ALCATOR C-MOD
FY2010-FY2012 WORK PROPOSAL

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Plasma Science and Fusion Center
Massachusetts Institute of Technology
Cambridge, MA 02139

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FY10-12 WORK PROPOSAL

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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 C-Mod 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 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.

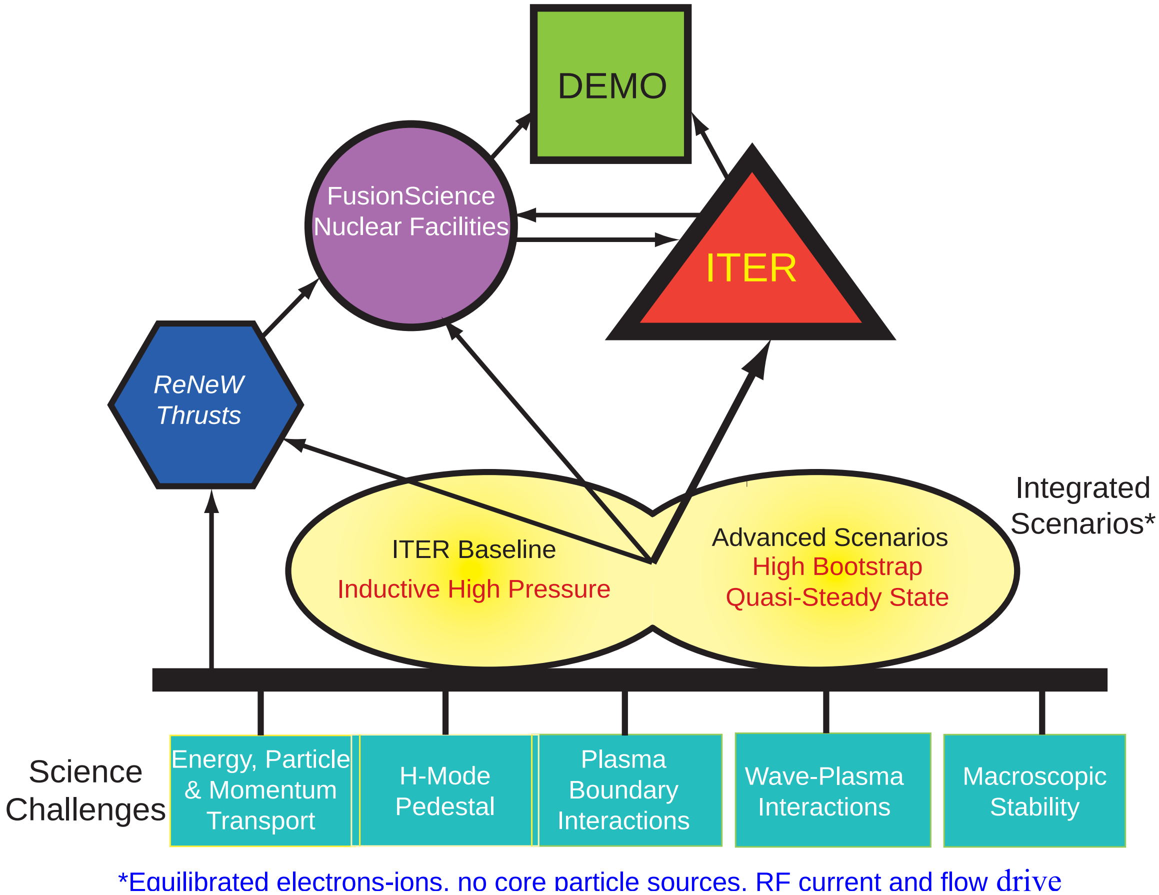

Figure 1.1 C-Mod Research is organized around Integrated Scenarios supported by topical science areas. These in turn make key contributions to plans for ITER operations, and are strongly aligned with the campaigns identified by the ReNeW Thrusts.

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 only reactor-relevant 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 or even exceeding 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, are contributing 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.

Educating the next generation of fusion plasma scientists 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 investigates many of the key outstanding issues that need resolution to support successful operation of ITER. Research is well advanced on many 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 include the following.

Integrated Scenarios, Baseline H-mode and Advanced Scenarios:

- Compatibility of core and boundary, extending beyond the last closed flux surface to open field lines, including baseline and advanced scenarios, with radiative divertor solutions
- 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
- Reference set of ITER scenarios for baseline H-mode, steady-state and hybrid operation, for databases and modeling
- Profile control methods
    - j(r) with combined LHCD and bootstrap, including sawtooth suppression, shear control
    - rotation/flow control with both ICRF and LHRF
- Advanced I-mode scenarios with non-inductive current drive

Transport

- Momentum transport/intrinsic rotation: Study and characterize rotation sources, transport mechanisms and effects on confinement and barrier formation (ITER rotation profile)
- Understand the collisionality dependence of density and impurity transport (ITER density profile)
- Quantitative tests of fundamental features of turbulent transport theory via comparisons to measurements of turbulence characteristics, code-to-code comparisons and comparisons to transport scalings.
- Test understanding of improved core transport regimes with reactor relevant conditions, specifically electron heating, T_e~T_i and low momentum input, and provide extrapolation methodology
- 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 β_N>1.3; effects of shaping.
- ELM control techniques: stochastic fields with external coils.
- 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
- Interrelation of pedestal structure and edge relaxation mechanisms in ELMy and ELM-suppressed regimes
- Investigation of the trigger mechanisms for H-mode in different configurations and the role played by local edge conditions in determining H-mode power threshold
- Investigation of mechanisms that give rise to the drastically different levels of particle and energy transport in the I-mode

Plasma-Boundary Interactions

- Tritium retention and tritium removal: solid high Z PFCs; disruption cleaning; plasma and nuclear damage; erosion; affects of active pumping
- Parallel and perpendicular SOL energy transport and resulting divertor heat-flux footprints for extrapolation to ITER
- Scaling present-day conditioning and operational techniques to future devices: boronization with high Z walls; ICRF induced impurity generation.
- Active between-discharge interrogation of first-wall conditions using RFQ accelerator and diagnostics

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.
- Toroidal variation of radiation during disruption mitigation
- Effects of plasma shape, especially elongation, on disruption runaway dynamics
- Develop reliable disruption prediction methods; real-time automatic mitigation using Digital Plasma Control System.
- Fast-particle physics: interactions between ICRF and Alfven Eigenmodes (AE's), including modulation of the ICRF heating in the AE frequency range
- 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.

Relationship of C-Mod Investigations to the Priorities Identified by ReNeW

The community-wide Research Needs process (ReNeW) has identified a broad series of thrusts and campaigns on the path to fusion energy.[1]
C-Mod is actively contributing to a large number of these, including:

- Thrust 2: Control transient events
  - Disruption avoidance and mitigation
  - Edge plasma transport and stability, emphasizing ELM-free regimes
- Thrust 4: Qualify operational scenarios and supporting physics basis for ITER
  - Startup, flattop, rampdown
  - Current and flow drive (LHCD, MC Flow Drive)

[1] http://burningplasma.org/web/ReNeW/ReNeW.report.web2.pdf

- Pedestal and ELMs
- ICRF/edge interactions
- Hybrid scenarios
- Transport and H-mode threshold physics

- Thrust 5: Expand limits for controlling and sustaining fusion plasmas
  - Develop and test control methods (notably ICRF, LHCD, transport), in DEMO-relevant conditions for T, j, n
  - Disruption avoidance in steady-state high performance scenarios
  - Regulate/control heat flow in steady-state high performance scenarios
- Thrust 6: Develop predictive models
  - Strong connection to theory/modeling
  - Essential contributions of data for model validation
  - Notable strengths in transport, RF, boundary topical areas
- Thrust 8: Dominantly self-heated and self-sustained burning plasmas
  - LH and Advanced scenarios programs important for assessing potential ITER scenario extensions
    - Notably through LHCD upgrades
- Thrust 9: Unfold the physics of boundary layer plasmas
  - World-leading edge physics program
    - state-of-the-art diagnostics
    - unique high power- and particle-density regimes
- Thrust 10: Science and technology of plasma-surface interactions
  - Refractory metal PFC's
  - DEMO-relevant divertor upgrade
- Thrust 12: Demonstrate integrated solution for plasma-material interfaces compatible with optimized core
  - High power density, elevated wall temperature
  - Combining with sustained non-inductive regimes should provide the basis for designing a dedicated PMI facility (e.g. LHCD issues)
- Thrust 1: Diagnostic development
  - ITER relevant field, density, geometry (e.g. polarimetry)
- Thrust 3: Understand the role of alpha particles in burning plasma
  - ICRF tails drive Alfven eigenmodes; active MHD/ICRF drive of stable modes
- Thrust 10: Improve power handling through engineering innovation
  - Temperature controlled tungsten PFCs
- Thrust 13: Science and technology for fusion power extraction and tritium sustainability
  - Provide key data for design of a Fusion Science Nuclear Facility (e.g. LHCD in high density H-modes on FDF)

– Hydrogenic isotope retention in first wall

Budget and Schedule

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, DE-FC02-99ER54512. 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 carrying out those plans, subject to budget constraints.

The baseline budget for the C-Mod project in FY2010 is based on appropriations from the Office of Fusion Energy Sciences, with total national baseline project funding of $26.46M, including $22.61M at MIT, and major collaborations totaling $3.85M. These budgets will accommodate 13 weeks of research operations in FY2010. Because of the addition of ARRA incremental funds in September, 2009, (935k for increased facility operations, and $4.96M for facility and diagnostics, we have increased the target for research operation in FY10 by 5 weeks, giving a target total of 18 weeks (±10%). For FY11A, our plans assume the February 2010 guidance from OFES based on the Administration's submission to congress. This results in a plan for 15 research weeks in FY2011. The FY11B budget assumes a 10% increment above the FY11A guidance, and results in a plan for 20 research weeks in FY11. For FY12A (guidance), we assume flat from FY11A, with a 2% increment for inflation, resulting in a plan which accommodates 15 weeks of research operation. FY12B assumes a 10% increment over FY11A, and results in a plan for 19 weeks of research operation. A 10% cut relative to FY11A leads to a plan for 9 research weeks in the FY2012D case. Finally, we have examined the requested "Ready/Standby" case, which does not accommodate any facility operation.

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,[2] 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 2011B and 2012B incremental cases, the 2011D decremental case, along with the requested "No Operations/Ready-Standby" case.

