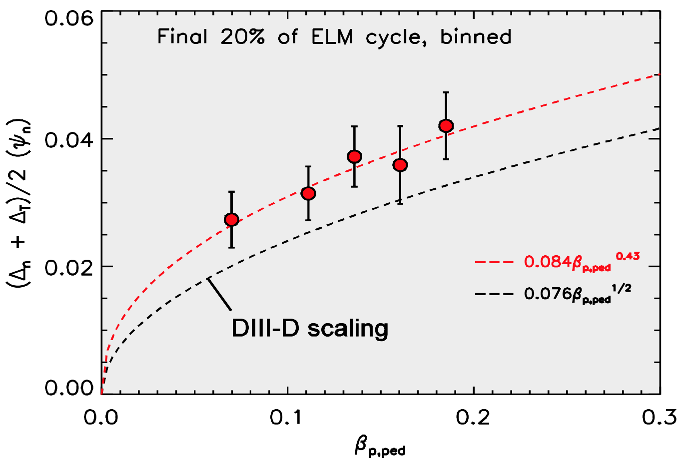

Figure 5.3: Characteristic pedestal width as a function of poloidal beta, in the last 20% of ELM cycles

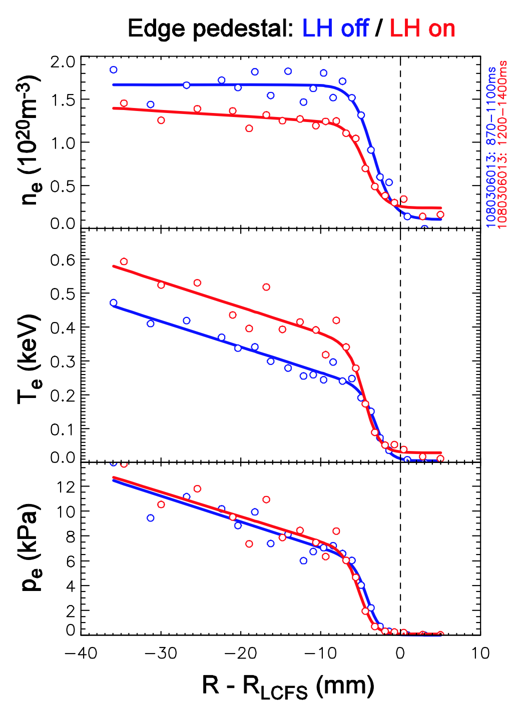

Figure 5.4: Pedestal profiles before (blue) and during (red) application of lower hybrid power to an ICRF-heated H-mode.

Research has also revealed significant influences of plasma shaping on pedestal parameters, confinement and edge relaxation mechanisms. In near double null (DN) configurations, when the distance between the primary and secondary separatrices (SSEP) becomes comparable to the pedestal width, H-mode character changes. Most notably, for discharges biased slightly such that the primary x-point is in the favorable ion ∇B drift direction, confinement improves significantly, and a regime of small ELMs is obtained more readily than in a more strongly single null discharge. These small ELMs were studied as part of an ITPA joint experiment (PEP-16) among C-Mod, NSTX and MAST with an immediate goal of mapping the operational space for these kinds of regimes [18]. Figure 5.5 shows the edge operational space for these ELMs in terms of dimensionless pressure and collisionality. Two threshold conditions that correlated equally well with obtaining the small ELMs were β_N>1.3 and T_{PED}>600eV. These are similar to the thresholds observed in earlier studies at a more standard shape, but the conditions were obtained with less ICRF input power.

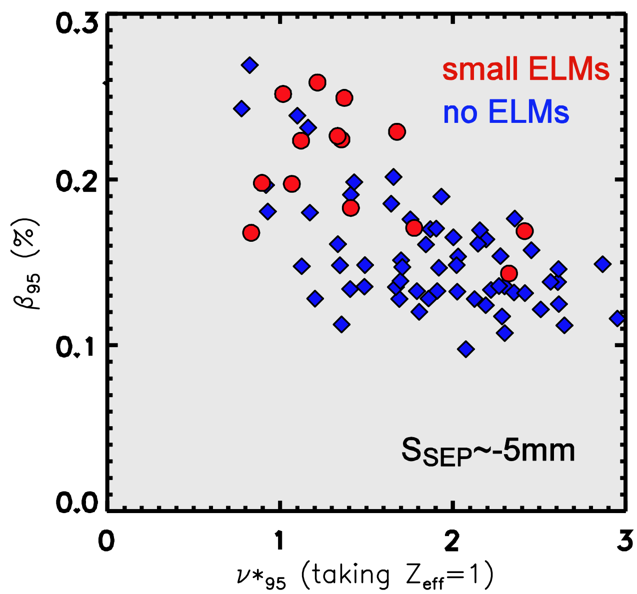

Figure 5.5: Operating space for the PEP-16 small ELM experiment, in terms of normalized edge pressure and collisionality

Significant experimental time has been devoted to questions of H-mode access, a particularly important question for ITER. On existing machines at moderate to high density, the H-mode power threshold increases with density, toroidal field and plasma surface area, and superficially seems easy to extrapolate to ITER. However at lower densities, many tokamaks have observed a sharp increase in the input power required to achieve H-mode. How this low-density limit n_{min} relates to other discharge parameters is critical for extrapolation to ITER, and the determination of whether ITER will have sufficient power for H-mode. C-Mod experiments tested the scaling of n_{min} with plasma current and found no dependence (*i.e.* the low-density limit is not a fixed Greenwald fraction). The scaling of n_{min} with toroidal field, illustrated in Figure 5.6, is more difficult to interpret, since, interestingly, the power thresholds seem to depend on the heating scheme employed (ICRF *vs*. Ohmic). Additional data from other devices participating in this high-priority ITPA joint experiment should help clarify these results.

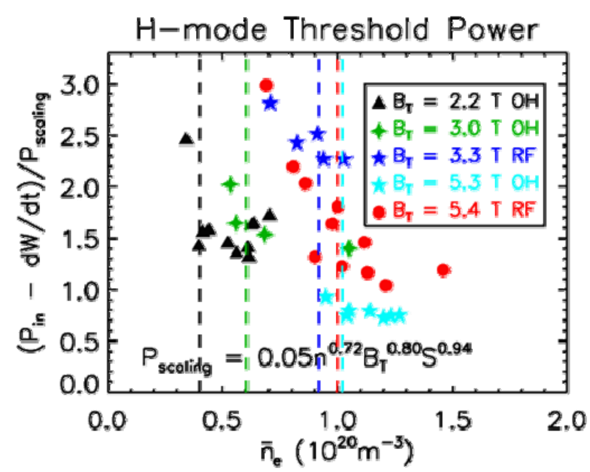

Figure 5.6: Ratio of input power at L-H transition to the ITPA power threshold scaling, as a function of density. Discharges of varying toroidal field and heating scheme are included

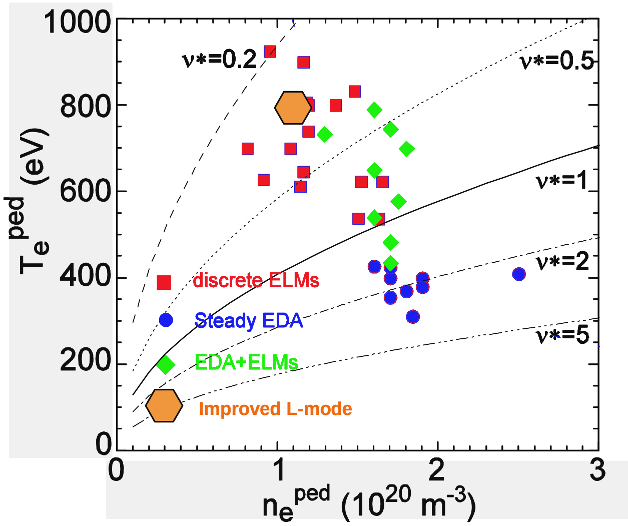

Figure 5.7: The pedestal conditions of improved L-mode yield local collisionality and beta similar to that in Type I ELMy H-modes, but are stable to ELMs.

Finally, experiments have been performed with the aim of suppressing the traditional L-H transition altogether and obtaining a form of improved L-mode, as first reported by ASDEX Upgrade [19]. These discharges have substantial temperature pedestals (nearly 1keV in the best cases) and increased edge particle transport relative to H-mode. The improved L-modes are made by running discharges with unfavorable ion ∇B drift direction, and maintaining input power slightly below the L-H power threshold, which is about twice that in a favorable drift direction case. These regimes have been maintained steady-state for several confinement times, and exhibit continuous particle and heat exhaust. Despite having pedestal collisionality and pressure comparable to Type I ELMy discharges (see Figure 5.7), no ELMs are observed. Rather, the particle transport seems to be regulated by a broad edge mode, reminiscent of the quasi-coherent mode (QCM) that regulates EDA H-mode. The low density (and collisionality) of this regime make it an attractive target for LHCD studies and advanced scenario development, and the narrow pedestal scale lengths, combined with high heat flux, are useful for studying power flux scalings in the SOL. The regime also merits study as an alternate operational regime for ITER, since it can achieve H_{98}~1 with no ELMs and does not require net torque from NBI.

5.2 Pedestal program plans

The fundamental goal of pedestal research is to project pedestals to larger tokamaks - ITER in the short term and power reactors in the far term. The general strategies for contributing to this forward projection are to use the unique features and parameter space of C-Mod to complement results from other machines and to exploit further the uniqueness of C-Mod to test more rigorously theory and simulation. To help accomplish this, we will pursue an improved physics-based understanding of the pedestal profile characteristics, in particular the width of the ETB region and the gradients established within this narrow radial region. The interrelation of pedestal structure and edge relaxation mechanisms will receive further study, in both ELMy and ELM-suppressed regimes. We will also continue our investigation of the trigger mechanisms for H-mode in different configurations, and examine the role played by SOL plasma in determining H-mode power threshold. Tools for controlling pedestal transport and stability will be investigated, and connections with theory will be made using available computational tools for pedestal simulation. These plans are extremely well aligned with high priority R&D called for by the ITER program over the following three years. This work will also directly support the FY11 Joint Theory/Facility Milestone, the general goal of which is to produce pedestal projections of high confidence for burning plasma.

Experimental research will leverage an extensive and continuously improving set of edge diagnostics in which a great deal of investment has been made in the previous five years. Figure 5.8 highlights a number of the high spatial resolution (<3mm) diagnostics utilized (or planned) on C-Mod. Concentrated efforts will be made across many operational regimes to use measured profiles of plasma density and temperature, ion flow velocities (poloidal and toroidal), neutral density and radiated power to accomplish the experimental goals laid out below.

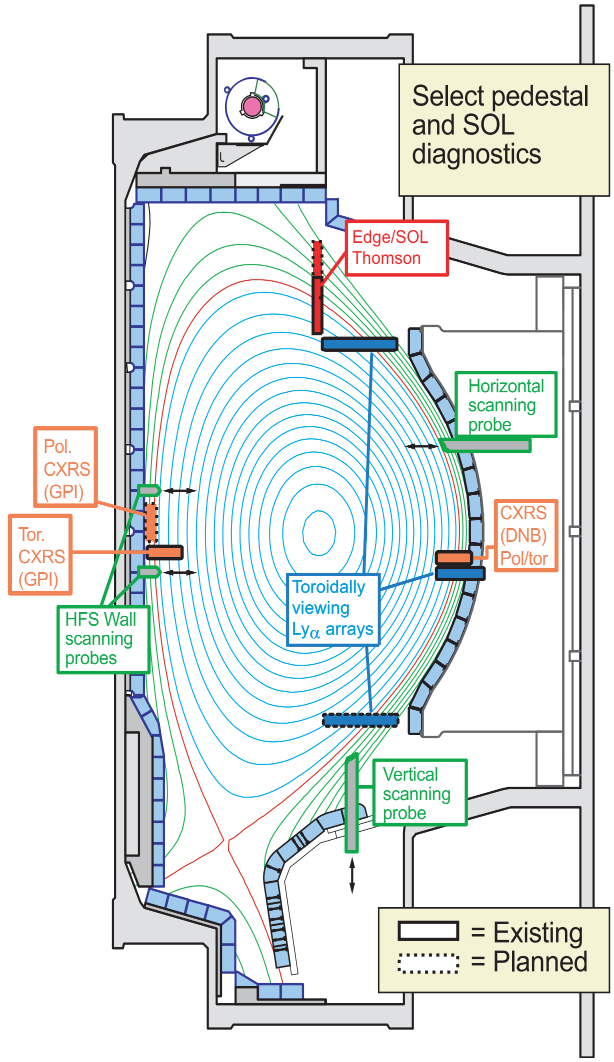

Figure 5.8: Existing and planned edge diagnostics pertinent to pedestal and L-H

Threshold power for H-mode access is of immediate concern to ITER, primarily because of uncertainties in projections based on simple scaling laws from existing machines. A joint task force has been formed between two ITPA topical groups (TC and PEP) in order to attack the question of the physics requirements for H-mode access, and how to relate that to input power. C-Mod is a contributor to this task force, and will participate in joint experiments to resolve some critical questions. Foremost is the identification of critical local parameters that determine the occurrence of L-H transitions. Edge CXRS measurements of impurity temperature and polodial and toroidal velocities will be crucial to these experimental efforts, as these measurements allow the inference of E_r profile in the vicinity of the separatrix. We will investigate the characteristic values of E_r shear associated with L-H transitions, as well as the concomitant pressure and velocity profiles. We will contribute well-resolved profile information to an international database (to be maintained at MIT), which can be used to examine radially dependent quantities like edge temperature, density, and flows, as well as their shear. Edge density fluctuations will be documented, and compared where possible with turbulence codes, as part of a path toward understanding the necessary conditions for L-mode turbulence suppression. In order to refine our discrimination of H-mode triggers, we will use our capability to operate for long durations in improved L-mode, and thus collect more data prior to the L-H threshold. Further analysis will be performed on data (referenced in Figure 5.6) from low-density L-H transitions, in order to uncover physical explanations for the low-density limit. Where feasible, new experiments will be performed in helium plasmas, as well as in deuterium plasmas, in order to address concern surrounding ITER's initial non-nuclear phase. Because ITER has requested an accelerated schedule for this task, considerable effort will come in FY09 and FY10.