[2] www.psfc.mit.edu/research/alcator/program/technical_proposal_submitted_web_version-2.pdf

Table 1.1: Research operation for guidance (10, 11A, 12A), incremental (11B-12B) and decremental (12D) budget cases

Fiscal Year	**10**	**11A**	**12A**	**11B**	**12B**	**12D**
National Budget ($M)	26.54	27.58	28.13	30.35	30.35	24.82
Research Operation Weeks	13*	15	15	20	19	9
Research Operation Hours	416*	480	480	640	608	288

*With incremental ARRA funding, added 5 research weeks (160 hours).

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.

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.

Hybrid Advanced Scenario investigation [Original target: Sept 2009/revised: Sept 2011]
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.
Status: Because of delays in the installation of the new Lower Hybrid Launcher, it was not possible to complete the experiments for this goal, and they had to be delayed. In the current schedule, the launcher will be installed in April, 2010, and Lower Hybrid experiments will resume in June, 2010. In addition, our Program Advisory Committee has recommended that we increase the priority of investigations into the apparent density limits for LH Current Drive, which appear in the ITER-relevant density range at and above $1x10^{20}m^{-3}$*, and defer our investigations into Hybrid Advanced Scenarios. As a result, we now anticipate devoting run time to the Hybrid Scenario experiments primarily in FY2011, and the completion of the milestone is thus moved to September, 2011.*

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]
The OFES Joint Facility Research Target for FY2011 will involve an integrated experiment, theory, and computational effort on tokamak pedestal physics. The general

goal is to understand the physics mechanisms responsible for the structure of the pedestal and develop a predictive capability for burning plasma devices. 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.

Proposed C-Mod Research Goals for FY2012

Lower Hybrid wave physics and effects of plasma density [Sept 2012]
Lower Hybrid Current Drive (LHCD) is one of the leading candidates for non-inductive current drive in a steady state tokamak reactor. Previous and current LHCD experiments (including those on Alcator C-Mod) have observed a density limit above which the effectiveness of LHCD is significantly reduced. C-Mod will carry out experiments to explore possible density limit mitigation strategies and use modeling tools to interpret the experimental results and to help evaluate projections for LHCD on ITER.

Strongly confining the heat, but not the particles, in the Alcator C-Mod fusion plasma [Sept 1012]
It is highly desirable to understand, and separate, if possible, the mechanisms that lead to the losses of energy and plasma particles in magnetically confined fusion plasmas. In general it is desirable to have a strong barrier near the edge of the plasma to keep the heat from escaping, thus ensuring high enough plasma temperatures for fusion, while at the same time having the plasma particles, in particular fusion ash and impurities, quickly swept out to keep the plasma clean and at the optimal density for fusion performance. Additionally, the edge barrier should be stable to prevent short-lived, but potentially damaging bursts of energy to the surrounding materials. Alcator C-Mod will explore plasma configurations that exhibit these behaviors with the goal of better understanding the transport and stability of the edge barrier.

Goals Accomplished in FY2009

Achieve research operating time of 9 weeks (±10%) [September 09] (OFES Facility 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 FY09 campaign concluded on September 28, 2009, with 9.1 research weeks accomplished. Quarter by quarter run statistics can be found at
http://www.psfc.mit.edu/research/alcator/facility/Operations/FY09_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.

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.

Spontaneous, self-generated flow (in the absence of external momentum input), with velocities up to 130 km/s in the co-current direction, has been observed in C-Mod ICRF and Ohmically heated H-mode discharges. The rotation propagates in from the pedestal region following the L-H transition, on a time scale similar to the energy confinement time. The magnitude of the change in rotation between L- and H-mode scales as the change in the stored energy normalized to plasma current. With an eye toward comparison with the results from other devices, and for connection to theory, this scaling can be made in terms of dimensionless plasma parameters. For the C-Mod results, a plot of the ion thermal Mach number as a function of normalized pressure is shown in Fig. 1.2. Currently there is no comprehensive first-principles theory which can explain this spontaneous rotation. An equally good correlation can be found with the pedestal pressure gradient, a more fundamental local parameter which perhaps relates to the underlying mechanism, such as the residual stress. Spontaneous rotation has been observed on many other devices, under a large range of operating conditions and heating techniques, which underscores the fundamental nature of the phenomenon. An inter-machine scaling has been developed and the results are shown in Fig. 1.3. Extrapolation to ITER is very encouraging, and depending on the operational scenario, spontaneous rotation velocities of several 100 km/s are expected. In terms of Alfven Mach number, which is the relevant parameter for resistive wall mode suppression, values in the range of 2-4% are reasonable, which are more than sufficient for RWM stabilization without relying on neutral beam injection, which itself may not be available.

It would be desirable to have a solid theoretical understanding of spontaneous rotation in order to be confident of extrapolations from scaling laws. For a complete understanding of the process, spontaneous rotation in L-mode plasmas must also be addressed. In L-mode, the parameter dependence of spontaneous rotation is much more complicated, and abrupt rotation direction reversals are often observed. This process exhibits hysteresis which may provide clues to the underlying cause. An extensive database of rotation in L-mode plasmas is currently being constructed from C-Mod data, including the full velocity profiles.

Momentum transport is being studied on C-Mod, following the rotation profile evolution following transient events such as the L-H transition, fast configuration changes (the L-mode rotation velocity is very sensitive to magnetic configuration) and RF modulation which has been shown to ‘drive’ rotation (both ICRF mode conversion flow drive and lower hybrid current drive).

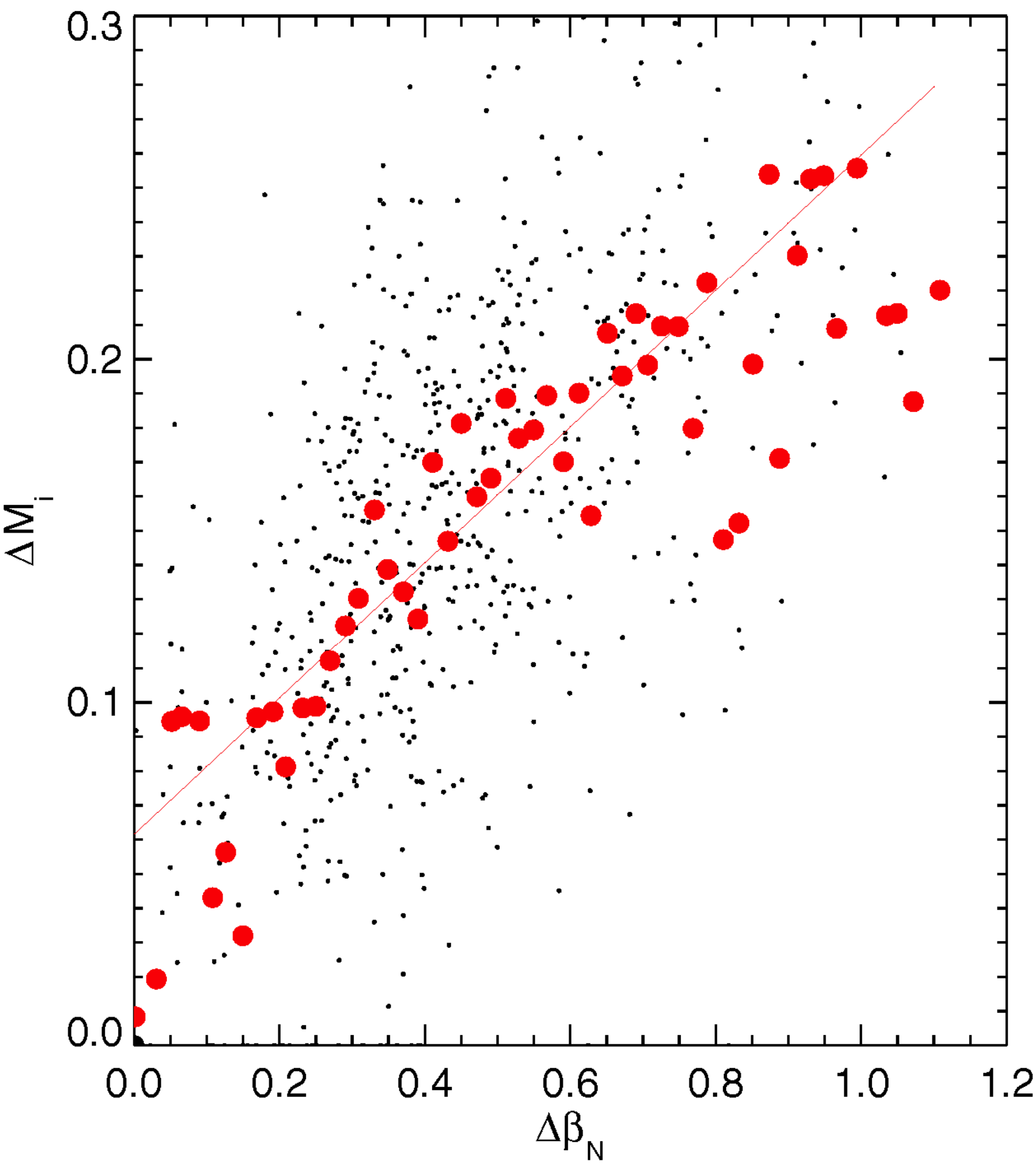

Fig. 1.2 Intrinsic rotation scaling data for C-Mod, showing the linear increase of ion Mach number with increase in normalized β. The small black dots show results from individual time slices, and the bigger red dots show the results of binning and averaging the data.

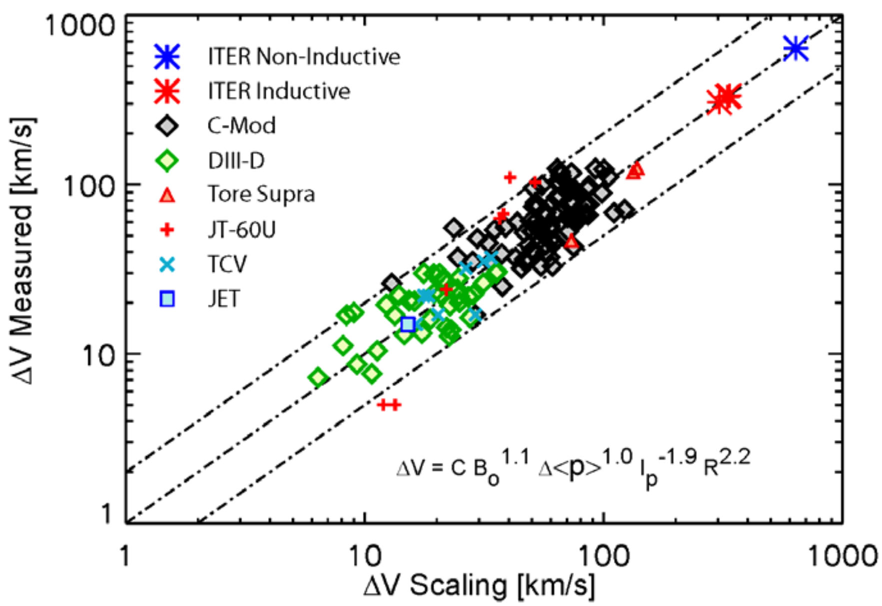

Fig. 1.3 Results from a multi-machine database for change in intrinsic toroidal rotation compared with the power scaling law developed from the data. Extrapolations to ITER using this scaling for the inductive (baseline) and non-inductive (advanced) regimes indicate large intrinsic rotation may result in ITER.

Recent scientific achievements and research plans for FY10-FY12 are detailed in subsequent sections of this Work Proposal, organized by scientific topics and integrated thrusts.

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

2.1. Recent Research Highlights

Development of predictive modeling is a major programmatic goal for the U.S. fusion program. A key element in our transport research is the validation of nonlinear gyrokinetic codes via comparisons between theory/simulation and our experiments. C-Mod researchers and collaborators are actively involved in the development of synthetic diagnostics which calculate the predicted signals from turbulence models [1]. Analysis is carried out by C-Mod staff and students, members of the PSFC theory division and with collaborators. Recent work has included the comparison of temperature profiles to nonlinear stability which discovered important contributions from collisionality [2]; particle transport in the core of ITBs including the discovery of a nonlinear upshift in the critical density gradient for TEM [1]; ITB thresholds and the role of ITG stability [3]; particle transport and density peaking at low collisionality and electron energy transport and turbulence in the linear low-density Ohmic regime [4]. This work directly supports ReNeW thrust 6. More details are included in the sections below.

2.1.1. Electron Transport

Electron transport is poorly understood, but will be important in future devices where $\tau_E > \tau_{ei}$ and neither transport channel is ignorable. Studies of electron transport in low density Ohmic plasmas revealed a discrepancy between experimental results and turbulence models [4]. In this regime without auxiliary heating, all power is coupled into the electrons and the electron-ion energy exchange is small resulting in an electron-channel dominated regime where energy confinement decreases linearly with density. 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. The measured fluctuation intensity agreed with simulation within experimental error (+/- 60%). In the saturated Ohmic regime, the simulated fluctuation k spectra 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. It will be critical for our overall predictive capability to understand and overcome this discrepancy. 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. Experiments to look at the role of the parallel electron drift velocity have begun.

2.1.2. Particle and Impurity Transport

C-Mod has investigated particle transport in low collisionality H-modes [5]. In collaboration with AUG and JET, scaling relations were derived which predict modestly peaked density profiles for ITER [6, 7].

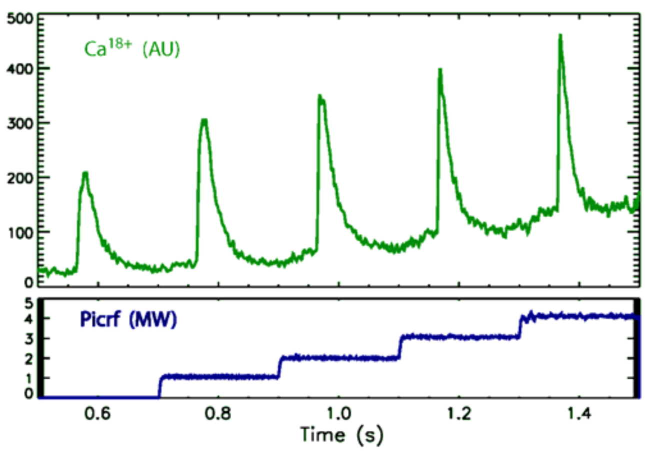

Fig 2.1: Traces of Ca^{+18} are observed with the hi-resolution x-ray diagnostics (HIREX) following multiple injections of CaF_2 into L-mode plasmas with the laser blow-off system.

Previously reported findings showed moderate peaking as collisionality was lowered to the values expected for ITER. The C-Mod results confirmed earlier experiments on JET and AUG 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. GYRO simulations have provided the first understanding of this phenomenon, and motivated revised analysis of similar plasmas in JET and AUG. These simulations began with the measured temperature and density profiles. This uniformly leads to strong overestimates of the heat transport (particularly for the ions) in some regions. The ion temperature profile is then adjusted by modifying the portion with $0.35 < r/a < 0.75$, while maintaining continuity with the measurements outside this region. The peaks and valleys in R/L_{Ti} are smoothed in order to avoid regions with strongly over-predicted heat transport and neighboring zones of stability. The resulting ion temperature profile is generally within 5-10% of the ITG threshold.

However, even when the total turbulent heat flux is close to the conducted power inferred by TRANSP, the ion/electron ratio is higher than inferred by TRANSP. With essentially no core fueling, the particle transport in these stationary plasmas must be close to zero for r/a<0.9, so GYRO is tested by varying the density gradient to find the density profile that produces the required null flux. The GYRO simulations robustly predict density peaking, independent of which Ti profile is used, and the density peaking is always stronger with lower collisionality. By varying the 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 addition, 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.

2.1.3. Impurity Transport

A powerful tool for the study of particle transport has recently been installed and commissioned. The new laser blow-off impurity injector 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. Using a multi-pulse YAG laser, the system is capable of injecting impurities up to every 100 msec throughout a plasma discharge allowing much more efficient parameter scans. Current work has focused on determining the most important dependences for impurity transport. An example of these studies is shown in figure 2.1, where calcium injections are observed by measuring the time histories of an x-ray line coming from the Ca^{+18} ion at different values of RF power in L-mode. The drop in impurity confinement time with power can be seen in the raw data this case, or from fits to the time traces (fig 2.2). The impurity confinement time is found to increase strongly with plasma current and more weakly with toroidal field. A weak dependence on plasma density has also been observed.

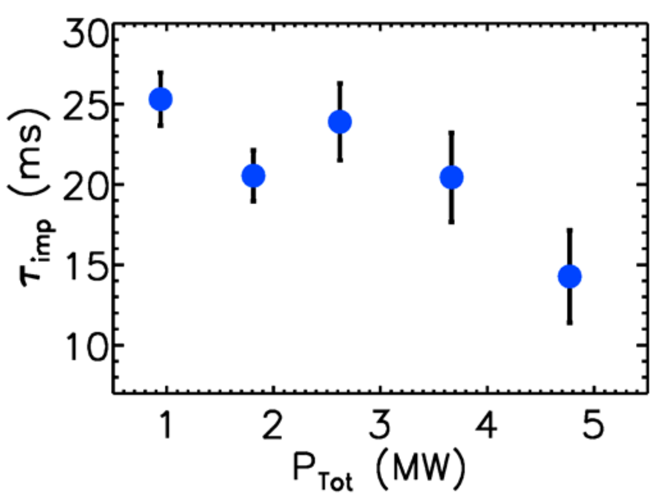

Fig. 2.2: The impurity confinement times, computed from the data in fig. 2.1 show a clear decrease with input power.

2.1.4. Intrinsic Rotation

Detailed parameter scans have been performed in order to shed light on the complicated dependences of L-mode intrinsic rotation. The behavior of the core toroidal rotation on plasma current is shown in Fig. 2.3, for the LSN configuration at fixed magnetic field and electron density. There is a linear increase with plasma current, trending in the co-current direction. It is interesting to note that for plasma currents above 0.9 MA, the rotation passes through zero and becomes co-current; under these conditions the plasmas often make a transition to Ohmic H-mode. A similar connection has been made in ICRF heated plasmas. The scaling with electron density is much more complex [8]. For very small changes in electron density, the core plasma rotation can abruptly reverse direction, with no apparent changes in other plasma parameters or performance. Similar observations have been made in TCV plasmas.

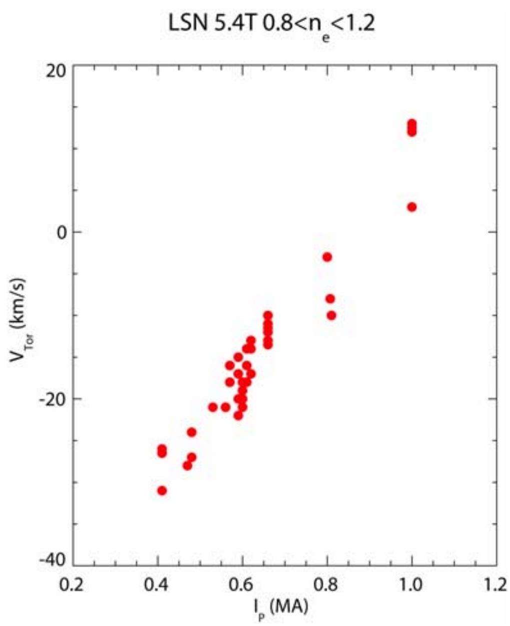

Fig. 2.3: The core L-mode rotation velocity as a function of plasma current.

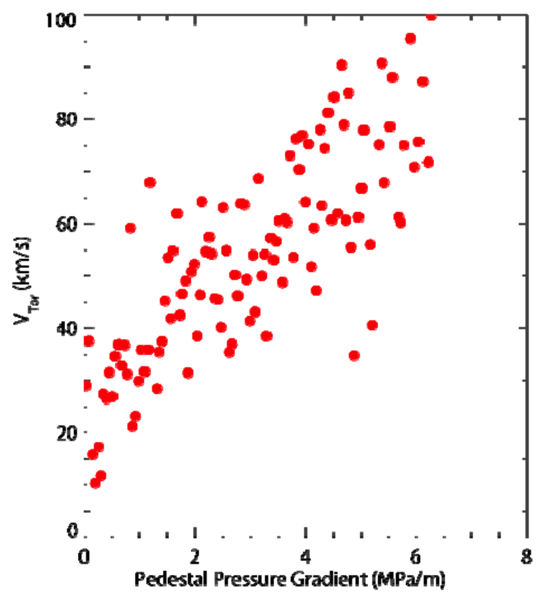

Fig. 2.4: The spontaneous rotation velocity as a function of pedestal pressure gradient.