Tied to the question of H-mode access is the means of H-mode *suppression*, and our optimization of improved L-mode. We will investigate the mechanisms that give rise to the drastically different levels of particle and energy transport, and work to actively control the suppression of the classical L-H transition and achieve steady state improved L-mode performance. We will continue to study the access conditions for this regime, as well as its pedestal and confinement scalings. Stability analysis of the edge will be performed and compared with that of ELMy and EDA H-modes. A mystery we intend to investigate is the driving physics behind the broad edge mode in improved L-mode, which appears to mitigate the density barrier. Possible similarities to the edge harmonic oscillation (EHO) first reported by DIII-D will be explored. Compatibility with ITER requirements, such as a detached divertor, will also be considered.

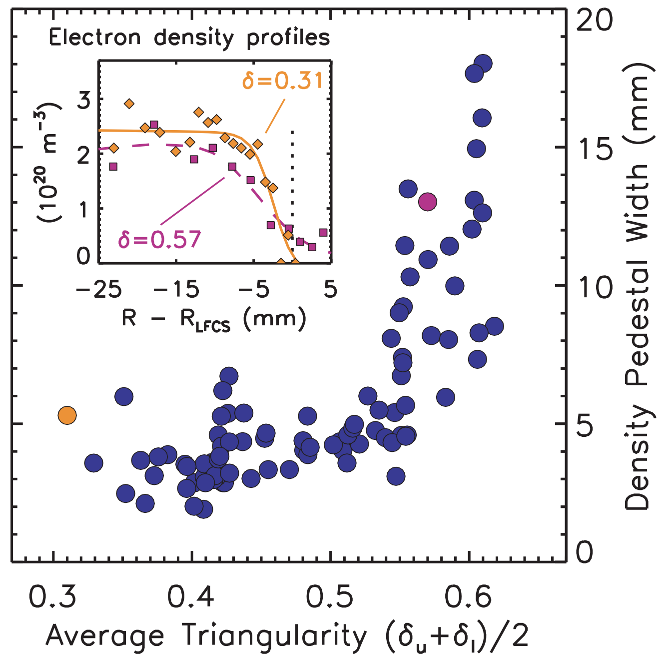

Figure 5.9: Measured density pedestal width in intra-discharge scans of traingularity. Sample profiles show differences in width

The physical mechanisms governing pedestal structure from another major research component on C-Mod, and will be an especially high priority topic for the next three years, in light of the FY11 Joint Milestone. Research in this area seeks to uncover scalings for pedestal width, and intrinsic limits on pedestal gradients, determined both by ELMs and in the absence of (or between) ELMs. In the Type I ELMy regime, additional parameter scans will be performed in order to more stringently test the poloidal beta scaling for the pedestal width discussed above. Other factors potentially affecting pedestal width will be tested across the gamut of H-mode regimes, including EDA. In the process of pushing toward less collisional H-modes, it will be possible to assess whether neutral sources become important at lower density. Inspired by results that show pedestal widening at both extreme shaping (as in Figure 5.9), as well as at high edge q, [13,14] experiments are being planned to test the influence of edge magnetic shear on the ETB width. This will be studied with broad scans in triangularity, elongation and plasma current; as lower hybrid power is increased in FY11, attempts will be made to modify magnetic shear near the edge actively using LHCD.

Pedestal models used to project toward ITER and other next-step devices typically postulate profiles limited by stability to Type I ELMs, while less consideration has been given to what sets profiles in ELM-free H-modes, or between ELMs. However, there is a strong impetus to reduce the ELM size on ITER, or to find feasible ELM-free regimes with good energy confinement. Thus, understanding the nature of the continuous transport processes that result in "stiff" pedestal profiles on C-Mod (and perhaps on other tokamaks) may be applied ultimately to model realistically edge profiles on next-step devices. We will continue to study the details of pedestal gradients, now with the inclusion of ion temperature data from CXRS, and test whether the

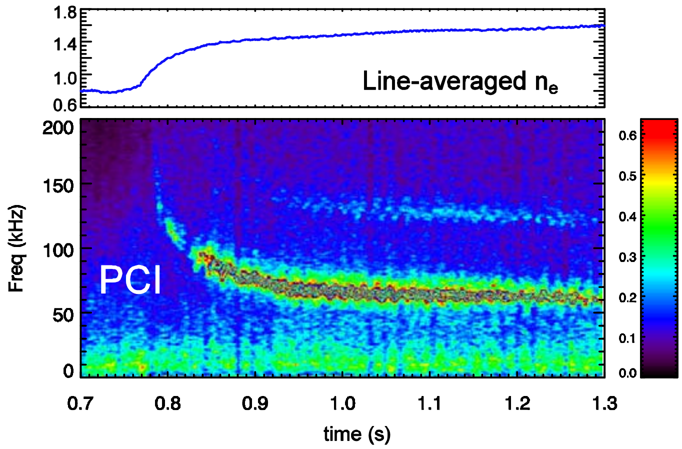

Figure 5.10: Steady core density achieved during EDA H-mode; Spectrogram of edge density fluctuations responsible for the

observed critical gradient phenomenology continues at lower values of edge ν*. The relationships between edge gradients and other local parameters (such as collisionality) will be tested further under alternate configurations (*e.g.* reversed field direction, extreme triangularity) and in Type I ELMy regimes.

Edge fueling of the pedestal will remain a focus area, particularly since C-Mod has the capability to operate with a neutral opacity in the SOL and pedestal approaching that of ITER. This feature of C-Mod (and ITER) results in inefficient fueling from the edge, and, according to 1D modeling [14], a density pedestal which is only weakly influenced by neutral source. Over the next two years, we will pursue 2D modeling of the ionization source in the C-Mod edge, in order to establish firmly the range of density over which the pedestal is insensitive to details of the poloidal ionization distribution. Experiments will also be designed to produce H-modes at the ITER value of B_T, $q95$, and determine the limits of edge fueling due to neutral opacity. Improved experimental measurements of neutral emissivity at multiple poloidal locations, highlighted in Figure 5.8, will be exploited as inputs into this interpretive modeling. This research will also benefit from more routine collection of upstream SOL T_e, n_e profiles, which will be provided by an upgrade to the Thomson scattering diagnostics suite. Contributions will be made to an ITPA multi-machine profile database for the purposes of modeling the sources in several tokamaks.

The primary goal for C-Mod's research on pedestal relaxation is to understand from first-principles the operational space for the various relaxation mechanisms. We seek a level of understanding for the quasi-coherent mode (QCM) that regulates the EDA pedestal, as well as other so-called "small-ELM" regimes, that is similar to the maturity of the peeling-ballooning model for Type I ELM triggers [20]. Pedestal structure, confinement and MHD stability will be analyzed in detail for the small ELM regimes that were developed and explored as a part of ITPA joint experiment PEP-16. We will continue experimental study of the QCM itself, employing the wide array of diagnostics that detect it. These include Phase Contrast Imaging, Gas-Puff-Imaging, reflectometry, probes, and magnetic pick-up loops. Since a single Langmuir probe inserted into the QCM region has the potential to perturb the mode, standard probe scans do not determine unambiguously the radial width of the QCM. An experiment using scanning Langmuir probes with radially spaced probe heads will resolve a long-standing question about the mode's radial extent. We will attempt to relate the QCM to similar continuous relaxation mechanisms, such as the broad oscillations observed in improved L-mode, as well as those observed on other devices, *e.g.* the EHO. Additionally, we propose to investigate the external stimulation of continuous edge modes such as the QCM using RF tools.

Further exploration of the operational space boundaries for the various relaxation mechanisms will occur as we utilize C-Mod cyropumping and shaping capabilities. In those plasmas exhibiting Type I ELMs (presently those with large δ_{lower}), we plan experimental studies of the boundaries in operational space, more detailed examination of the ELM energy losses, and more detailed examination of the structure and dynamics of the ejected ELM filaments. Analysis of all of the stability boundary results will be helped greatly by the acquisition of a suite of data handling codes [[21]], developed at DIII-D, that assemble and fit profile measurements of kinetic quantities, and then reconstruct plasma equilibria using those and magnetic data (so-called "kinetic EFITs"). In collaboration with the DIII-D group, we are implementing this capability.

In addition to mapping out the operational space for edge relaxation mechanisms and understanding the "natural" scalings of pedestal width, we will explore the use of available tools to actively modify these pedestal phenomena. A dramatic effect on pedestal structure was already observed upon application of LH power to an EDA H-mode (see Figure 5.4). We plan experiments to explore the range over which this technique can be applied, and to make sense of the physics that governs the change in particle transport. We will also examine the effects of LH on ELMy plasmas, both in terms of pedestal structure and ELM behavior. Ultimately (FY11 and beyond), when sufficient LH power is available, we propose to test whether application of LHCD near the pedestal can influence the frequency of the ELMs. This work will extend our efforts to characterize the role of magnetic shear on pedestal width, as well. As part of a newly proposed ITPA joint experiment (PEP-22), results will be compared with electron cyclotron current drive and heating techniques on other devices.

The C-Mod program engages theorists to provide experimental tests of predictions for pedestal scalings, including for example those for pedestal width, ion flows and momentum generation through residual Reynolds stress. Besides analytic theory, simulation and computation also form a key component of our plan. A substantial development effort has been undertaken in the computational community to develop powerful tools for edge transport and stability calculations in the pedestal region. Increasingly these computational tools will be used to support the C-Mod program. One of the more successful codes, ELITE, will be used for ideal MHD stability analysis within the context of the peeling-ballooning model for ELMs. Further work will also be done with the resistive MHD code M3D. We will also take advantage of the burgeoning array of edge transport codes in development. The XGC0 code is a neoclassical edge transport code, developed mainly to model pedestal structure and edge flows with a self-consistent radial electric field, while the recently developed XGC1 turbulence code calculates the contribution of electromagnetic turbulence to transport in 3D geometry, complete with an X-point. These codes will make possible improvements to pedestal modeling and more realistic calculations of flux-gradient relationships in the pedestal. Predictions for pedestal structure and scalings will be validated using C-Mod data.

Recently, increased focus has been placed on the integrated use of simulation codes, for the purposes of simulating complex time-dependent edge phenomena. Classic examples of these phenomena, which may be driven by multiple physical processes coupled in highly non-linear ways, include the L-H transition and the complete ELM cycle. While the goal of simulating these processes will not be realized for some time, progress in code development and integration is anticipated over the next three years, spearheaded by the SciDAC initiative known as the Center for Plasma Edge Simulation (CPES). We will work closely with the CPES to validate codes against C-Mod pedestal data and use the integrated workflow they develop to begin modeling time-evolving edge processes like H-mode evolution.

5.3 Research contributions to ITER and ITPA priorities

This proposal is well aligned with ITER R&D, including both "short-term" (2008—2010) and "medium-term" (2011+) priorities, as indicated by the "ITER Physics Work Programme 2009-

2011”, a document generated by the ITER Science and Technology Advisory Committee at an October 2008 meeting. The relevant priorities are paraphrased below.

Determination of the following auxiliary power thresholds in ITER-like plasma conditions, and development of strategies for minimization
H-mode, both Type III ELMy and Type I ELMy
H-mode with H98=1
In current ramp-up/ramp-down phases
The above, but in H and He plasmas
Pedestal width, pedestal energy and uncontrolled ELM energy loss in ITER
Determination of density, temperature and pressure pedestal width scaling versus dimensional and dimensionless edge parameters.
Quantification of the role of neutral penetration in the determination of the pedestal width and its interrelation with the pedestal pressure/edge power flux.
Determination of ELM energy loss dependence on dimensional and dimensionless pedestal plasma parameters.
Development of alternative regimes providing high fusion performance in ITER without or with small ELMs compatible with overall scenario requirements
Analysis of the role of plasma edge stability and plasma conditions for grassy ELM regimes and their extrapolability to plasma conditions required for advance regimes in ITER
Determination of the role of plasma edge stability and edge plasma collisionality for high collisionality regimes with small or no ELMs (Type II, EDA, etc.) and possible extension towards lower q95 and lower edge plasma collisionality conditions
Characterization of the Type III ELMy H-mode regimes both in conditions of low and high collisionality and evaluation of the pedestal pressure degradation with respect to Type I ELMy H-mode.
Development of new small ELM regimes and enhanced confinement features.
Development of alternative methods for ELM control/suppression in ITER and integration with scenario requirements
Demonstration of ELM control by stationary modification of the edge current distribution
Determination of mechanisms leading to plasma rotation and improvement of the physics basis to predict the expected level of plasma rotation in ITER.

The C-Mod plan is consequently able to contribute to specific research tasks put forward by the ITPA Pedestal Topical Group in recent work plans.

Improve predictive capability of pedestal structure
Test pedestal poloidal beta and ν^* scaling of pedestal width across devices and parameter regimes; develop the theoretical basis for this scaling
Explore dependencies of pedestal structure and ELM behavior on heating source. Quantify the impact of torque on the pedestal structure and ELMs
Establish pedestal conditions required for L-H transition through cross machine experiments and theory
Examine role of neutral penetration length on pedestal density width
Determine compatibility of divertor detachment and robust pedestal pressure
Incorporate comprehensive neoclassical and turbulence transport models into pedestal simulation codes
Improve predictive capability for small ELM regimes, quiescent H-mode regimes and ELM control techniques
Assess applicability of low collisionality small ELM regimes
Develop and test nonlinear MHD and turbulence models of ELM evolution

Current pedestal-relevant ITPA joint experiments in which C-Mod will plays a role in the next three years are

PEP-6: Pedestal Structure and ELM stability in double null
PEP-16: Small ELM regime comparison on C Mod, NSTX and MAST
PEP-22: Controllability of pedestal and ELM characteristics by edge ECH/ECCD/LHCD
TC-2: Power ratio – Hysteresis and access to H-mode with H~1
TC-3: Scaling of the Low-Density Limit of the H-mode Threshold
TC-4: H-mode transition and confinement dependence on ionic species

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6. Plasma Boundary Interactions

6.1 SOL transport

Heat Transport in the SOL

In preparation for a 2010 DoE Joint Facilities Milestone with DIII-D and NSTX on heat transport in the SOL (and divertor), we have reprioritized our research in the edge physics area. An extensive array of divertor heat-flux diagnostics will be deployed *much earlier* than originally planned – for the 2009 and 2010 run campaigns. The existing lower-divertor structure will be utilized for this. The new diagnostic set includes: (1) divertor surface temperature sensors, calorimeter sensors and bulk tile temperature sensors, (2) a new divertor tile thermography system, and (3) a new divertor bolometer array.

This reorientation will provide timely physics information on SOL heat transport. A key ingredient of the investigation is the 'mapping' of the narrow heat-flux channel from the outer midplane to the heat-flux "footprint" on the divertor surface. Scaling laws, which have been developed for the heat-flux channel width at the outer midplane [Kallenbach, A., et al., J. Nucl. Mater. 337-339 (2005) 381.], indicate a clear linear dependence on

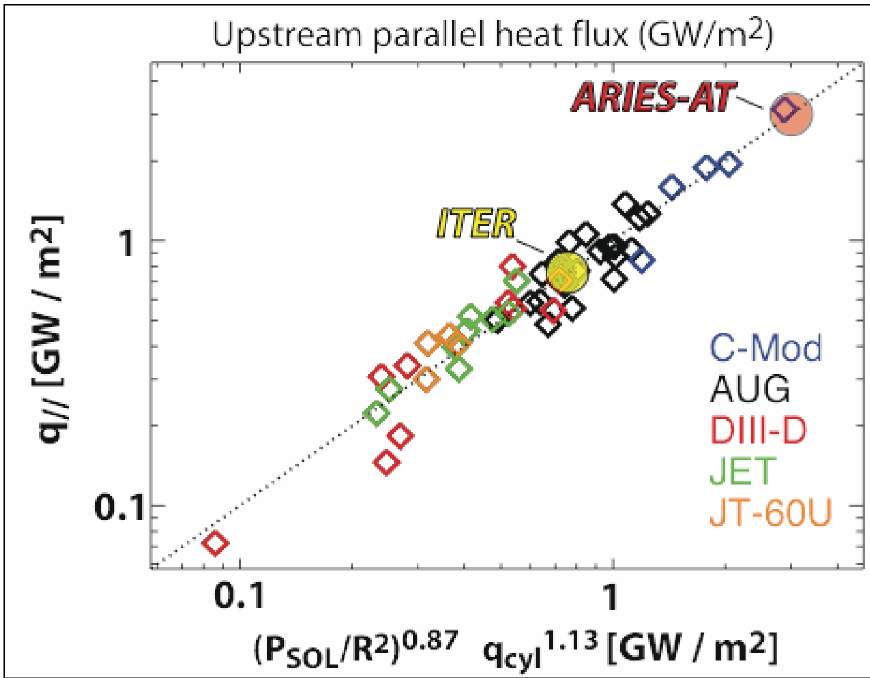

Figure 6.1 Measured or predicted (ITER and Aries-AT) parallel heat flux just outside the separatrix from a number of machines, including C-Mod, vs. regression scaling on device parameters P_{SOL}, R, and q. From the ITPA database [A. Kallenbach, et al. J. Nucl. Mater. 337-339 (2005) 381.]

tokamak major radius. But this observation stands in sharp contrast with the scaling laws assembled for the width of the heat flux 'footprints' [Loarte, A., et al., J. Nucl. Mater. 266-269 (1999) 587.] – these indicate no obvious major-radius scaling! As can be seen from Fig. 6.1, C-Mod has unique access to high power flux density regimes for this investigation (i.e., input power input per unit exhaust area approaching that of a DEMO). C-Mod's relatively small major radius also provides strong leverage in multi-machine experimental scaling relations.

Turbulent transport in the edge and SOL

Measurements on C-Mod and elsewhere have for some time shown the perpendicular particle transport in the SOL to be dominated by turbulence. C-Mod has developed and deployed a number of diagnostics that measure the characteristics of this turbulence. One of the workhorse techniques has been Gas-Puff-Imaging. For future run campaigns we are expanding and improving the suite of GPI diagnostics with a new 2D fast (2MHz) system deployed at the outboard midplane, an upgrade of the inboard SOL view to 2D, and upgrades of the X-point and midplane fast-camera views to quartz fiber bundles.

Recent "time-averaged" profile experiments (detailed below) have pointed to the importance of flow shear in the SOL. At the same time, GPI experiments have revealed two counter-propagating poloidal phase velocities for the turbulence in the separatrix/shear layer region. The phase velocities are observed to "flip" intermittently on a fast (~100 μs) timescale, and there is preliminary evidence that the velocity "flips" are correlated with the radial ejection of "blobs". With the diagnostic upgrades we will seek to document "microscopic" evidence of velocity shear effects upon radial transport and continue our investigation of dependences of the edge/SOL turbulence characteristics (velocity shear, structure scales, magnitude scaling, etc.) on local and global plasma parameters such as n_e, collisionality, $T_e(r)$, $P_e(r)$, and configuration. These will be coordinated with the extensive "time-averaged" measurements of edge/SOL characteristics made by the C-Mod reciprocating probe and Edge-TS systems. The upgraded systems will also be used to investigate the 3D structure and transport of the turbulent "blob" structures.

Time-averaged transport and SOL flows

Previous research on C-Mod has shown the importance of SOL flows for various plasma phenomena, e.g. L-H transition physics and plasma fueling. From time-averaged profile measurements there is also considerable evidence that asymmetry in perpendicular transport is a primary driver for SOL parallel flows. Furthermore, there is strong evidence from C-Mod for "critical gradient" transport in the near-SOL, steep gradient region. Now, recent experiments have shown that perpendicular *flow shear* plays a key role both in defining where the steep gradient forms in the SOL (i.e., the transition from 'near' to 'far' SOL regions) and also in setting the value of 'critical pressure gradient' that the steep gradients obtain.

During the next run campaigns, we will continue to explore the role of equilibrium plasma flows and flow shear on boundary layer transport and critical gradient phenomenon. For example, the role of flow shear as a precursor for the L-H transition is an important topic that begs for experimental investigations. We will also continue to investigate the important topic of plasma flows that circulate the confined plasma [LaBombard, B., et al., Nucl. Fusion 44 (2004) 1047.] and the transport physics that drives them. These studies have been helped by the use of specially developed high-heat flux “Gundestrup”-type scanning Langmuir-Mach probes and will be improved by the implementation of a new “Mirror Langmuir Probe” technique developed at MIT [LaBombard, B. and Lyons, L., Rev. Sci. Instrum. 78 (2007) 073501.] that could reveal the turbulence mode structure (drift-wave versus resistive- or ideal-ballooning) as well as to directly assess the role that flow shear may be playing in regulating SOL turbulence.

6.2 Plasma surface interactions

Hydrogenic retention

The study of hydrogenic retention in C-Mod has continued to be an important emphasis of the C-Mod group in a number of ways. Data from experiments in multiple run campaigns since 2005 have been analyzed and used to show that the retention observed in non-disruptive discharges is occurring in the molybdenum and tungsten tiles, most likely concentrated at the outer divertor. The retention is due to the implantation of the incident ion flux, which, after diffusion into the tile bulk, resides at potential wells in the tile lattice (traps). The retention is found to be ~ 1-2% of the incident D+ flux independent of density, plasma heating (Ohmic or RF) or confinement mode (L- or H-mode).

Most recently the research has concentrated on disruptive discharges where it was shown that, averaged over a run campaign, gas released during disruptions occurring during the flattop phase of discharges approximately balanced the retention occurring during the majority of discharges where there was no disruption. The research described above has been accepted for publication in Nuclear Fusion.

MIT scientists have also been leaders of inter-laboratory and international studies of what the current understanding is of fuel retention and how this applies to ITER. Dennis Whyte presented and published a study of the effect of neutron damage on fuel retention and what the implications are for ITER and future reactors [Whyte-PSI08]. Bruce Lipschultz and Dennis Whyte both played a central role in an international workshop held at MIT to assess the current understanding of all the processes leading to hydrogenic retention in a range of materials and how we can apply that knowledge to predicting retention in ITER.

The study of hydrogenic retention over the next two years will primarily emphasize the analysis of existing data. There will be continued analysis of the fuel recovery during disruptions – the dependence on the kind of disruption, the effect of sequential disruptions, the effect of the amount magnetic vs. plasma thermal energy at the time of disruption. This information, together with the existing database of retention in non-disruptive discharges will serve as the basis of inter-laboratory studies for the 2009 DoE Joint Facilities Milestone together with DIII-D and NSTX. Nascent numerical modeling of fuel transport in the Mo/W will be applied to these experiments to better understand their implications.

Given that the C-Mod retention fraction (1% of incident ion flux) is significantly higher than that found through linear plasma machine retention studies we have initiated a study of samples from C-Mod in the PISCES plasmas as a function of temperature and boron content in the material. Of course an important part of the future work is development of the in-situ (in C-Mod) ion beam-based analysis of tiles that is based on a 1 MeV D+ beam and the nuclear products it engenders in the near surface of tiles. That project, funded separately by a DoE diagnostic development grant with Dennis Whyte as PI, has already started. The upcoming work includes refurbishment of the existing MIT RFQ accelerator, development of detectors and installation on C-Mod. Lastly, an important development for future work is the Demo-like divertor (see section 6.3 below) whose specifications include the capability to vary the outer divertor temperature to examine the effect of temperature on retention in a tokamak.

ICRF sheath rectification

The erosion of molybdenum and resultant levels in the core plasma are clearly a limiting factor in C-Mod operation. Over the past year there was a re-analysis of 2001 run campaign data that showed that the installation of boron nitride tiles on the antenna limiter did not affect the enhancement of plasma potential on field lines in the shadow of the limiter (and thus Mo erosion). This surprising result led to a reexamination of the potential models of plasma potential enhancement. We have recently started a collaboration with J. Myra of Lodestar on potential processes that could lead to sheath enhancement and that has led us to plan a number of new diagnostics for the 2009 run campaign and beyond. Since the emissive probe is not a robust diagnostic (the hot filament tends to fail) we have developed an alternative diagnostic of plasma potential that measures the energy distribution of secondary electrons using a much more robust 'ball-pen' probe design. We are planning to install sheath diagnostic platforms consisting of an emissive probe, a ball-pen probe and RF probes at 3 locations in the SOL as well as in the Surface science station (S^3) for 2009. The RF probes will assist us in identifying whether the local RF wave is a slow wave as one model of enhanced sheaths predicts [PRL08 Myra]. Depending on those results we may also expand the coverage in subsequent campaigns.

Material migration

During the past year we initiated a study with J. Brooks (Purdue) using the WBC code to model the C-Mod erosion data first published ca 1990 [Pappas, Wampler]. The idea was to finally bring to bear local modeling of erosion and re-deposition to bear on the surprising experimental result that roughly half of the gross erosion is re-deposited locally; preliminary results show that given the large Mo mass and the high magnetic field, that the re-deposition should be closer to 100% and net erosion significantly smaller than measured. That work, will continue into FY10.

We initiated another study in FY08 to look at material migration over a larger spatial scale than the Brooks study. We have removed a poloidal set of Mo tiles from the inner and outer divertor following the 2008 run campaign. Since there is a complete row of tungsten tiles at one poloidal location in the high heat flux region of the outer divertor the analysis of the level of tungsten on the poloidal set of tiles will give us the campaign-integrated migration of eroded W from the W tiles. This is first a gross measure of W migration which can potentially also be used in modeling studies.

6.3 DEMO-like divertor

We have finished an initial study of the proposed demo-like outer divertor for C-Mod. This new divertor, planned for installation in FY12, is envisioned to have a simplified vertical geometry that will have all tungsten tiles in the high heat flux region and will be heatable up to 600 ^{o}C. The initial engineering studies of the issues associated with supporting the hot divertor as well as how that divertor heats other components in the chamber indicated that we needed to do a better job of modeling the IR heat transfer between the hot outer divertor and the rest of the chamber including diagnostics. To this end we have initiated a collaboration with the PPPL engineering staff to build a model of that heat transfer and we have proposed 2 potential additional changes to the divertor region: 1) change the tiles over the EF1 pocket to raise them up close to the x-point to reduce the IR coupling of heat from the outer to inner divertor; and 2) determine whether it is possible to thermally isolate the tiles on the approximately horizontal section of the outer divertor. The design study work will continue over the next year with a parallel engineering design, again as collaboration with PPPL.