The rotation in H-mode plasmas is generally in the co-current direction and has a much simpler parameter dependence, with the rotation velocity scaling as the stored energy normalized to the plasma current, or Mach number proportional to the normalized plasma pressure [9]. A nearly identical scaling has been found for the rotation in I-mode plasmas. This robust scaling has been observed on many devices, utilizing a variety of different heating schemes. It is interesting to note that the plasma current scaling in H-mode is the inverse of the scaling in L-mode. Since the H-mode transition is known to originate at the plasma edge, and since the subsequent co-current rotation propagates in from the edge following the H-mode transition [10], it would not be surprising if the core rotation was dependent upon an edge parameter rather than just the global stored energy. Shown in Fig. 2.4 is the core rotation velocity as a function of pedestal pressure gradient, a local quantity. This is a possible signature of the role of the residual stress in driving intrinsic rotation.

We have used GYRO simulations to explore whether the turbulent transport of toroidal angular momentum might be the cause of the 'intrinsic' rotation by removing angular momentum, and thereby spinning up the plasma in the opposite direction. The toroidal rotation profile measured by the x-ray crystal diagnostic commonly exhibits finite radial gradients (indicating local generation of 'intrinsic rotation') in the 'transport region', $0.3 < r/a < 0.8$, where application of GYRO is justified. We selected several well documented L-mode discharges for nonlinear GYRO simulations. We estimate the intrinsic rotation speed gradient predicted by GYRO by varying the input rotation speed gradient until we 'bracket' the target state of zero momentum flux that is appropriate for these stationary plasmas with no externally applied torques.

Two of the cases have strong gradients in the measured toroidal rotation; the third does not, so it may be considered a 'control' case. The results are qualitatively encouraging, in that the sign of the rotation speed predicted by GYRO agrees with experiment, and the estimated magnitude is generally within a factor of two of the experiment - GYRO tending to be lower, but not everywhere. The 'control' case is also qualitatively encouraging in that GYRO predicts an intrinsic rotation gradient that is a factor of 3-5 smaller than in the other two cases. A more theoretically consistent treatment of rotation transport in GYRO is now available, and it will be used in future.

2.1.5. Driven Rotation

ICRF mode conversion flow drive has been demonstrated in C-Mod plasmas [11]. Rotation velocities well in excess of the intrinsic rotation scaling have been observed, as shown in Fig 2.5. In order to optimize the driven flow, a detailed parameter study has been undertaken. Among the parameters scanned were the magnetic field, ^{3}He fraction, electron density, plasma current, ICRF frequency, wave phasing and ICRF power, in L- and H-mode target plasmas. For fixed ICRF frequency, an optimum magnetic field and ^{3}He fraction were found, consistent with a fast wave mode conversion picture. The dependence on the phasing was not so easy to interpret, as co-current rotation was observed with $+90^o$, -90^o and dipole phasing. At the lower ICRF powers, all three cases were similar, while at the highest powers, $+90^o$ had the highest rotation. A power-law regression analysis to the parameters yielded a rotation velocity scaling: $V \sim P_{icrf}^{\ 1.3}\ T_e^{\ 1.3}\ I_p^{\ 0.5}\ n_e^{\ -1.0}\ f[\mathrm{Mhz}]^{-1.0}$, suggestive of momentum input scaling with power per particle. This has a favorable trend with lower collisionality and higher plasma current.

2.1.6. Momentum Transport

Momentum transport can be characterized taking advantage of the apparent edge source of rotation following the L-H transition [11]. Assuming the momentum flux has the form $\Gamma = \chi_\phi \nabla V + v_c V$, the momentum diffusivity, χ_ϕ , and the momentum pinch velocity, v_c, can be determined. The momentum diffusivity is often found to be close to the ion thermal diffusivity, or a Prandtl number of unity. Peaked velocity profiles suggest the

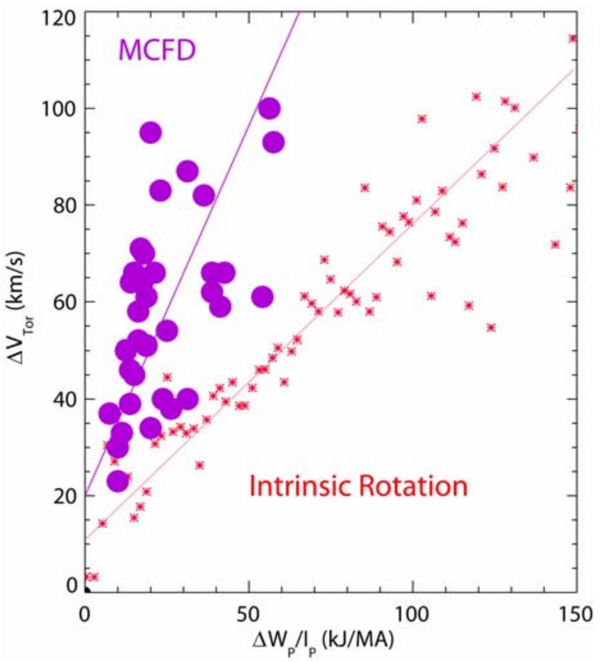

Fig. 2.5: The mode conversion flow drive velocity compared to the intrinsic rotation scaling.

workings of an inward momentum pinch. Several recent theories of the momentum pinch predict a value of -R v_c/χ_ϕ = 4 + R/L_n . This is not generally observed in C-Mod, suggesting either something missing from the pinch models, or bringing the above form of the momentum flux into question. It has recently been suggested [12] that an additional term should be added to the momentum flux, one not proportional to the velocity or gradient, namely the residual stress. The residual stress is the only component of the momentum flux that can spin up the plasma from rest and is a candidate for driving the intrinsic rotation. If the Prandtl number is taken to be one, and with the pinch velocity taken from theory, the residual stress can be the free parameter in the momentum flux. Velocity profiles are currently being analyzed in this way.

Momentum transport can also be characterized from the velocity profile response to MCFD modulation. Several discharges with MCFD modulation have been run and data collected, and analysis is underway.

2.1.7. Internal Transport Barriers

Investigations of internal transport barrier (ITB) physics examined the role of intrinsic rotation in forming and maintaining the ITB. Previous comparisons with simulations indicated that these barriers were formed by reduction of the ITG drive term, R/L_T and the action of the neoclassical pinch [1, 3]. Alcator C-Mod is unique in displaying the presence of internal transport barrier development in H-mode plasmas with monotonic q-profiles ($q_{min} \leq 1$) without the introduction of external torque to the plasma. Despite the absence of neutral beams, an intrinsic rotation develops in all C-Mod plasmas, but is especially strong after the transition to H-mode in the co-current direction. In off-axis heated plasmas the core toroidal rotation decreases and often reverses direction after the H-mode transition while the rotation outside of the half radius shows little change. In plasmas where an ITB forms, this strong core rotation continues to decrease as the ITB density becomes more peaked. The hollow rotation profile results in the formation of a radial electric field well which in turn produces an **ExB** shearing rate that is significant in the region where the ITB foot is located. Comparison with gyrokinetic stability calculations from earlier ITB plasmas suggests that this shear has sufficient magnitude to suppress the ITG growth rate that would otherwise be expected in that region. The time history (Fig. 2.6.) shows that the increase in the shearing rate occurs rapidly following the L to H-mode transition, and is notable well before the density peaking is observed. It is clear that further gyrokinetic studies must explore both the role of the intrinsic rotation as well as that of profile modification in the suppression of instabilities and the ITB formation and control.

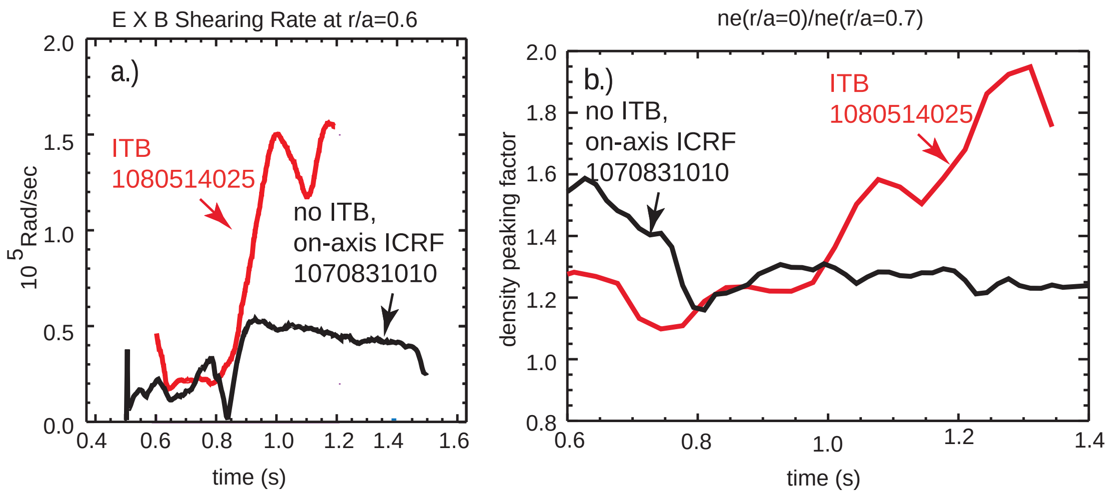

Fig 2.6: Time history of a.) EXB shearing rates at r/a=0.6 for off-axis ICRF heated EDA H-mode plasma with ITB (red) compared with centrally ICRF heated H-mode, no ITB (black) b.) density peaking for the same plasmas.

2.2. Core Transport Research Plans

Since development of predictive modeling is a major programmatic goal for the U.S. fusion program, the C-Mod transport program emphasizes close and careful comparisons with theoretical models and simulations. As discussed above, a key element in our transport research is the validation of nonlinear gyrokinetic codes via comparisons between theory/simulation and our experiments. C-Mod has unique characteristics and unique capabilities which support these studies including higher field and density; strong ion-electron coupling, standard operation with no core particle or momentum source, flow-drive via ICRF and LH for manipulation of rotation profiles and efficient off-axis current drive for manipulation of magnetic shear. The C-Mod team has active collaborations with the gs2, GYRO and TGLF 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. This work directly supports Renew Thrust 6.

2.2.1. Electron Transport

Understanding electron transport is one of the most important ongoing topics in tokamak research, and perhaps the least understood. Its importance stems from the fact that alpha particles will be slowing down mostly on electrons in ITER and in alpha dominated fusing plasmas; thus understanding the rate of energy loss through the electron channel (χ_e) is of paramount importance. While conventional wisdom following from gyrokinetic codes asserts that TEM (and to a lesser extent ETG modes) should be the dominant form of turbulence causing the loss of energy through the electron channel, to our great surprise, recent results from Alcator C-Mod contradicted this assumption [4]. In particular, in the linear ohmic transport regime ($\tau \propto nq$) we have been able to separate the electron and ion transport regimes (low density, ohmic driven plasmas) and while TRANSP clearly indicates that the energy loss is through the electron channel, nonlinear GYRO predicted the dominant turbulence to be ITG modes, and thus χ_i dominating over χ_e. This result is not understood and will be investigated in great depth in the future.