7. Wave-Plasma Interactions

7.1 Ion Cyclotron Range of Frequencies

C-Mod provides a unique opportunity 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 the availability of advanced diagnostics for RF wave measurements. The primary core physics research areas are simulation validation of wave propagation and absorption, characterization of flow and current drive, and RF and plasma physics integration. ICRF utilization is often limited as a result of the antenna performance. We have identified a number of physics and technological issues where C-Mod can contribute: coupling physics; compatibility with high performance discharges and metallic plasma facing components; reliably maintain coupled power despite load variations; availability to deliver ICRF power on demand without burdensome antenna conditioning.

7.1.1 Highlights

The first observation of toroidal and poloidal flow by mode converted waves was achieved. In Figure 7.2, the central rotation velocity (flow) for two discharges with similar plasma parameters but heated by the mode conversion (red) and minority (blue) absorption scenario respectively. The toroidal flow velocity (flow), derived from the Doppler shift of the x-ray spectra of Ar impurity measured by high resolution x-ray spectroscopy (HIREX), is significantly higher in the mode conversion heated plasma than that in the hydrogen minority heated plasma. We found that the velocity change, ΔV_ϕ, scales approximately linear with injected power, indicating a mechanism dependent on power dissipation rather than directly related to the RF field strength. While the magnitude ΔV_ϕ phase is larger for co-current wave injection than those in heating and counter current wave injection phases, the difference is fairly small (<10%) and remained in the co-current direction for all phases. When compared with the empirically determined intrinsic rotation scaling law the rotation change in these MC plasmas is generally a factor 2 larger than that from minority heated plasmas with identical plasma shape, magnetic field, current, and density. The interaction of the mode converted wave with ions appears to be important for flow drive. Experimentally, no

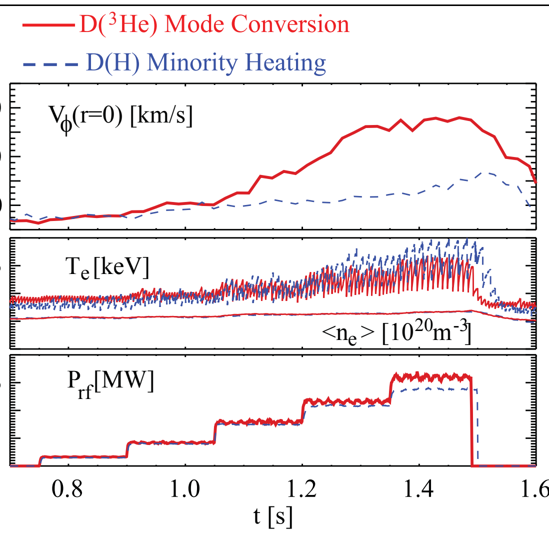

Figure 7.□ SEQ Figure * ARABIC □2□: Comparison the toroidal rotation response for two ICRF heated discharges where the mode conversion and minority absorption scenarios are

flow drive has been observed when majority of ICW power is absorbed by electrons and TORIC indicates significant power to the ions when rotation is observed.

To inject ICRF power, the antenna is situated near the plasma edge and one of the primary ICRF utilization challenges is to reduce/eliminate specific ICRF impurity production. Previous prescription to ameliorate impurity production was developed for experiments with carbon plasma facing components (PFC). With metallic PFCs, ICRF impurity production needs to be reduced to be compatible with high performance discharges. In C-Mod, low Z-films are often utilized to ameliorate ICRF generated impurities and ITER plans to utilize Be armor. In C-Mod, the low-Z film erosion rate is estimated to be 15-20 nm/s for ~3 MW injected power indicating the eroding species energy is much higher than that normally found in the SOL. Using emissive probes, plasma potential measurements confirm the presence of an enhanced sheath with ICRF when the probe is magnetically linked to the active antenna. Plasma potentials were typically 100-200 V for ~1.25 MW injected ICRF power. As shown in Fig 7.2, the plasma potentials are about twice as large in H-mode than L-mode for the same injected power. Furthermore, an RF sheath was unexpectedly present with insulating limiters. These observations suggest that other scrape-off layer and plasma-surface characteristics are influencing the resulting RF sheaths and require refinement of the RF sheath model.

A real-time ICRF antenna matching system has been successfully implemented in the E antenna matching network. The network is a triple-stub tuning system working where one stub is fixed and a pre-matching stub and the other two stubs use fast ferrite tuners (FFTs) to accomplish real-time matching. The system utilizes a digital controller for feedback control (200μs per iteration) using real-time antenna loading measurements as inputs and the coil currents to the FFT as outputs. The system has achieved and maintained matching, 1% power reflection, for a large range of plasma parameters, including L-mode, H-mode, and plasmas with edge localized modes (maximum mismatch of 10% in power). The system has succeeded in delivering up to 1.85 MW into H-mode plasmas at maximum voltage of 37 kV on the unmatched side of the matching system.

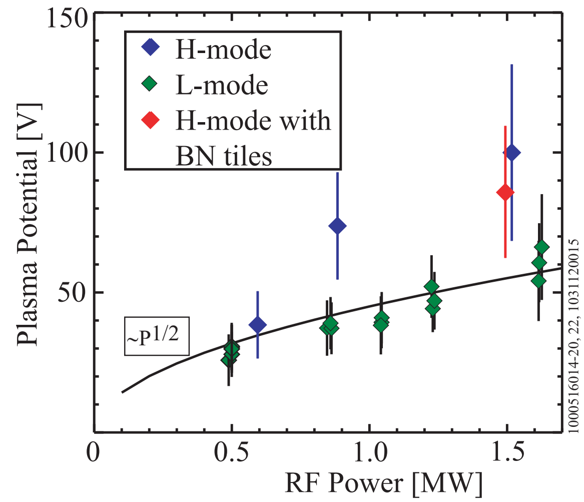

Figure 7.3: Plasma potential scaling with injected RF power for L-mode (green) and H-mode (blue). Also plotted is the plasma potential in H-mode with insulating RF limiter tiles which were expected to eliminate the RF sheath.

7.1.2 Plans

ICRF Wave Propagation and Absorption

ICRF is utilized in C-Mod for the bulk auxiliary heating and relies primarily on D(H) minority heating scenario (minority in parentheses). The presence of energetic ions in the plasma may influence ICRF wave absorption and modify the deposition profile and energy partition between ions and electrons. Measuring the resulting hydrogen ion distribution provides an opportunity to validate ICRF upgraded simulation capability that includes non-Maxwellian ions, AORSA-CQL3D. The Compact Neutral Particle Analyzer (CNPA) is the primary diagnostic to measure this fast ion distribution function and a fast ion charge exchange (FICX) will be available, through collaboration with UT-FRC, to measure the fast ion spatial distribution. Experiments where the minority tail energy and spatial distribution is varied, by injected power for example, would provide a benchmark for simulation. Thus, simulation algorithm and physics kernel could be assessed.

In ICRF mode conversion, the long wave length fast wave mode converts to two short wavelength modes, ion Bernstein and ion cyclotron waves. Using Phase Contrast Imaging (PCI) diagnostic (DoE Diagnostic Initiative), the fluctuation profile of the mode converted waves can be measured. Since the original work[[22]], the system has been upgraded to measure higher wavenumber (k) and its calibration improved. Furthermore, the synthetic PCI diagnostic in TORIC has been upgraded in collaboration with the RF SciDAC Group and the agreement between previous experiments and simulation amplitude has been significantly improved. Thus an outstanding issue regarding amplitude discrepancies can potentially be resolved because of these diagnostic and code improvements. The studies will initially focus on D(^{3}He) and D(H) mode conversion scenarios. Since mode conversion is critically dependent on the minority concentration, one of the principal experiments will be to measure the RF density fluctuation profile with PCI and compare with the simulations. This is enabled through collaboration with the RF Sci-DAC Initiative where we have local access to the MARSHALL and LOKI Beowulf clusters to perform full-wave simulations routinely.

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 a challenge for simulations to accurately predict power deposition. An outstanding question regarding D(^{3}He) is the role of parasitic absorption, e.g. B minority resonance, loss mechanisms near the plasma edge. A similar parasitic minority resonance will be present in ITER from the planned use of Be coating on plasma facing components (PFC's). In the near term, experiments are planned to move the parasitic ion edge resonances either into the plasma core or out of the plasma. We also plan to continue experiments where a direct comparison 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 will investigate the effect of additional heating power on heating efficiency with higher power density and consequently higher bulk plasma temperature, where we expect to increase the single pass absorption in D(^{3}He). This work will also take advantage of new simulation capabilities where self-consistent ion distributions can be evolved within the simulation to investigate the impact of energetic ions on wave absorption.

An emerging area of interest is second harmonic heating due to both its utilization planned in ITER and theoretical calculations which suggest that it may have stronger damping than previously thought. With the compact neutral particle analyzer[[23]], the ion distribution can be measured on C-Mod and used to benchmark the simulations. A code to code comparison (often referred to as code verification) was performed on the ITER scenario 2 which is dominated by second harmonic absorption and electron Landau damping. Initial experiments will investigate second harmonic H absorption and allow benchmarking of the codes to experiment (code validation). Future experiments could examine the role of parasitic edge ion absorption, predicted by the codes which could have significant effects on impurity generation.

Mode Converted Flow Drive (MC)

Rotation control offers an opportunity to increase plasma stability and improve plasma confinement and development of an ICRF flow actuator is a high priority. The initial observations of both toroidal and poloidal flow have shown that the existing theory is inadequate to describe the observations.[[24]] We seek to characterize and optimize the flow drive with respect to RF and plasma parameters. In particular, we wish to test the flow drive dependence on ion versus electron absorption of the mode converted waves and this will be investigated via a minority concentration scan. Both plasma density and current influence the wave characteristics and one may also expect an inverse density relationship for a constant power. We will also investigate flow drive in H-mode and He and H majority discharges. We can also utilize a narrow magnetic field scan to vary the power partition between IBW and ICW waves and utilize 80 MHz at 8 T to characterize flow drive with higher frequency and toroidal field. The results from these experiments may provide insight and direction for the theoretical model development. Once a theory is proposed, we would perform the necessary experiments to test theoretical model. In conjunction with the transport group, RF driven flow could be utilized to investigate the RF power required to affect transport and trigger or maintain internal transport barriers.

ICRF Current Drive

While not expected to be as efficient as LHCD, ICRF current drive (mode conversion (MCCD) and ion cyclotron minority (ICCD) current drive) can be used to tailor the local current profile for controlling instabilities and Fast Wave current drive (FWCD) can provide the central seed current for fully non inductive advanced tokamak scenarios.

We have investigated MCCD both experimentally and theoretically. We have found that MCCD is a good candidate for sawtooth pacing where the local current profile is modified to destabilize the sawteeth.[25] This should be beneficial for high performance discharges where large sawtooth crashes can terminate the high performance phase, and also may prevent neoclassical tearing modes (NTM) by reducing or eliminating the seed island. In conjunction with PCI, MCCD experiments can provide important information about the wave conversion and inherent up-down asymmetry associated with the mode converted waves. Mode conversion experiments with heating and current drive phasing have shown that the heating phase may have net driven current. This would be consistent with the up-down asymmetry in the mode converted spectrum predicted by simulation. Further experiments are planned to test this prediction and these experiments will provide a good test of simulations and their associated current drive models.

An important application of MCCD for C-Mod is sawtooth pacing where the sawtooth period and amplitude are kept short to avoid the crash of monster sawteeth. We have shown that the sawtooth period can be shortened or lengthened by changing the antenna phasing or deposition location. A principle question yet to be addressed is sawtooth pacing in the presence of a stabilizing energetic ion population. We plan to add MCCD to discharges with monster sawteeth to investigate the current drive power required to pace the sawtooth period. These experiments will be performed within the framework of the ITPA experiment MDC-5.

For the reference C-Mod advanced tokamak discharge, a central seed current of approximately 20 kA is required.[26] Fast wave absorption on electrons is a strong function of plasma β giving a centrally peaked absorption and current drive profile. Initial experiments will focus on the deposition profile in a variety of conditions to determine the deposition profile and absorbed power fraction. The absorption fraction should increase as higher β plasmas are obtained. Another issue will be impurity generation due to the low single pass absorption anticipated for these discharges. Furthermore, the current drive experiments will provide added data to experimentally benchmark the simulation codes and their respective current drive models.

A third current drive technique, ICCD, should be useful for local current profile tailoring, in particular sawtooth pacing. The physics of ICCD is complicated but essentially there are two regimes: classical and finite orbit. In the classical regime current is carried by passing particles and has peak efficiency near the critical energy. The finite orbit regime generates current as a result of finite width trapped particle orbits. C-Mod has access to both regimes and will allow benchmarking of the simulation codes. Furthermore, the two regimes have opposite dipole current profiles and will have opposite affect on instabilities, like sawteeth. Experimental time devoted to this current drive mechanism is resource dependent.

Antenna coupling and Antenna/Plasma interactions

A number of physics and technological issues can be addressed at C-Mod relevant to ITER and future devices: coupling physics; compatibility with high performance discharges and metallic plasma facing components; reliably maintain coupled power despite load variations; and availability to deliver ICRF power on demand without burdensome antenna conditioning. At C-Mod, antenna operation of the two 2-strap (D and E antennas) and a 4-strap (J antenna) antenna at high power density (>10 MW/m^2) has become routine. We plan to install a new four strap antenna in FY10 to replace the J antenna and if successful a second in FY11 to replace the D and E antennas. One of the primary design goals is to reduce the RF sheaths by antenna design.
The compatibility of high power ICRF with all metal plasma facing components (PFC) and high plasma performance is a critical issue for C-Mod and future devices, such as ITER. Erosion of low Z films and impurity production are likely a result of enhanced sputtering due to RF sheaths.[27,28] A new model indicates the RF sheath is a result of slow wave propagation in the scrape-off layer.[29] Past results and the new model motivated the re-instrumentation of the plasma limiter with plasma potential and RF probes. Measurements from these diagnostics will allow a characterization of the RF sheaths with RF and plasma parameters. In the future, we will be able to compare the RF impact on plasma potential between different antenna designs. To identify erosion/impurity sources, we have installed molybdenum tiles coated with 0.003" of boron on the outer divertor shelf, plasma limiter, and RF limiters. After the campaign we will again measure the B thickness; thus, be able to identify the locations of greatest erosion. If linked to an antenna, we will have identified a location where RF sheaths are important. Another outstanding question is impurity penetration. One thought is that the convective cells driven by the RF-enhanced sheaths may be responsible for enhanced impurity penetration from this location. With the installation a new prototypical ITER ICRF antenna reflectometer (in collaboration with ORNL), we can measure the local density near to determine the extent of the up-down density asymmetry near the antenna and infer strength of the convective cell through modeling. We also need to develop a technique to improve the boronization lifetime or reduce the impurity influx. Low Z-coatings deposited by vacuum plasma spray onto specific tiles are one possibility but unlikely to translate to fusion grade plasmas. Another means is to reduce the parallel electric field excited by the antenna to reduce the enhanced sheath potential and the strength of the convective cells through changes to the antenna design.

Although ICRF coupling or antenna loading in C-Mod is quite robust, the development of tools and techniques to manage or control coupling would greatly improved antenna performance in future experiments. The new scrape-off layer (SOL) reflectometer will allow direct monitoring of the SOL density profile. The influence of ICRF power and attempts at SOL density profile control through gas puffing are among the first items to characterize. The latter experiment is part of ITPA ISO 5.2 and C-Mod occupies a unique parameter space in that the SOL is opaque to neutrals as expected on ITER.

Plasma load variations are commonly encountered during L/H transitions and edge localized mode activity (ELM's). In the past campaign, we have successfully deployed a fast matching system into the E antenna matching network injecting up to 1.85 MW. The fast ferrite tuning (FFT) system is designed to perform real-time matching by varying the effective electrical length of the stub tuners via currents in the magnetic coils surrounding the ferrite materials. Implementation of additional systems is expected in the near future. 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.

We plan to continue our collaboration with the RF SciDAC Group on the validation of an electromagnetic solver that has a realistic ICRF antenna geometry coupled to 3-D plasma field solver. In the near term, we will revisit these measurements with the SOL reflectometer providing the density profile information lacking the in previous comparison. 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 ELM's.

In the future, the antenna voltage and power handling requirement is expected to be more difficult. Furthermore, the reliability to deliver ICRF power on demand without burdensome antenna conditioning is also expected to become more challenging. We have developed a small test facility to investigate the role of magnetic field and materials in RF breakdown. We plan to evaluate a series of materials to identify materials with improved breakdown characteristics. We previously found that the breakdown pressure is significantly lowered when the external magnetic field is parallel to the RF electric field. An outstanding question for the so-called neutral pressure limit is the initial fault. With our upgraded device, we should be able to replicate the neutral pressure limit condition. In this device, we will also be able to characterize the changes to the material surface as the structure is conditioned.

Support for ITER and Connection to ITPA Activities

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.

ITPA-ITER High priority Research Tasks

Improve characterization and understanding of rotation sources.
Characterize the level and processes involved in RF enhancement of erosion.

ITPA Joint Experiments:

TC-14 RF-driven rotation
IOS-5.2 Maintaining ICRH Coupling in expected ITER regime
MDC-5 Comparison of sawtooth control methods for NTM suppression.

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7.2 Lower Hybrid Range of Frequencies

The main motivations for current drive experiments on Alcator C-Mod are two-fold: i) to augment the bootstrap current, and possibly a residual inductive current, in order to produce either fully steady-state or hybrid modes in C-Mod that extrapolate to the Q = 5 target in ITER; and ii) to inform the decision on adopting LHCD for ITER by developing experimentally benchmarked simulations that can reliably predict the benefits of adding lower hybrid current to ITER's HCD portfolio.

7.2.1 FY 2008 - 2009 Progress

Up to 1.2 MW of RF power in the lower hybrid range of frequencies (4.6 GHz) has been coupled to Alcator C-Mod plasmas by means of a grill composed of 88 waveguides arranged in 4 toroidal rows of 22 waveguides each. The parallel index of refraction $n_{//} = k_{//} c / \omega$ of the launched waves is controlled by setting the phase progression of the electric field at the mouths of the waveguides facing the plasma; for example, with a 60° phase progression, the predominant component of the spectrum of launched waves is at $n_{//} = 1.6$, while at 90°, it is at $n_{//} = 2.3$. The phase progression, as well as the amplitude of the field at each waveguide mouth, is electronically controlled in real- time by an IQ modulator inserted between the output of a master oscillator and the input to each of 11 klystrons. Each klystron generates up to 250 kW of RF power at f = 4.6 GHz. Although the klystrons are capable of CW operation, the pulse length of the RF system is limited by the high-voltage power supply to ~ 5 s, which is also the maximum discharge length achievable in Alcator C-Mod at B_T = 5.4 T. The 5 s pulse length is substantially longer than the current redistribution time in Alcator C-Mod, which is typically 100 -300 ms. As previously reported, LHCD has been successful in transiently reversing the loop voltage and holding it below 0 V for durations as long as 200 ms in 1 MA Alcator C-Mod discharges. Stabilization of sawteeth is routinely observed in L-Mode discharges, indicating that off-axis current drive due to LHCD is sufficient to increase the central safety factor above unity.

Highlights from the past 2 campaigns include:

- MSE measurements of LHCD current density profile;
- Comparison of LHCD results with simulations based on ray-tracing and Fokker-Planck codes, from which fast electron diffusion coefficient and a possible pinch term are derived;
- Observation of LHCD-induced counter-rotation and investigation of possible mechanisms;
- Coupling of LH waves into ICRF-heated plasmas, with favorable pedestal modifications;
- Proof of principle detection of LH waves via reflectometry;

- Observation of a possible density limit and exploration of underlying physics;

- Development of a new full-wave simulation tool that allows full calculation of LH wave fields from klystron to plasma core;

We describe a few of these highlights below.

MSE measurements

MSE measurements of current profile as a function of $n_{||}$ have been made and compared with simulations using the ray-tracing code GENRAY together with the Fokker-Planck code CQL3D. As expected, the current drive efficiency decreases with increasing $n_{||}$ as indicated by the data shown in Figure 7.3. In these discharges the central safety factor increases above unity for $n_{||} \leq 2$, and sawteeth cease, while for $n_{||} > 2$ and there is only a minor effect on sawteeth. Figure 7.4 shows that the current profile broadens during LHCD by comparing the fraction of total current inside the radius r/a = 0.44 with that outside this normalized radius for identical discharges with and without current drive. One again sees that the current drive efficiency is higher for the lower value of $n_{||}$. In the case of high efficiency, new MHD activity suggestive of a tearing mode is sometimes observed. This is further indication of a strong perturbation to the current profile.

The total RF-driven current predicted by simulations, including the effect of a residual electric field, is in fairly good agreement with that measured as are the predictions of the energy-resolved profiles of fast-electron Bremsstrahlung emission derived from a synthetic diagnostic in the simulation. However, the simulated current profile tends to have more structure than that indicated by the MSE measurements, although the latter is somewhat sensitive to the method in which the current density is calculated from the observed change in poloidal field. One can force better agreement between simulation and experiment by adding fast electron diffusion, and possibly a pinch term, to the GENRAY-CQL3D model. The dynamics of the fast electron population may play a role in the observed LHCD-induced counter rotation described below, and will continue to be a focus of LHCD experiments during the 2009-2010 campaigns.

LHCD induced rotation

LHCD-induced counter rotation is one of the most interesting and unexpected results from LHCD experiments carried out thus far. Figure 7.5 shows a discharge in which strong counter rotation develops after application of LH power and profiles of various discharge parameters are shown in Figure 7.6 [1]. It can be seen from Figure 7.6 that the rotation is confined to the core, in contrast to ICRF-induced rotation which is broadly distributed across the entire profile. The strongly peaked negative radial electric field suggests that the rotation might be associated with a population of fast electrons in the plasma core. One possible explanation is that these electrons accumulate there due to an RF pinch effect [2], similar to the action of the inductive electric field on trapped particles which gives rise to the Ware pinch. Another possibility is that the fast electrons diffuse in the turbulent fields more slowly than the those in the bulk. In order to restore the ambipolarity, a radial electric field is established which drives the rotation. It is also interesting to note that the rate of LH-injected wave momentum is sufficient to explain the rate of momentum buildup in the core plasma. Assuming most of the current is carried by the fast electrons leads to a comparable rate of momentum transfer. Clearly LH induced rotation is an interesting result with possible implications for ITER. It will be the subject of additional investigation in the upcoming campaigns. An additional piece to the puzzle is provided by pinch and diffusion terms that can be extracted from fast electron bremsstrahlung profile measurements, an activity which is in progress and is yielding intriguing results.

Coupling into ICRH H-modes

One of the issues that had been a concern for the LHCD experiments was whether adequate coupling efficiency could be maintained during an H-Mode. The concerns were i) whether there would be adequate density in the SOL plasma to ensure that $\omega_p > \omega$ at and beyond the grill; ii) whether there would be destructive interactions between the ICRF and LH antennas that would pose problems for the LH antenna and reduce its efficiency; and iii) whether LH waves would penetrate the pedestal where the WKB condition $k^2 / k' >> 1$ is violated. The first two concerns have proven to be non-issues, *except* when the ICRH antenna is magnetically connected to the LH grill. This occurs only in the case that the antenna located in the port adjacent to the LH grill is excited (D-port). Thus we use the only the E- and J-port ICRH antennas in discharges when both LH and ICRH antennas are simultaneously driven. Propagation through the pedestal is being examined with the full-wave simulations mentioned above in the highlights, and preliminary results suggest that some loss of coupling efficiency might be expected, depending on the strength of the pedestal gradient. These simulation results will be expanded on in the coming year, and will be experimentally corroborated using measurements of the SOL and pedestal density made with a newly installed reflectometer.

While the density in C-Mod H-modes is generally too high for LH waves to drive significant current at present power levels, an interesting and potentially significant result in H-mode has been discovered. As shown in Figures 7.7 and 7.8, applying LH power to an H-mode plasma can cause a pumpout of density as well as a change in rotation in the pedestal. This leads to an increase in the pedestal temperature while maintaining constant pedestal temperature. As higher pedestal temperatures lead to improved confinement, a tool to reduce the pedestal density would be useful for confinement optimization. As this is a newly discovered effect, it is not understood whether the pumpout is due to current driven in the pedestal region or other RF effect. Clearly this is a fertile area for further investigation in the upcoming campaigns.

7.2.2 Plans for 2010-2011 Campaigns

While the results of LHCD experiments are in large measure in accord with our understanding based on theory and simulation, a somewhat disappointing aspect of the LH experience to date is the limited power (≤ 1.2 MW, about half of the installed power) that has been successfully coupled. There are two issues to be addressed here, namely i) relatively high losses in the transmission system (~ 50 %), which limits the maximum available power to ~ 1.5 MW, and ii) breakdowns in the splitter and launcher system which often occur at a coupled power level of 700-800 kW. The positive aspect regarding item ii) is that with very few exceptions we have not experienced breakdown in the coupler containing the vacuum windows nor at the coupler-plasma interface. Nevertheless, success in using LHCD to develop nearly steady-state advanced modes will depend in mitigating these issues, as well as adding more RF capability. *Thus, the key element in our proposal for the coming years is to increase the available klystron power to 4 MW and to develop simpler launchers that have both lower loss and improved power handling capability.*

We currently have 10 of the original klystrons working up to specification. To this complement, we plan to add 7-8 new tubes, with 3 built from components that have already been purchased and the rest from a new procurement. We are fortunate to be able to take advantage of a large klystron procurement contract (24 tubes) that has been made by the Chinese Institute for Plasma Physics with the same vendor, which has brought the unit price down substantially. These klystrons are identical to the klystrons used as the source of power in our LH experiments, with the exception of a more robust collector. Sixteen tubes would then provide a robust 4 MW of source power, and the remaining 1- 2 tubes would be used as spares.

The second element in our plan is the development of simplified couplers that would have lower loss and higher power handling capability. Figure 7.9 shows a key component of the new design, as well as an isometric view. The main simplifying features relative to the present launcher are summarized in Table 7.1. The new coupler is simpler and less expensive to fabricate than the one used so far, and we expect that it will have significantly higher power handling capability as well as reduced losses.

Present Design	**New Design**
Three 3 dB power splitters	One 3 dB power splitter
Forward and rear waveguide assemblies formed from 25 plates with joints in areas of high RF current	One continuous waveguide feeding two adjacent columns, no joints
24 simultaneous window brazes required	Windows are brazed one at a time

Table 7.1 Comparison of new LH coupler design with present approach

The schedule for the new coupler calls for it to be installed for the 2009 campaign. The ten klystrons from the original set will be used to power the new launcher. Much of the FY09 campaign will be used to shake down and evaluate the performance of the new launcher. Assuming that it performs well, it will be reproduced and a second launcher will be installed for operation in FY 2011. With 2 MW source power for each launcher, we expect in FY 2011 to couple up to 3 MW of LH power into C-Mod plasmas. Based on our measured current drive efficiency, this is should be sufficient to drive ~ 300 kA at the densities required for our advanced scenario simulations. The objectives for this aspect of the LHCD program are more fully described Section 3.