Our main turbulence diagnostic in this research is the Phase Contrast Imaging (PCI) system which has been used successfully to measure turbulence in C-Mod plasmas (as well as on DIII-D). The measured turbulence is being compared with GYRO predictions. To present date no obvious differences have been found between the synthetic turbulence predictions between GYRO and experiment. An important physics concept that is missing in GYRO (and GS2) is the impact of ohmic electron drift on micro- stability. Past theoretical works indicate that ion-acoustic like modes (driven by ohmic electron drift) should not have a large impact on electron transport. Nevertheless, our experimental results indicate a direct proportionality between electron drift velocity and transport. Namely, the largest transport is observed at the largest electron drift speed, at the lowest densities. We have begun systematic experiments to map out such relationships, including the dependence on B field and q values. The improved Thomson scattering and ECE diagnostics are greatly aiding our interpretations of the results and reducing error bars that would otherwise make such comparisons meaningless. The ion temperature profiles are determined by the HIREX diagnostics and central neutron measurements and additional TRANSP code modeling of the ion temperature profiles. This work will be continued in FY 2011. In FY 2012 we will consider reconfiguring the PCI system to allow CO_2 scattering measurements to obtain the local fluctuation levels on an absolute scale. The range of k values in such a scattering geometry may be limited to about 25-30 cm^{-1} due to the port extension geometry. Detailed design studies will be carried out to assess this possibility.

We will also consider buying a second detector system and optics, so both PCI and scattering capabilities would be available. This may be a key part of the diagnostic renewal proposal for FY 2011 through the OFES diagnostic initiative that supports the PCI hardware system, while C-Mod supports student RAs. A new detector array for the PCI system will also be considered in FY 2012, since the present system has some dead channels and thereby limited sensitivity and range. In addition, the PCI system is also being used to measure ICRF waves, and in the future will also be used to measure Lower Hybrid Waves (see Chapters 5 and 6).

Other important areas of future transport research include the following: a. Transport and fluctuation studies during I mode operation, especially when the electron temperature is well above the ion temperature (up to factors of two) so that the electron and ion transport channels can be separated; b. Further transport studies in H mode regimes with high power ICRF and Lower Hybrid heating so that Te is well above Ti; c. Transport studies in Lower Hybrid current driven plasmas, especially with profile modifications. This would take place most likely in FY 2012 when off-axis LHCD should provide current profile broadening, and in particular, shear reversal. Studies of turbulence and transport (both electron and ion) in such AT regimes are of great importance and will be an essential component of C-Mod research, unique in the US program. Both PCI and CO2 scattering systems will be deployed to study turbulence, and results will be compared with other diagnostics such as CECE and Doppler Reflectometry (see sections below). If verified, reduction of turbulence and related transport in reversed shear plasmas is of the greatest of interests in AT research, with significant consequences for reactor design.

2.2.2. Particle Transport

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. 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 density fluctuations from reflectometry and PCI will also be collected. Comparison with simulation will focus on the interplay among 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.

Tuning of input profiles for GYRO simulations of C-Mod plasmas with peaked density profiles will be aimed at achieving a close match to the heat transport inferred by TRANSP, and more quantitative comparison of the predicted density profiles will then be possible. Multi-species simulations will be carried out to determine if hydrogen isotopes are predicted to have different degrees of peaking. (Preliminary simulations indicate this will be the case for HD and DT combinations.)

Peaked density profiles have profound consequences for several aspects of ITER operation, and differential peaking of D and T could be of great importance. This motivates experiments to determine the H/D density ratio in C-Mod plasmas with peaked density. The PCI diagnostic is capable of directly observing mode-converted waves, and the location of the mode conversion provides a relatively precise measure of the local H/D density ratio. Spectroscopy provides the H/D ratio at the plasma edge, and we may be able to use reflectometer systems to determine the ratio at several locations in the outer half of the plasma that is inaccessible to the PCI system. Initial experiments are planned for 2010.

The laser blow-off impurity injector is a powerful new tool for the study of particle transport. A new crystal has just been installed in the HIREX diagnostic which will allow measurement of the full Ca^{+18} profiles and their evolution. The analysis approach will be to calculate impurity profiles and fluxes, using spectroscopy and broad-band x-ray arrays combined with modeling of the relevant atomic physics with the STRAHL code [13]. From this modeling, dependences on relevant plasma parameters can be tabulated (n, T, q, L_n, L_T, L_S) and compared to gyrokinetic models. 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 in which 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 GYRO simulations of impurity transport will be compared with experimental data for a range of species that have been injected into C-Mod plasmas by the laser blow-off system. The transport will be characterized by a diffusivity and pinch at each of several radii, and the resulting transport model will be used in a particle diffusion equation to predict impurity confinement times. 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. Preliminary simulations with multiple impurities show that interactions between species are 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 well 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 and the new polarimeter system.

2.2.3. Rotation and Momentum Transport

One theme in the upcoming research plans is isolation and identification of the residual stress. This can be addressed using the L-H transition as described above, as well as from modulation experiments, using both MCFD (data have been collected and are under analysis) and LHCD. LHCD is known to give rise to counter current rotation [14] (an effect which is not fully understood), and can also be modulated in similar fashion. Determination of momentum transport coefficients is a high priority for the ITPA momentum transport database. Another method for determining the residual stress is through observation of velocity fluctuations. A fast throughput x-ray spectrometer has been designed and built in collaboration with PPPL, and is being installed on C-Mod. It is an upgrade of the currently operating imaging spectrometer system, to measure the velocity (and ion temperature) on a sub-millisecond timescale. The prototype spectrometer should be operational shortly. This diagnostic may also be useful in measuring zonal flows. An additional area of study is rotation reversals, which occur in L-mode discharges at low density and seem to be some sort of velocity bifurcation, and which may shed further light on the intrinsic rotation mechanism. This is being pursued in a joint experiment with TCV, under ITPA TC-9.

Aside from intrinsic rotation, driven rotation is also an important research topic at C-Mod. A top priority is optimization of MCFD, which is being pursued with JET under ITPA TC-14. MCFD in JET was observed in the counter-current direction, and differences from the C-Mod results need to be understood. From the above scaling law, the velocity increases with ICRF power; those experiments were performed with only one ICRF antenna, and methods for MCFD at 8T with full power at 80 MHz are under study. Rotation velocity profile control with LHCD and MCFD look very promising, and may be the only tools available on future devices like ITER. LHCD was shown to give rise to hollow velocity profiles in H-mode plasmas at high density, even though the waves were not expected to penetrate.

GYRO simulations of momentum transport in several C-Mod plasmas will be repeated using a recently developed algorithm that is based on a consistent theoretical treatment (the original method had only a heuristic justification). The chosen plasmas exhibit 'intrinsic rotation', and the initial round of GYRO simulations agreed with the sign of the rotation, but with magnitudes that differ by factors of a few.

2.2.4. Internal Transport Barriers

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 in the core of high-density discharges (which are inaccessible to MSE) and the installation of a frequency-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, the questions we plan to address are: 1) What are the dominant instabilities which are present before the barrier forms? 2) What is the stabilization mechanism? 3)What mechanisms govern 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 to control the flow shear [15]. 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.

2.2.5. Additional Theory Comparisons

We have begun validating the TGLF transport model for Alcator C-Mod discharges. TGLF is a gyro-landau-fluid theory based transport model with trapped particle response and finite Lamour radius effect, which has comprehensive physics to approximate anomalous transport due to drift-ballooning modes in tokamaks.[16] The validation procedures involve testing the performance of this model in linear stability analysis, in quasi-linear calculation of saturated particle and energy fluxes, and in predicting reasonable steady-state ne and Te profiles based on the experimentally measured particle and energy fluxes. The results from linear stability analysis will be compared with linear gyrokinetic calculations. The quasi-linear saturated particle and energy fluxes predicted by TGLF will be compared with the experimental levels computed from self-consistent power balance using TRANSP. Finally, we will use the XPTOR code to predict density and temperature profiles based on the TGLF model that best fit the given TRANSP calculated fluxes and compare with these with experimental measurements.

2.2.6. Development of Fluctuation Diagnostics

As discussed above, the use of nonlinear gyrokinetic simulations to analyze and model experimental data with the goal of improving predictive capabilities for turbulence-induced transport is a crucial aspect of our research. Traditionally, nonlinear simulations are performed post-experiment and the transport results are compared with inferred experimental transport power fluxes or diffusivities that are obtained from power balance transport codes (e.g. TRANSP). It is also possible for information about measurable turbulence characteristics, such as amplitude, spectrum, correlation length, etc., to be obtained from these post-experiment nonlinear simulations. These results can then be quantitatively compared against fluctuation measurements in experiments via synthetic diagnostics. Here we propose to use the simulations to guide design of diagnostics and experiments to best test the models.

2.2.6.1. CECE Diagnostic and Doppler Reflectometry

It is important to make comparisons not only with density fluctuations, as has been done recently in detail at C-Mod [17], but with other fluctuating fields as well. However, local measurements of temperature or potential fluctuations in the core of tokamak plasmas are much more difficult to make than density fluctuation measurements. One method for measuring local, long wavelength electron temperature fluctuations, Correlation Electron Cyclotron Emission (CECE) [18, 19], has been successfully used at DIII-D [20] to measure electron temperature fluctuations in the core of tokamak plasmas, and these measurements have been compared with gyrokinetic simulations [21].

A new CECE diagnostic for Alcator C-Mod would help address the long-term goals of transport research on C-Mod, providing new data relevant for electron thermal transport studies. A CECE diagnostic would provide new electron temperature fluctuation data from the unique region of parameter space that Alcator C-Mod can explore: ITER relevant RF heated, high-field, high-density plasmas that are free of external particle and momentum sources. The CECE data will be used to support validation efforts aimed at improving the predictive capability of turbulence transport models.

In particular, two physics areas of interest at C-Mod where a CECE diagnostic can have the most impact are 1) providing new turbulence measurements that can be used to further investigate disagreements between GYRO and experiment in low density plasmas [4] and 2) studying differences in particle and energy transport in the improved confinement I-mode regime at C-Mod. [22]. The new CECE data, combined with synthetic diagnostic modeling would add a new constraint on the GYRO simulations of low-density, Ohmic plasmas and may help identify reasons for the disagreement between theory and experiment that has been observed. In I-mode, there is a substantial temperature pedestal, but the density pedestal is lower than in standard H-modes. Hence I-mode has low particle and impurity confinement but high-energy confinement. Measurements of both density and temperature fluctuations locally in the core plasma, and the correlations between them, will provide information about the turbulence that may aid in interpreting the differences between the transport channels. A CECE system would provide electron temperature fluctuations, which theoretically are more important for determining the turbulence-driven electron heat flux than turbulence-driven particle flux.

As part of longer term fluctuation diagnostic plans, a new fluctuation reflectometer diagnostic will be constructed and will be coupled with the CECE diagnostic to allow for simultaneous measurements of electron temperature and density fluctuations [23]. The new reflectometer will be operable as a Doppler Backscattering System (DBS) [24, 25] and also as a standard fluctuation reflectometer. When used as a DBS system, the reflectometer will provide local measurements of intermediate-k density fluctuations as well as plasma rotation measurements. When operated as a standard reflectometer, the system can be used with the coupled CECE system to correlate the electron temperature and density fluctuations. These simultaneous measurements of density and temperature fluctuations can be used to tightly constrain the GYRO code and measured correlations between the two fields can provide, with the aid of a validated turbulence model, information about both particle and energy transport.

2.2.6.2. Short Term Plan for CECE Antenna

Electron thermal fluctuations have not been observed on Alcator C-Mod with the FRCECE diagnostic even though the instrumental bandwidth would seem to be adequate. By addressing this issue first -- before completing the design for the advanced CECE diagnostic, we can inform the design for the advanced system and further assure its success and possibly achieve some initial results sooner. The absence of evidence for electron temperature fluctuations in the observations with the FRCECE diagnostic suggests that improvements in the optics are needed to reduce the size of the imaged volume. The FRCECE system uses cylindrical mirrors to collect the ECE radiation. An elliptical mirror focuses the radiation in the poloidal direction. Two smaller parabolic mirrors focus the radiation in the toroidal direction. Each of the two parabolic mirrors focuses the radiation on to separate waveguides which conduct the radiation to the RF sections. This results in an asymmetric focal spot in the plasma with the toroidal direction having a spot size approximately 3 times the poloidal spot size. It was believed that long coherence lengths in the toroidal direction would make fluctuation

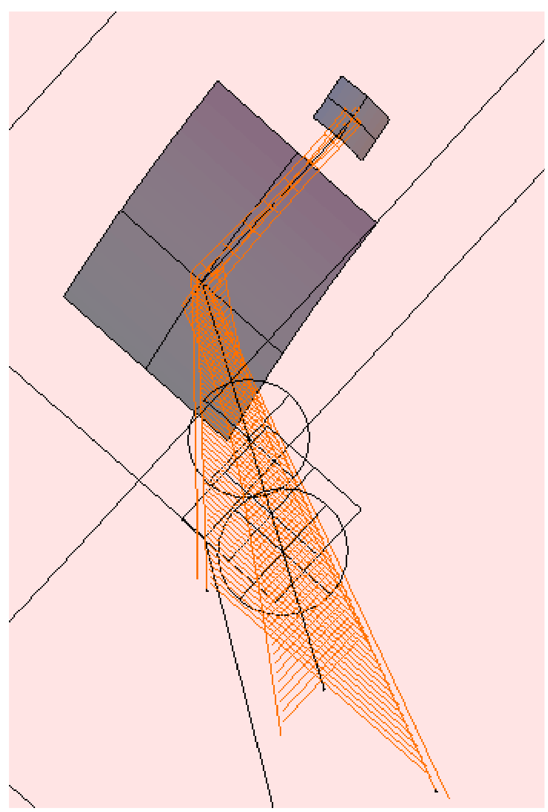

Fig. 2.7: Optical layout for CECE

measurement possible even with the large spot size. Fluctuations are not observed. This suggests that new optics are required to produce a smaller, symmetrical focal spot. This change will result in additional modifications downstream in the RF and IF modules.

The major system modifications, being carried out in collaboration with the U. Texas-Austin team,include a new first mirror and a reduction in number of waveguides to the RF units from two to one (see fig. 2.7.). A new ellipsoidal mirror will focus the plasma radiation on to one tapered waveguide. The poloidal spot size will remain the same (~2cm) but the toroidal spot will be reduced to be comparable to it. Since one waveguide will replace two between the mirror and the RF unit, quasi-optical filters will be added to redirect the low and high frequency radiation to the appropriate RF units. Wire filters, similar to the ones we now use to eliminate the third harmonic frequencies, will be used to reflect the low frequency radiation to the low frequency RF unit while passing the higher frequency on to the high frequency RF unit. An additional high resolution IF section will be added to detect fluctuations. This would be similar to the correlation IF system used in the initial FRCECE system. It would consist of an IF amplifier, power splitter and anywhere from 4 to 8 narrow band filters (~200MHz) that would be closely spaced to allow cross-correlation to reduce the plasma noise. An alternative to this new IF section would substitute tunable filters for the fixed filters. This would allow the frequency of each channel to be selected and optimized for the measurements.

2.2.6.3. Advanced CECE Diagnostic Design Using Nonlinear GYRO Results

Integrating nonlinear GYRO simulation results into the design of the new CECE diagnostic, which ultimately must successfully detect the predicted fluctuations, will itself be a test of the turbulence model employed by the gyrokinetic codes and is a very unique and novel opportunity in the area of fluctuation diagnostic development. Perhaps the most important factor in the design of any new turbulence diagnostic is assessment of the sensitivity to broadband, low amplitude fluctuation levels in a relevant wavenumber range. Design of turbulence diagnostics always relies therefore on some knowledge, either based on theory or past experiments, of the turbulence characteristics expected in the plasma of interest.

To develop the new CECE diagnostic for C-Mod we will employ traditional experimental methods for optical and radiometer receiver design, and we have some experimental knowledge of long wavelength density fluctuations in C-Mod from past work with reflectometers and Phase Contrast Imaging diagnostics. However, given the sophistication of presently available turbulence simulations capabilities, we can also extensively use nonlinear GYRO simulation results to inform design of the CECE system at C-Mod. Nonlinear GYRO simulations will provide us with predictions for the electron temperature turbulence in C-Mod plasmas using the most complete gyrokinetic turbulence model available. To incorporate GYRO results into the CECE design process, we will need to develop and implement several different synthetic CECE diagnostics to analyze results from GYRO. This set of IDL-based synthetic CECE diagnostics will allow us to produce predictions from many plasma conditions at C-Mod for the percent level of fluctuations, "synthetic" frequency power spectrum, and correlation length given different possibilities for the antenna pattern, filter width, filter separation and radiometer sensitivity. An existing database of nonlinear GYRO simulations of Ohmic, L-mode and H-mode C-Mod plasmas is available locally, as several graduate students in our group have used nonlinear GYRO runs to analyze a variety of plasmas at C-Mod.

For a CECE system, important limits on the detectable wave number range of turbulence are determined both by the EC emission physics and by the fundamental principles of radiometer-based ECE diagnostics [20]. The design of both the optical system used and the details of the intermediate frequency (IF) section of the radiometer will benefit from GYRO-based diagnostic design.

First, the optics used to collect the thermal black body radiation from the plasma and couple it to the radiometer receiver must provide a small enough spot size (antenna pattern) to allow for resolution of turbulence in a wavenumber range of interest. Specifically, the maximum resolvable poloidal wavenumber is directly related to the diameter of the antenna pattern, or the spot-size of the system. It is therefore very important to understand what spatial scale (wavenumber range) of turbulence is expected in order to design an new optical system that minimizes the impact of wavenumber filtering; allowing for more improved turbulence measurements. Nonlinear GYRO results provide not only the wavenumber spectrum of the turbulence, but IDL post-processing can be used with synthetic diagnostics to directly model the measured frequency spectrum and fluctuation levels - as they would be detected by the future CECE diagnostic. This will provide information essential to the design of the new optical system.

Second, if a frequency decorrelation CECE system [20] is used, knowledge of the expected radial correlation length of the turbulence informs the design of the intermediate frequency (IF) section of the CECE radiometer. Both the width of the IF filters and the spacing in frequency will impact the measurements. This is because the reported value of a relative electron temperature fluctuation level is obtained from a two-channel (two-point) measurement. The two-point measurement is required in order to decorrelate the thermal noise present in a single radiometer channel. The thermal noise level is a fundamental property of radiometer measurements and cannot be reduced to levels below turbulence fluctuation levels with improved single channel design considerations alone. Thus, for the frequency decorrelation CECE method to measure low amplitude, broadband temperature fluctuations, two radiometer signals, separated by less than the correlation length of the turbulence, are cross-correlated in order to discriminate the temperature fluctuation information, $\tilde{T}_e/T_e \sim 0.3-2.0\%$, from the background thermal noise. Note that for a standard ECE radiometer channel the irreducible thermal noise level can be in the range $\tilde{T}_{rad}/T_{rad} \sim 5-20\%$. Nonlinear GYRO results combined with IDL post-processing can be used with synthetic diagnostics to directly model the effects of the finite turbulence radial correlation length on the CECE measurements, as synthetic CECE channels can be radially separated in the simulations. This will allow for optimal selection of the IF filter width and spacing for the new CECE system.

2.2.7. Support for ITER and Connection to ITPA Activities

Research on C-Mod directly is well aligned with ITER high-priority transport issues. Most of this program is described in Chapter 8, which describes research on baseline scenarios. In addition, the core transport program addresses the following issues (which also correspond to parts of Renew thrust 4).

- Momentum transport/intrinsic rotation: Study and characterize rotation sources, transport mechanisms and effects on confinement and barrier formation (ITER rotation profile)
- Understand the collisionality dependence of density and impurity transport (ITER density profile)
- Quantitative tests of fundamental features of turbulent transport theory via comparisons to measurements of turbulence characteristics, code-to-code comparisons and comparisons to transport scalings.
- Test understanding of improved core transport regimes with reactor relevant conditions, specifically electron heating, Te~Ti and low momentum input, and provide extrapolation methodology

C-Mod is involved in or considering participation in a subset of the ITPA joint experiments on core transport. These include:

- TC-2 Access to $H_{98} \sim 1$ Regimes
- TC-9 Scaling of intrinsic momentum
- TC-10 Experimental identification of ITG, TEM and ETG
- TC-11 He profiles and transport coefficients
- TC-13 Ion temperature profile stiffness
- TC-14 RF rotation drive
- TC-15 Dependence of momentum and particle pinch on collisionality

2.2.8. References

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3. 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

1. Commiting resources to improving diagnostics in the scrape-off layer and pedestal regions
2. Designing dedicated experiments to study pedestal transport, edge relaxation mechanisms and L-H thresholds
3. 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 lower-density machines. H-mode studies have enjoyed access to a wide range of 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 [1], allowing empirical scalings in pedestal parameters to be examined with significant variations in dimensional and non-dimensional parameters. Variations in plasma shaping and the addition of active pumping have also been used to extend our H-mode data set.