The secondary goal of the C-Mod LHCD program is to develop the physics and practical aspects of LHCD for informing a decision on installing LHCD on ITER. The novel coupler that is being developed for the upgrade of the LH source power for Alcator C-Mod experiments may in fact have relevance to the design of a relatively simple and efficient LH launcher for ITER. Thanks to the encouragement of the ITER STAC, a proposal for a 20 MW LH system is being prepared for ITER. The reference frequency is 5 GHz, which is close to the 4.6 GHz used in Alcator C-Mod experiments. The effort to prepare the ITER proposal is being led by the European Party and members of the C-Mod LH team are actively involved. It is estimated that such an LHCD system could save about 30% of the V-Sec required for startup, which would be very significant both in facilitating low internal inductance startup as well as in extending ITER's pulse length. The LHCD program on Alcator C-Mod is highly relevant in providing both scientific and technological support for the ITER LH system proposal

The essential task in developing the physics of LH current drive for ITER is to develop a comprehensive, validated simulation (or set of simulations) that can be used to confidently predict the performance of a LHCD installation on ITER. This requires a detailed understanding and modeling of:

> Plasma-grill interactions, including predictions of coupling efficiency and synergies (or anti-synergies) with the ICRH antennas, and power handling capacity, including models that can be used to describe the plasma buildup in the scrape-off layer via localized gas puffing and interaction with the LHRF waves;
>
> Propagation of LH waves in the scrape-off layer, through H-mode barriers, and toward the core plasma until they are absorbed by Landau damping ;
>
> Interactions of the LH waves with the electron distribution function and validation of FP and QL models (including effects of trapping) that lead to accurate calculation of current drive efficiency and current density profile.

Relatively long-distance coupling is required for application of LHCD on ITER. Although the separatrix-grill spacing in C-Mod is limited to ~5 cm, the perpendicular wavelength is typically ~ 5mm. Therefore measured in terms of λ_{perp}, long-distance coupling means about 10 wavelengths from the grill to the separatrix. We remark that the use of the commercial software package COMSOL has, in the past few months, led to a breakthrough in LH wave modeling that seamlessly models the microwave properties of the entire launching structure, the conversion of waveguide modes in the grill to plasma waves, and the propagation of LH waves into the core plasma. It remains to couple this code to a FP code to self-consistently calculate the interaction of the L waves with the electron distribution function, which is a high priority for the ongoing modeling effort.

An additional issue of great interest for further study is whether there is a density limit for LHCD, beyond effects of reduced accessibility and/or the usual $1/n$ fall off in current drive efficiency. An example of a density dependent limiting process is the onset of parametric decay instabilities, which could cause the launched LH wave (or pump) to lose power as the result of decay into daughter waves or quasimodes. PDI decays into LH and IC waves have already been observed in our experiments, however it is not clear if significant power from the pump is being lost by this mechanism. Present indications are that PDI's are not limiting the efficiency at high density, however we will continue to examine their effect.

Simulations based on ray-tracing code GENRAY and the FP code CQL3D have been invaluable to the rapid progress that has been made. Especially useful in comparing experimental results with simulations are the synthetic diagnostics imbedded in the simulations, principally X-Ray profiles and spectra, non-thermal ECE spectra and profiles of the magnetic field pitch angle on the outer mid-plane through which $j_{\varphi}(r)$ can be reconstructed. A new diagnostic which would assist in this effort and which is being considered is a vertically viewing ECE spectrometer, which could unambiguously resolve the spatial location of relativistically downshifted non-thermal ECE emission. An additional new diagnostic which is being installed for the FY09 campaign is a reflectometer imbedded in LH grill which should enable accurate characterization of the density profile from the grill to the separatrix and the effect on the profile of both ICRF and LHRF power.

Many of the issues needing resolution would benefit by measurement of the LH wave fields, which can be inferred from the associated density perturbations. For this reason, we are considering installation of a CO2 laser scattering system, similar to that which was successfully deployed on Alcator C. If feasible, this would likely be installed for the FY11 campaign. We also plan to expand use of the multi-frequency reflectometer system now being used to study lower frequency fluctuations, specifically the QC mode, in a modified form to map out the spatial dependence of the 4.6 GHz LH wave fields near the grill. A proof-of-principle test was successfully carried out in the FY08 campaign and it will continue to be developed in the FY09-FY10 campaigns.

7.2.3 Lower Hybrid RF References

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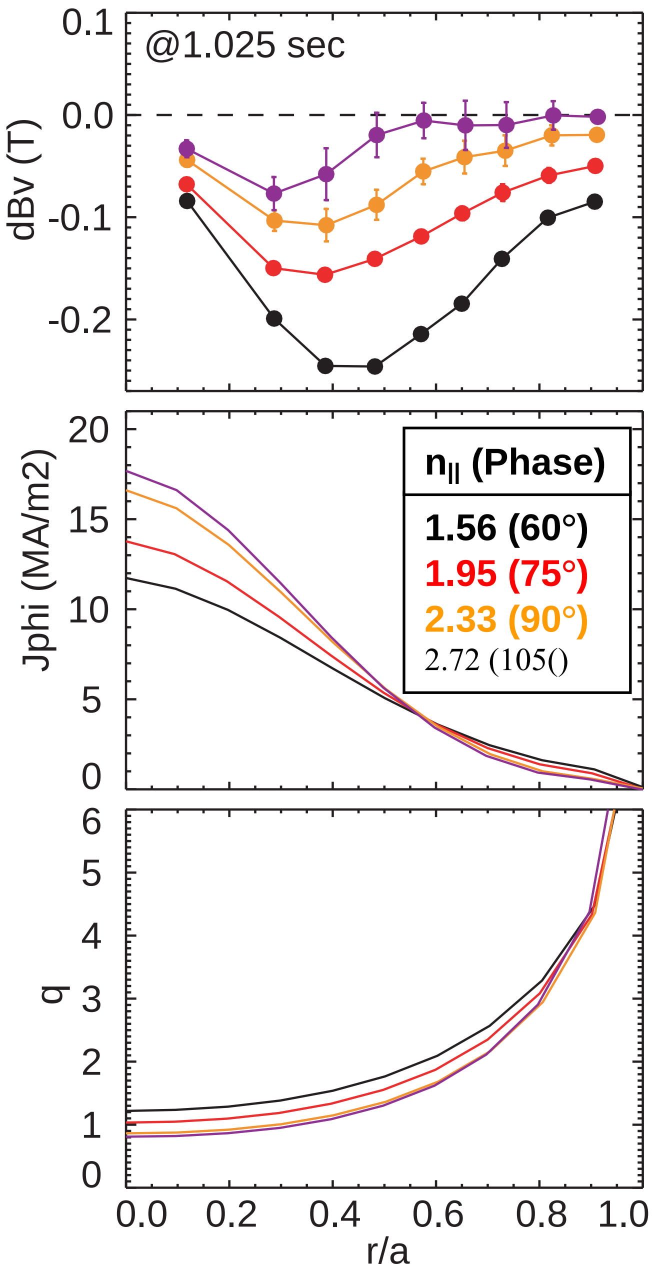

Figure 7.3 Radial profile of change in mid-plane poloidal field due to LHCD for four values of $n_{||}$ (Top). Corresponding mid-plane current density (Middle) and q-profiles (Bottom).

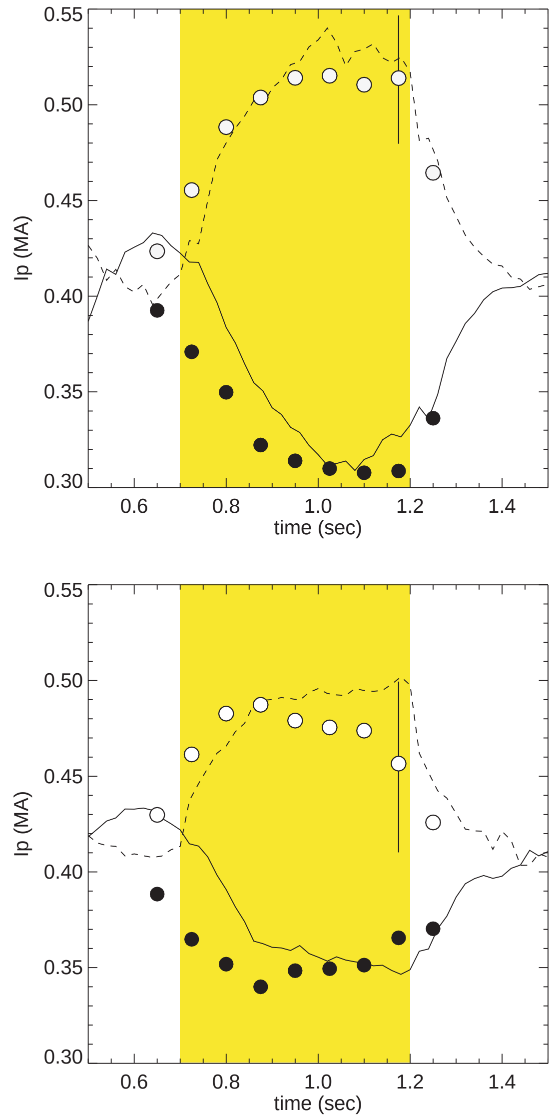

Figure 7.4 Fraction of current inside (closed circles) and outside (open circles) r/a = 0.44. In the upper panel $n_{||}$ = 1.56, while in the lower one $n_{||}$ = 1.95.

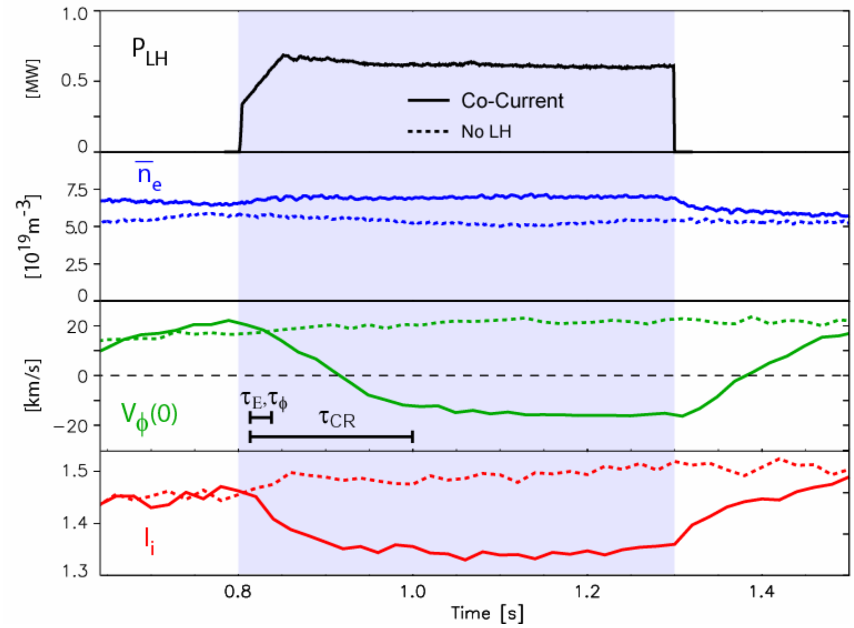

Figure 7.5 Onset of counter rotation after application of LHCD.

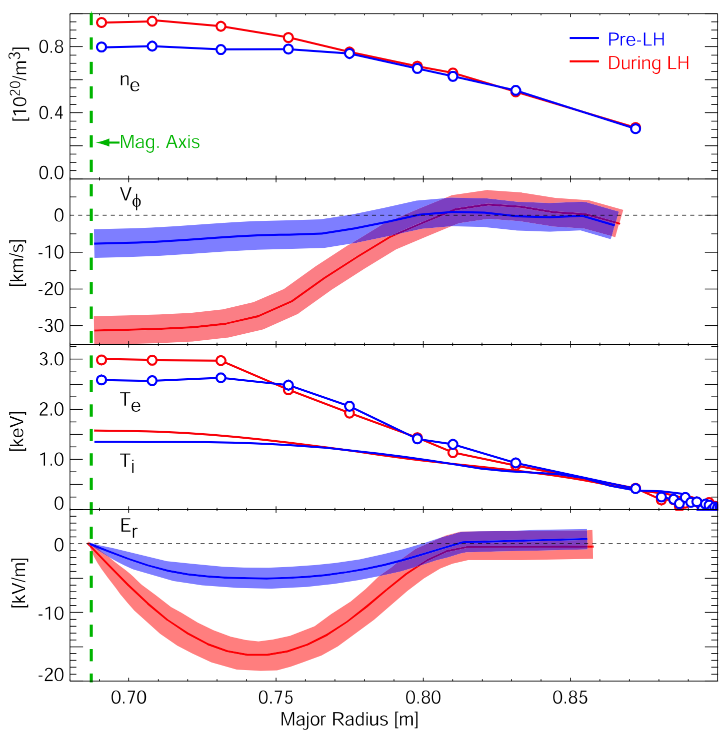

Figure 7.6 Profiles of density, toroidal rotation, temperatures and inferred radial electric field before and during application of LH power.