3.1. Recent Pedestal Physics Highlights

Significant experimental time has been devoted to the question of H-mode access, which constitutes not only a complex physics problem, but also a critical practical question for ITER. This is due to the fact that available power is expected to be marginal with respect to H-mode threshold power, as projected from existing scaling laws. Recent experiments have examined thresholds for H-mode access in atypical plasma regimes. For example, prompted by questions about H-mode access in the non-activated phase of ITER operation, we found the H-mode threshold power to be substantially higher in helium than in deuterium plasmas. This experiment will be discussed further in Chapter 8. Also, recent experiments have exploited operation with unfavorable grad-B drift direction to access high performance I-modes (discussed later in this chapter), and in the process we have learned more about L-H power thresholds in these configurations. As seen in Figure 3.1, in addition to increasing with B_T, as is routinely observed, the power threshold increases with decreasing q_{95}. By elevating $P_{L\text{-}H}$ factors of 2—3 above the standard scaling law predictions, we suppress H-mode formation and promote I-mode. A new fast 2D gas puff imaging diagnostic is now allowing characterization of fluctuations and their propagation in the edge, prior to and across the L-H transition [2]. This diagnostic and some of its initial results are described in Chapter 4.

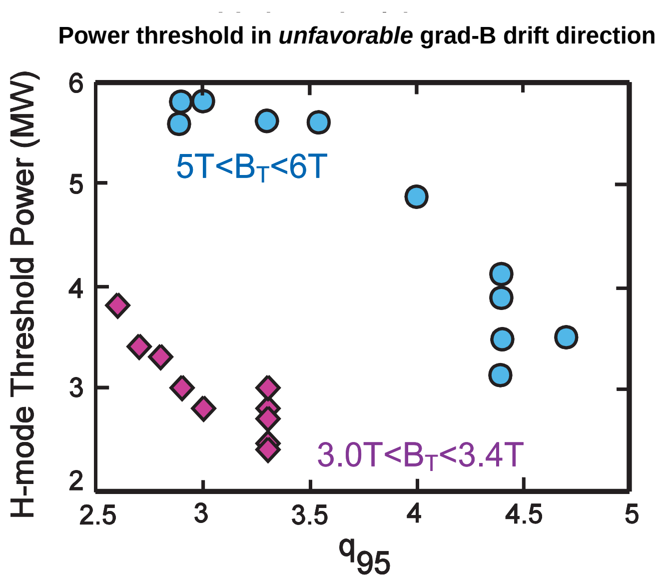

Fig.3.1: Absolute power threshold for H-mode access in discharges with unfavorable grad-B drift direction. Not only does $P_{L\text{-}H}$ increase with magnetic field, but a strong q_{95}-dependence is observed.

Because of its substantial leverage on performance in ITER scenarios, there is a significant effort on all major tokamaks to identify the important physical mechanisms determining the H-mode pedestal structure. A fundamental consideration is the pedestal width, Δ. 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 [3], 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 most common 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 3.2). In collaboration with P. Snyder (GA), experiments were designed in which the EPED1 model [4] for pedestal height and width could be tested on C-Mod. For the initial experiments, run in 2009, predictions of the pressure pedestal were made in advance over a range of plasma current. Preliminary results show that experiment roughly agrees with prediction in discharges with prevalent ELMs. We plan to expand the ELMy data set in FY11 and use it to test more rigorously the

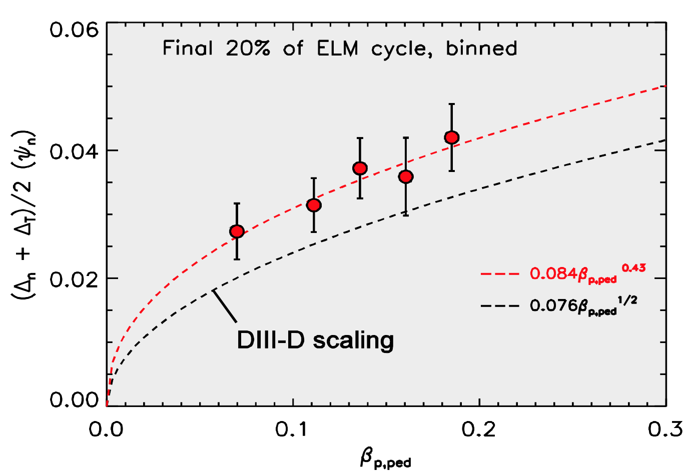

Fig. 3.2: Characteristic pedestal width as a function of poloidal beta, in the last 20% of ELM cycles.

EPED family of pedestal height models.

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 (ΔR_{SEP}) becomes comparable to the pedestal width, H-mode character changes. Examples of this influence are shown in Figure 3.3. Substantial degradation in energy confinement and particle inventory are observed when discharges are run near DN, but biased slightly in the unfavorable ∇B drift direction. However, for discharges biased slightly such that the primary x-point is in the favorable ion ∇B drift direction, confinement is generally quite good. This result is favorable to ITER, which is designed to run with a ΔR_{SEP} value that would scale to -5mm on C-Mod. In the $H_{98}\geq 1$ near-DN discharges, 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 [5]. Two threshold conditions that correlated equally well with obtaining the small ELMs were $\beta_N>1.3$ and $T_{PED}>600$eV. 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.

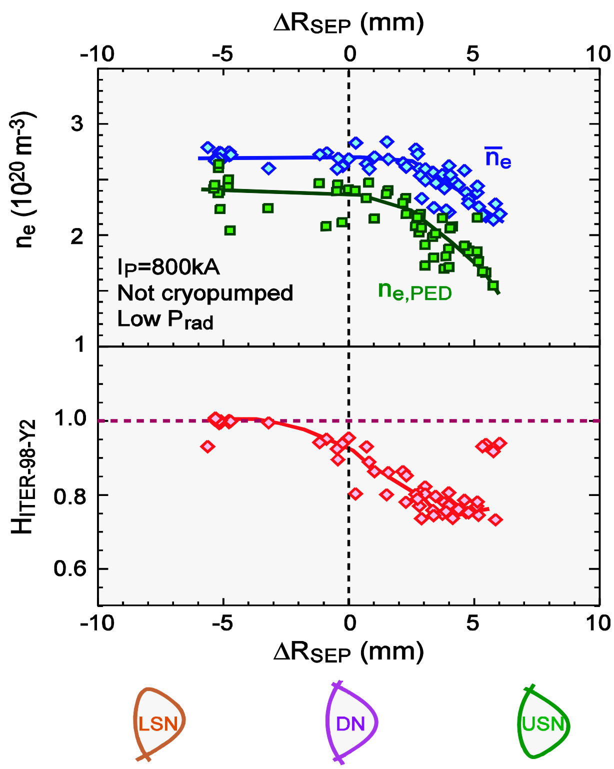

Fig. 3.3: Confinement and pedestal parameters are sensitive to magnetic balance in proximity to double null.

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 ($n\sim B_T/R_0$) [6,7]. 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 [8].

Significant advances have been made recently in the exploration of the I-mode regime [9]. These discharges have substantial temperature pedestals (approximately 1keV in the best cases, as shown in Figure 3.4) and particle transport similar to that in L-mode. I-modes are made by running discharges with unfavorable ion ∇B drift direction, and maintaining input power slightly below the L-H power threshold, which, as described above, is 2—3 times that in a favorable drift direction case. First described on ASDEX Upgrade as an "improved L-mode" [10], it is now clear that these discharges have H-mode-like energy confinement complementing their L-mode like particle edge. I-modes have been maintained steady-state for several confinement times, and exhibit continuous particle and heat exhaust. An E_r well has been observed to form in the I-mode pedestal, with ExB shearing rate intermediate between that of L-mode and H-mode [9]. Scoping studies that employed strong shaping, high current and cryopumping have yielded normalized confinement H_{98}>1 and determined techniques for maintaining the I-mode for many confinement times without a transition to H-mode. Despite having pedestal collisionality and pressure comparable to Type I ELMy discharges, ELMs are not typically 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 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.

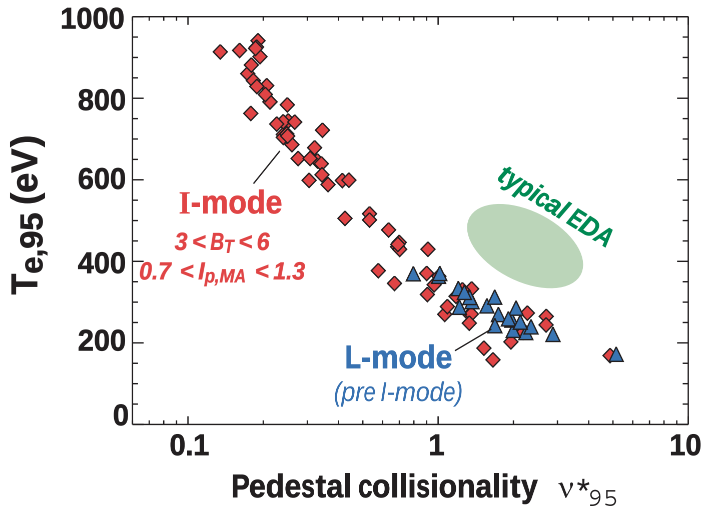

Fig. 3.4: Pedestal collisionality is substantially reduced in I-mode, relative to typical EDA H-mode operation.

3.2. 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 longer 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 local edge conditions 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 OFES 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 already been made. Figure 3.5 highlights a number of the high spatial resolution (<3mm) diagnostics utilized 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.