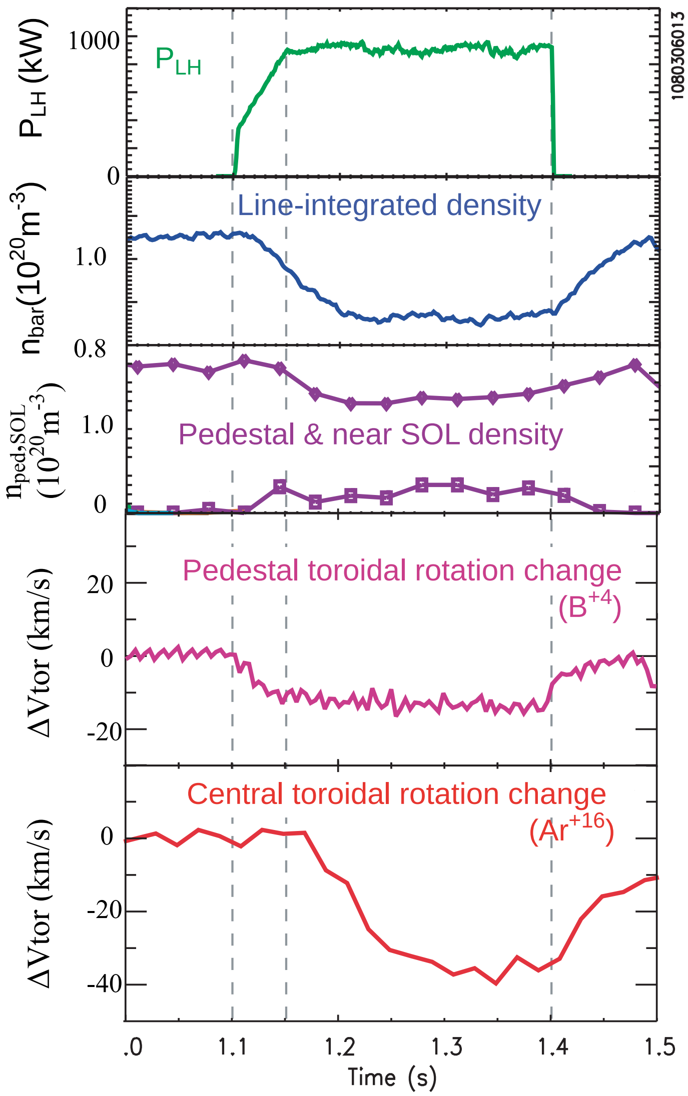

Figure 7.7 Change in plasma parameters in H-Mode during LHCD.

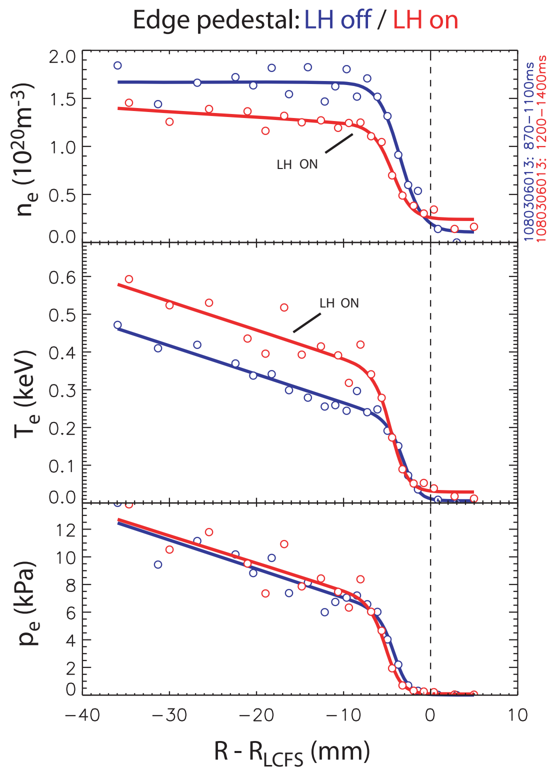

Figure 7.8 Effect of LHCD on pedestal density, temperature and pressure for the discharge in Fig 7.6.

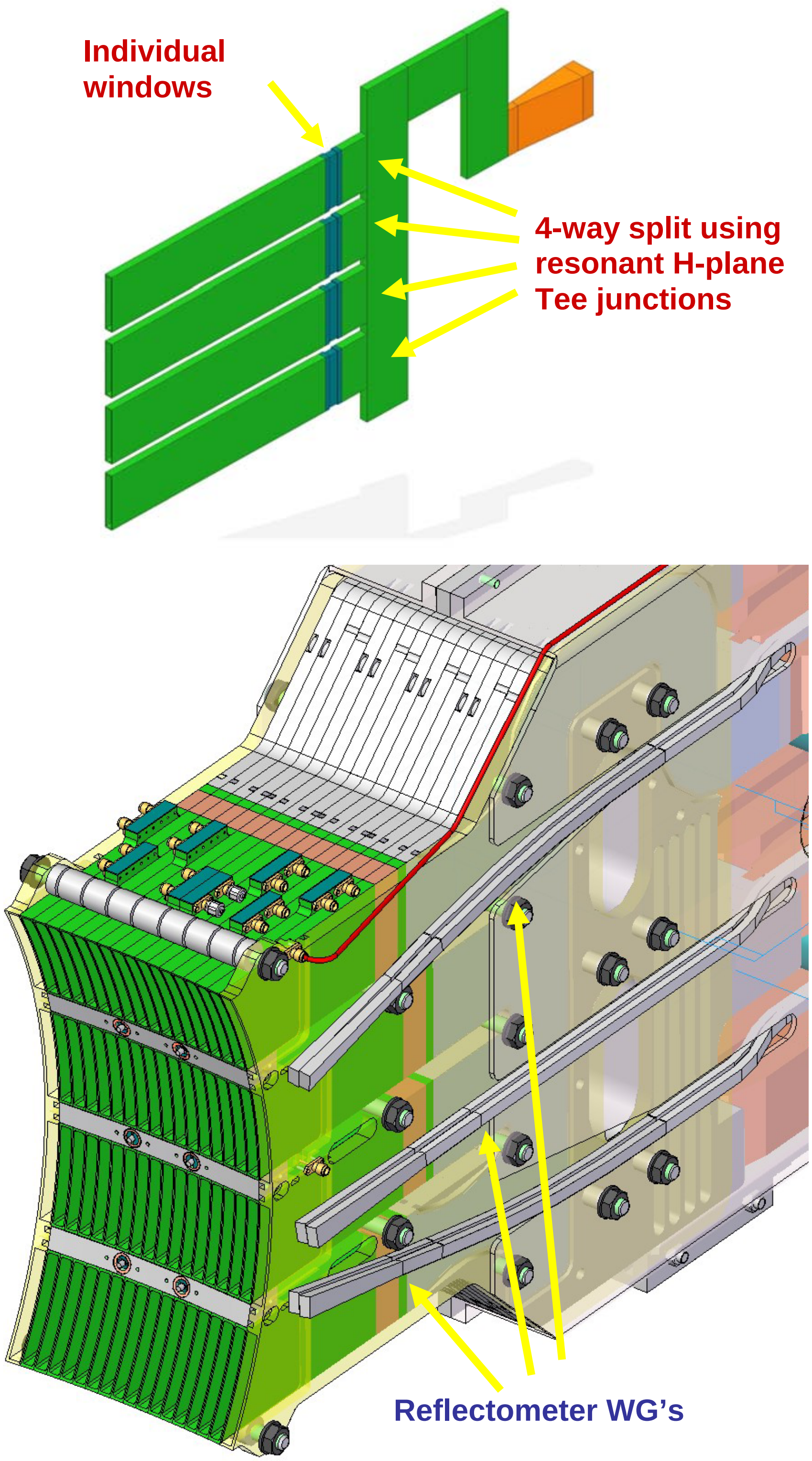

Figure 7.9 Top: 4-way power resonant divider used in the fabrication of the new LH launcher. Bottom: Isometric view of the assembled launcher, showing location of reflectometer waveguides, which will be used to map out SOL and pedestal density profiles.

8. 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.

Disruptions and disruption mitigation

Disruptions are one of the most urgent ITER physics issues. Over the past few years there has been extensive research on Alcator C-Mod to study disruption mitigation using gas jet injection of noble gases. These experiments have shown notable success at reducing the thermal loads on the divertor and the $J \times B$ electromagnetic loads due to halo currents. Our efforts have now begun concentrating on the disruption runaway electron (RE) issue, which is predicted to be much more of a problem in ITER than in present machines because of the avalanche process's exponential scaling with plasma current. Currently the ITER approach is to quench the avalanche process by massive gas injection to reach the Rosenbluth density ($\sim 10^{22}$ m^{-3}), but this has serious implications for the ITER cryopumps and tritium handling plant, particularly with mixed noble gases. Some present experiments strongly suggest that other mechanisms may suppress avalanching by enhancing RE transport losses with stochastic magnetic fields. Alcator C-Mod has a number of key tools which make it ideal for studying RE physics, including LHCD to generate a large seed population of suprathermal electrons, spatially-resolved HXR, diagnostics for imaging relativistic emission, a gas jet system to trigger disruptions, and error field coils to produce perturbed field lines.

Initial RE experiments during the most recent campaign have shown that suprathermal electrons do not survive past the thermal quench in normal C-Mod discharges, in contrast to observations in a number of other tokamaks, all of which are either circular cross-section, or limited, or both. This suggests that elongation and/or vertical stability might have something to do with observations of runaways during the current quench (CQ). This hypothesis has significant implications for REs on ITER, since it also will not be circular or limited. In the 2009-11 time frame we plan to test this both experimentally and theoretically, and this research is now a high-priority ITPA topic. Comparisons of runaway behavior in both elongated and near-circular (and possibly limited) C-Mod discharges are planned, using all of the RE relevant tools at our disposal. Extension of the NIMROD modeling that has been done in support of the gas jet mitigation research to model the effects of elongation of C-Mod equilibria on formation of stochastic regions and the effect on runaway electron confinement has begun as well.

In addition to the runaway electron issue, recent work has also concentrated on studying the toroidal variation of radiated power with gas jet mitigation. An understanding of this asymmetry is necessary in order to specify the number of gas jet valves needed on ITER.

To date, only a single C-Mod disruption has been analyzed in detail, and measurements using a number of solid-state diode (AXUV) arrays at various locations around the torus show that prior to the thermal quench (TQ), the radiated power is concentrated near the gas jet outlet, but it becomes much more symmetric both toroidally and poloidally immediately after the TQ and throughout the CQ. Over the next two years, the diagnostic set will be expanded to include fast visible spectroscopy, Lyman-α detectors, and more AXUV coverage. Data from many more disruptions will be analyzed to determine the variability and statistics of radiation asymmetry, including the dependence on different gas mixtures.

Further development of real-time disruption prediction, and possibly automated mitigation and/or avoidance activation, will continue. The ultimate goal is to be able to recognize and act on an impending disruption due to a number of different causes, such as locked modes, impurity injections, β-limits, density limits, VDEs, etcetera. This may be a significant challenge given the characteristically short response times necessary to mitigate C-Mod disruptions. A redesign of the gas jet delivery system may also be necessary to shorten the gas delivery time by moving the fast actuating valve closer to the plasma. The real-time prediction could possibly make use of recent work in C-Mod on advanced controllers and Kalman filter techniques.

Axisymmetric stability of ITER-like equilibria

ITER needs to be confident of axisymmetric control when operating close to machine limits. ITER also has stringent limits on measurement noise and fluctuations in the feedback loop. During the last campaign, C-Mod participated in a very successful joint ITPA experiment (maximum controllable displacement) to benchmark the vertical stabilization capabilities of current machines, and assess the implications for ITER. Based on the results of this study, it was concluded that the ITER vertical stability system needed to be enhanced, and as a result, internal position control coils have been added to the ITER design (along with internal ELM and RWM control coils). Some additional work on quantifying vertical stability metrics will continue on C-Mod, but most of the work in this area that is planned for FY2009-11 will concentrate on the design and experimental validation of high order controllers to improve the controllability of high-κ, high-ℓ_i plasmas. As mentioned previously this work may also be useful in developing robust real-time disruption predictors.

Our research on stability control during this time frame will also better characterize the effects of noise on feedback, and will include development of noise rejection/suppression algorithms, possible emphasizing model-based Kalman filters. We will also further develop ‘safe scenarios’ and adaptive algorithms that could better accommodate power supply saturation or failure using adaptive pulse rescheduling that would automatically interpolate to a less demanding, but compatible target equilibrium.

Effects of non-axisymmetric fields

A series of JET identity experiments designed to test neoclassical toroidal viscosity (NTV) theory of non-resonant magnetic braking of rotation have been concluded. These gave a null result, implying that NTV theory is not fully consistent between the two machines. Any further work in this area will have to wait for a proposed upgrade to the A-coil power supplies and coilset, which depends on actual budgets received in the FY10-11 period.

During our most recent campaign, the A-coil system was used to explore the application of resonant magnetic perturbations for affecting/controlling edge pedestals and ELMs. Comparison of discharges with the A-coils at zero current and at maximum available current failed to show observable effects on pedestal profiles or plasma rotation. At maximum A-coil current, the Chirikov parameter, σ_{CH} ~ 1.0 at q_{95}=3.0, i.e. just marginal with respect to island overlap and formation of a stochastic edge. At lower q_{95} and lower ν^* there was a hint of an effect on ELMs, possibly related to a decrease in the QC mode. The proposed upgrade to the A-coil power supplies and coilset would improve the chanced of affecting ELMs.

TAE modes, Alfvén cascades, and fast particles

Past work in this area has concentrated on characterizing the dependence of intermediate-*n* TAE damping rates on plasma parameters, particularly elongation, and on benchmarking codes (NOVA-K, AORSA/CQL3D) against experiment. Recent emphasis has been on studying the TAE-induced loss of the fast ICRF-generated minority ions. A somewhat unexpected finding is that TAE stability is seen to increase with moderate ICRF power in some shots. This implies that the fast ions driven by the ICRF must be increasing the TAE damping rate. New modeling with NOVA-K shows qualitative agreement, i.e. fast particles with energies below the TAE-particle resonance contribute to stabilizing the TAE.

A large research effort in the next few years will continue on reversed-shear Alfvén eigenmodes, both during I_p rampup and also during sawtooth cycles. The observed frequency up/down chirping and other details of the frequency evolution provides precise knowledge of q_{min} (~1%), and detailed information on the *q*-profile near the plasma center, and its evolution in time. Proper interpretation of the results relies on comparison with PCI measurements, in conjunction with NOVA-K modeling. This research may quite possibly result in a very sensitive method for measuring the effects of LHCD on the current profile in C-Mod.

Several diagnostic upgrades for TAE/FP research are being carried out in the near-term, and several new diagnostics are planned in the FY09-11 time frame, dependent on budgets. One of the current active MHD antennas is being repositioned for the next run campaign to improve selectibility of toroidal mode numbers. The compact neutral particle analyzer (CNPA), which provides information on the ICRF-generated fast ions that neutralize by charge-exchange and escape from the plasma, is being expanded to become a spatially-resolving array. Future proposed upgrades include a fast ion H_α diagnostic to look at confined ICRF ions, as well as a fast ion loss diagnostic.

Appendix A

Alcator C-Mod Publications –2008 to present

Papers Published in Refereed Journals:

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Conferences:

IAEA International Workshop "Challenges in Plasma Spectroscopy for Future Fusion Research Machines", Jaipur, India, Feb. 2008

Talks
Bitter, M., Hill, K.W., Scott, S., Ince-Cushman, A., Reinke, M., Rice, J.E., Beiersdorfer, P., Gu, M.F., Lee, S.G., Broennimann, Ch., Eikenberry, E.F., "A novel x-ray imaging crystal spectrometer for Doppler measurements of ion temperature and plasma rotation velocity profiles"

International Sherwood Fusion Theory Conference, Boulder, CO, March 2008

Talks
Bonoli, P.T., “Full-Wave Electromagnetic Field Simulations of Lower Hybrid Wave Propagation in ITER Relevant Regimes”

21st US Transport Task Force Workshop, Boulder, CO, USA, March 2008

Talks
LaBombard, B., “Scrape-off Layer Flows, Toroidal Rotation and Critical Gradient Phenomena in the Tokamak Edge

Snipes, J., “Fast Electron Driven Alfvén Eignemodes in the Current Rise in Alcator C- Mod”

Porkolab, M., “ Experimental Study of Reversed Shear Alfvén Eignenmodes During ICRF Minority Heating and Relationship to Sawtooth Crash Phenomena in Alcator C-mod”

Ince-Cushman, A., “Toroidal Rotation Profile Modification by Lower Hybrid Waves”

McDermott, R., “Radial Electric Field Structure on Alcator C-Mod”

Hughes, J.W., “Modifications to H-Mode Pedestal Structure via Particle Control and Topology Variation on Alcator C-Mod”

Cziegler, I., “Structure and Velocity Shear in the Broadband Edge Turbulence in Alcator C-Mod”

Posters
Smick, N., “Parallel and Perpendicular Plasma Flows in the Edge of Alcator C-Mod”

Lin, L., “Turbulence Studies with the Phase Contrast Imaging in Alcator C-Mod”

Marr, K., “Comparison of First-Order Neoclassical Flow Theory with CXRS Measurements from Alcator C-Mod”

18th International Conference on Plasma Surface Interactions, Toledo, Spain, May 2008

Invited Talks
Lipschultz, B., “Hydrogenic retention in a high-Z plasma facing component tokamak - Alcator C-Mod”

Posters
Harrison, S., “ARRIBA: A novel in-situ plasma surface interaction (PSI) diagnostic for magnetic fusion devices”

Hughes, J.W., “H-mode edge profile modification and density control with pumping and magnetic balance regulation in Alcator C-Mod”

5th International Conference on the Physics of Dusty Plasmas, Ponta Delgada, Azores, May 2008

Talks
Hutchinson, I.H., "Direct Calculation of Plasma-Dust Interactions", invited plenary at 5th International Conference on the Physics of Dusty Plasmas, Ponta Delgada, Azores

17th Topical Conference on High-Temperature Plasma Diagnostics, Albuquerque, New Mexico, May 2008

Talks
Rowan, W.L., Bespamyatnov, I.O., Granetz, R.S., "Wide-View CXRS Diagnostic for Alcator C-Mod"

Poster
Ko, J., Scott, S., Bitter, M., Lerner, S., Shiraiwa, S., "Design of a new optical system for Alcator C-Mod MSE diagnostic"

**35th European Physical Society Conference on Plasma Physics.
Crete, Greece, June 2008**

Talks
Wright, J.C., Bonoli, P.T., Valeo, E., Phillips, C.K., Brambilla, M., Bilato, R., "The importance of the effects of diffraction and focusing on current deposition of lower hybrid waves"

Posters
Hughes, J.W., "H-mode optimization using magnetic topology variation in Alcator C-Mod"

Snipes, J.A., "Ip and Bt Dependence of the H-mode Threshold Low Density Limit on Alcator C-Mod"

IAEA Fusion Conference, Geneva, Switzerland, Oct. 2008

Talks
Wright, J.C., "TER Relevant Simulations of Lower Hybrid and Ion Cyclotron Waves with Self-Consistent Non-Maxwellian Species"

Posters
Snipes, J.A., "Characterization of Stable and Unstable Alfven Eigenmodes in Alcator C-Mod"

50th Annual Meeting of the APS Division of Plasma Physics, Dallas, TX, USA, Nov. 2008

Invited Talks
Edlund, E., "Observation of Reversed Shear Alfvén Eigenmodes During the Sawtooth Cycle in Alcator C-Mod"

Ernst. D., "Role of Zonal Flows in TEM Turbulence through Nonlinear Gyrokinetic Particle and Continuum Simulation"

Lin, Y., "Observation of ICRF Mode Conversion Plasma Flow Drive on Alcator C-Mod"

Lipschultz , B., "Characteristics of Hydrogenic Retention in a High-Z First-Wall Tokamak"

McDermott, R., "Edge Radial Electric Field Structure and Connection to H-Mode Confinement in Alcator C-Mod Plasmas"

Rice, J., "Intrinsic and RF Driven Rotation in Alcator C-Mod Plasmas"

Wright, J., "An Assessment of Full-Wave Effects on the Propogation and Absorption of Lower Hybrid Waves"

Contributed Orals
Cziegler, I., "Rapid Changes of Turbulence Propagation Direction in the Edge of Alcator C-Mod"

Harrison, S., "ARRIBA: A Novel In-Situ Plasma Surface Interaction (PSI) Diagnostic for Magnetic Fusion Devices"

Hubbard, A., "Effects of Lower Hybrid Current Drive on H-Mode transition, Pedestal and Confinement on Alcator C-Mod"

Ko, J., "MSE Measurement of Current Density Profile Modification in LHCD Experiments in Alcator C-Mod"

LaBombard, B., "Relationship between Edge Gradients and Plasma Flows in Alcator C-Mod"

Lin, L., "Studies of Turbulence and Transport in Alcator C-Mod Ohmic Plasmas with Phase Contrast Imaging and Comparisons with GYRO"

Marmar, E., "Alcator C-Mod Research Highlights J. Rice Counter-Current Rotation in Alcator C-Mod LHCD Plasmas"

Reinke, M., "Flux Surface Assymetries in VUV/SXR Emission on Alcator C-Mod"

Sips, G., "ITER Relevant Current Ramping Experiments in Alcator C-Mod"

Schmidt, A., "Evidence of Spectral Control over Lower Hybrid Power Depostition on Alcator C-Mod"

Terry, J., "Observations of Edge Turbulence in Midplane- and Xpt-regions of Alcator C-Mod"

Wilson, R., "Overview of LHCD Experiments on Alcator C-Mod"

Posters
Bader, A., "Measurements of Fast Ion Distribution in ICRF Heated Plasmas"

Bonoli, P., "Overview of Physics Research in the SciDAC Center for Simulation of Wave - Plasma Interactions"

Dominguez, A., "Study of Lower Hybrid Wave Propagation in Alcator C-Mod Using Reflectometry"
Ferrara, M., "Poloidal Current Anti-Saturation Control Routine with DPCS Multi-processor Architecture"

Fiore, C., "Transport Studies in Alcator C-Mod ITB Plasmas"

Graf, A., "Main Ion Plasma Flow in the Edge of the Alcator C-Mod Tokamak"

Granetz, R., "Studying Disruption Runaways on Alcator C-Mod, and why it may be good news for ITER"

Howard, N., "Design and Construction of a Multi-Pulse Laser Blow-Off System for Impurity Transport Studies on Alcator C-Mod"

Hughes, J., "Pedestal and Confinement Properties on Alcator C-Mod with Varied Shape and Magnetic Topology and with Application of Lower Hybrid Power"

Hutchinson, I., "Magnetized Ion Collection by Oblique Surfaces Including Self-Consistent Drifts: Mach-Probes of Arbitrary Shape"

Kessel, C., "Progress Using LHCD and ICRF on Advanced Tokamak Discharges in Alcator C-Mod"

Lau, C., "SOL Reflectometer for Alcator C-Mod"

Ma, Y., "Improved Absolute Calibration of Thomson Scattering Diagnostics on the Alcator C-Mod Tokamak"

Marr, K., "Comparison of Neoclassical Flow Theory with CXRS Velocity Measurements from Alcator C-Mod"

Meneghini, O., "TOPLHA and ALOHA: Comparison between Lower Hybrid Wave Coupling Codes"

Ochoukov, R., "Study and Optimization of Boronization in Alcator C-Mod"

Parker, R., "Progress in Lower Hybrid Current Drive Experiments on Alcator C-Mod"

Porkolab, M., "Studies of Turbulence and Transport in Alcator C-Mod H-Mode Plasmas with Phase Contrast Imaging and Comparisons with GYRO"

Scott, S., "Direct Comparison of GEMR Edge Turbulence Simulations with Alcator C-Mod SOL Turbulence Measurements"

Sears, J., "Influence of ICRF Heating on the Stability of TAEs"

Shiraiwa, S., "Development of an Operational Scenario for a Current Hole Plasma Using TSC on Alcator C-Mod"

Smick, N., "Parallel and Perpendicular Plasma Flows in the Edge of Alcator C-Mod"

Tsujii, N., "Measurements of Mode Converted ICRF Waves with Phase Contrast Imaging in Alcator C-Mod"

Wallace, G., "Interaction of Lower Hybrid Waves with the Scrape Off Layer"

Wukitch, S., "Overview of ICRF Experiments in Alcator C-Mod"

Xu, P., "Development of C-Mod FIR Polarimeter"

Atomic Processes in Plasmas conference in Monterey, CA, March 2009

Posters
Podpaly, Y., "High-Resolution Spectroscopy Of 2s1/2 - 2p3/2 Transitions In W71+ Through W65+"

Other Invited Talks

Rowan, W.L., "Light Impurity Transport at an Internal Transport Barrier in Alcator C-Mod", presented to the University of Texas, Plasma Physics Seminar, February 29, 2008

C-Mod Related Science Talks, Presented to Audiences Primarily Composed of Fusion Science Researchers

Lin, Y., "ICRF physics and application on tokamaks", Lecture at Chinese Plasma Physics Winter School, Hefei, China, January 2008

MIT IAP Talks

Hubbard, A., "Advanced' Scenario Research on Alcator C-Mod: Towards a better tokamak reactor", Jan. 2008

Granetz, R., "Fusion Energy on Albany Street: ITER-relevant research on Alcator C-Mod", Jan. 2009

Awards and Prizes

Paper nominated for the 2008 Nuclear Fusion Award - one of the 10 best papers published in 2005: LaBombard B, Evidence for electromagnetic fluid drift turbulence controlling the edge plasma state in the Alcator C-Mod tokamak (11), pp. 1658-1675, Massachusetts Institute of Technology, Plasma Science and Fusion Center

Appendix B. Summary National Budgets, Run Time and Staffing

		FY09	FY10A	FY10B	FY11A	FY11B	FY11D
			Request	Increment	Guidance	Increment	Decrement
Funding ($ Thousands)							
Research		6,700	6,700	7,400	6,850	7,650	6,180
Facility Operations		13,634	14,234	15,655	14,615	16,000	13,230
Capital Equipment		400	0	0	0	0	0
PPPL Collaborations		3,372	3,372	3,709	3,456	3,802	2,910
UTx Collaborations		340	340	440	350	450	320
LANL Collaborations		100	100	110	100	110	90
MDSplus		155	155	155	155	170	140
International Activities		80	80	80	80	80	35
Total (inc. International)		24,781	24,981	27,549	25,606	28,262	22,905
Staff Levels (FTEs)							
Scientists & Engineers		52.1	53.0	57.4	53.8	57.5	50.2
Technicians		28.5	30.3	34.3	30.3	34.3	28.3
Admin/Support/Clerical/OH		18.0	18.5	20.0	18.0	20.0	16.2
Professors		0.3	.3	.3	.3	.3	.3
Postdocs		2.0	2.0	3.0	2.0	3.0	2.0
Graduate Students		32.3	28.9	31.1	28.9	28.9	25.9
Industrial Subcontractors		2.0	2.0	2.2	2.0	2.2	1.5
Total		135.2	135.0	148.3	135.3	146.2	124.4
	FY08 Actual	FY09	FY10A	FY11D	FY11A	FY11B	
			Request	Reduced	Guidance	Incremental	
Facility Run Schedule							
Research Run Weeks	15.7	10	13	6	13	18	
Users (Annual)							
Host	41	39	40	38	40	42	
Non-host (US)	65	65	64	62	64	66	
Non-host (foreign)	50	48	50	45	50	52	
Graduate students	32	29	28	26	28	28	
Undergraduate students	4	5	5	5	5	7	
Total Users	192	186	185	174	185	194	
Operations Staff (Annual)							
Host	70	67	68	65	68	70	
Non-host	4	4	4	3	4	5	
Total	74	71	72	68	72	75	

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