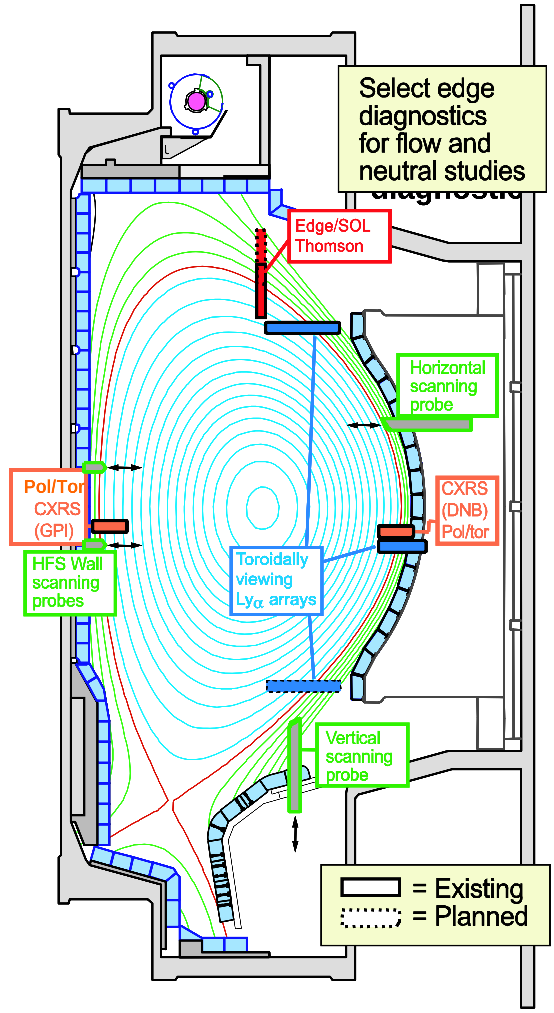

Fig. 3.5: Existing and planned edge diagnostics pertinent to pedestal and L-H transition studies.

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. Because ITER has requested an accelerated schedule, considerable effort will come in FY10—11, continuing in FY12. Foremost is the identification of critical local parameters that determine the occurrence of L-H transitions (PEP-26). 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

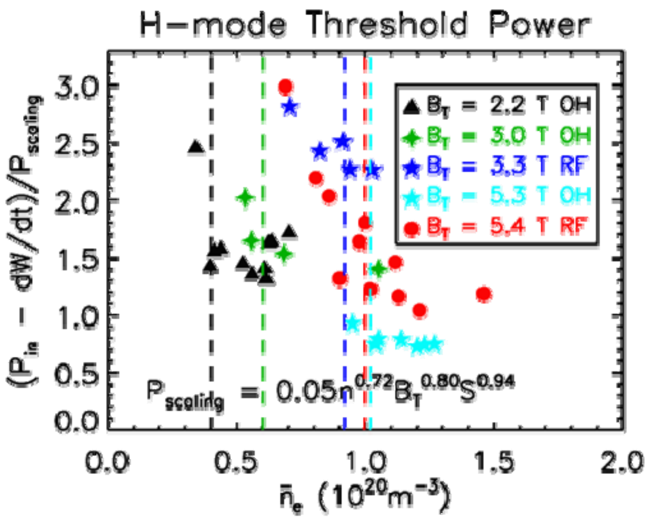

Fig. 3.6: Ratio of input power at L-H transition to the ITPA power threshold scaling, as a function of density. Discharges of varying toroidal field and heating scheme are included.

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 I-mode, and thus collect more data prior to the L-H threshold. Experiments are being performed or designed to test the effects on the L-H transition of neutral fueling location, proximity to double null topology (PEP-6), and X-point proximity to wall/divertor surfaces (PEP-28). Finally, as part of a high priority ITPA joint experiment (TC-3), C-Mod has examined the phenomenon of the “low-density limit” for H-mode across a range of plasma current and magnetic field. As seen in Figure 3.6, in ICRF heated plasmas, the power required to induce a L-H transition increased sharply as line-averaged density decreased below a value of approximately $1\text{x}10^{20}$ m^{-3}, independent of current and field. Low-field Ohmic plasmas demonstrate evidence of a reduced low-density limit. Further analysis will be performed on these data, in order to uncover physical explanations for the low-density limit.

Tied to the question of H-mode access is the means of H-mode *suppression*, and our optimization of I-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 I-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 broadband edge mode in I-mode, which appears to mitigate the density barrier. Possible similarities to the edge harmonic oscillation (EHO) first reported by DIII-D will be explored.

The physical mechanisms governing pedestal structure form another major research component on C-Mod, and will be an especially high priority topic for the next three years, particularly 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. Inspired by results that show pedestal widening at both extreme shaping, as well as at high edge *q*, [3,6] 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 FY12, attempts will be made to modify magnetic shear near the edge actively using LHCD.

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. Experiments to systematically analyze the transient evolution of pedestal profiles toward their stationary values, and to correlate the onset of stiffness to changes in edge turbulence, have begun in collaboration with R. Groebner (GA). High resolution profile diagnostics are often slow compared to the transients following an L-H transition, and so ensemble data sets from repeated H-modes will be used. Figure 3.7 shows an example of pedestal pressure evolution measured with effective resolution of 2—3ms. These data, combined with pedestal fluctuation measurements, should provide insight into (a) parameters controlling turbulence onset and (b) the role of fluctuation induced transport in setting edge gradients.

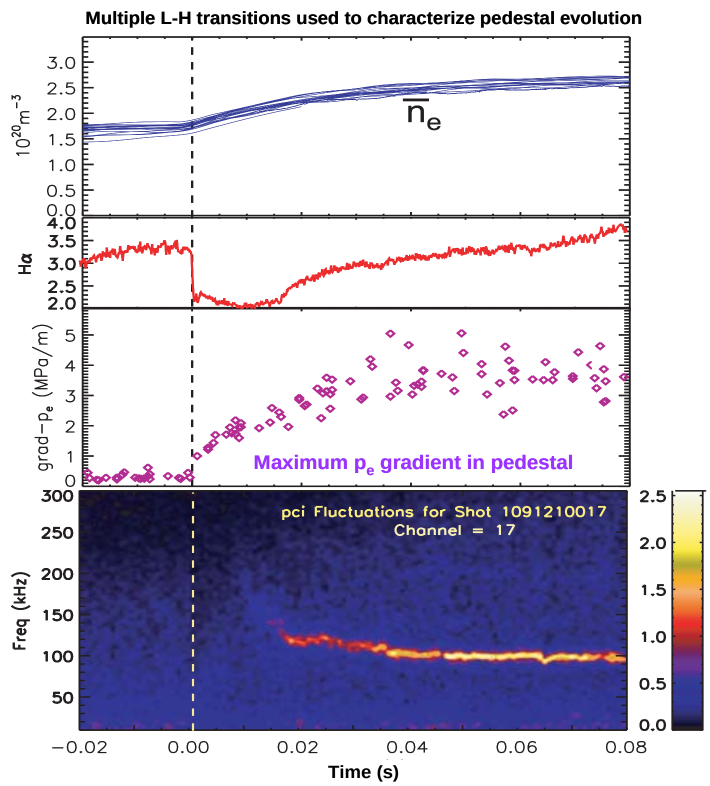

Fig. 3.7: Ensemble data sets allow the characterization of pedestal transients with high temporal resolution.

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, including that of ion temperature, and test whether 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 [6], a density pedestal which is only weakly influenced by neutral source. 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 and q_{95}, and determine the limits of edge fueling due to neutral opacity. Improved experimental measurements of neutral emissivity at multiple poloidal locations, highlighted in Figure 3.5, 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 diagnostic suite. Contributions will be made to an ITPA multi-machine profile database for the purposes of modeling the fueling? 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 (an example of the mode is shown in Figure 3.7), 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 [11]. 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 I-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.

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 [12], 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").

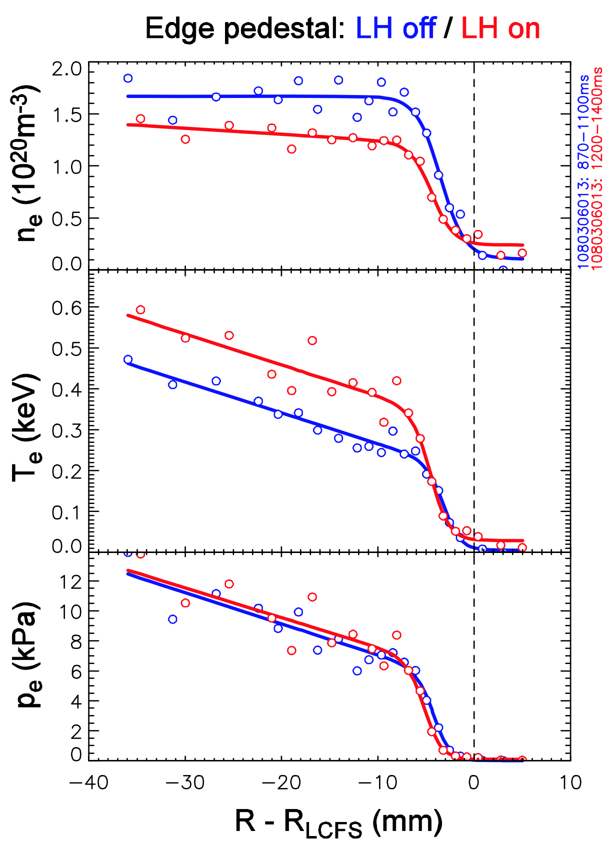

Figure 3.8: Pedestal profiles before (blue) and during (red) application of lower hybrid power to an ICRF-heated H-mode.

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. Application of lower hybrid (LH) power into EDA H-mode has been 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 [13]. An example of the modification to the electron temperature and density profiles is shown in Figure 3.8. Increases in the edge particle transport are observed to be quite prompt and require relatively small amounts of LHRF power, suggesting the pedestal modification does not require core current drive. The particle pump-out, while not well-understood, suggests that LHRF could provide a useful tool for H-mode density control. 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 LHRF on ELMy plasmas, both in terms of pedestal structure and ELM behavior. Ultimately (FY11 and beyond), when sufficient LHRF power is available, we propose to test whether application of LHCD near the pedestal can influence the frequency of the ELMs. This work will also extend our efforts to characterize the role of magnetic shear on pedestal width. As part of a recently 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.

3.3. Research Contributions to ReNeW Thrusts, and ITER and ITPA Priorities

Pedestal research on C-Mod makes contributions relevant to a number of thrust components described in the 2009 ReNeW report, “Research Needs for Magnetic Fusion Energy Sciences”:

- Thrust 2: Control transient events
    - Edge plasma transport and stability, emphazing ELM-free regimes
- Thrust 4: H-mode access and dependence on ion species.
    - Heating power required for attaining several regimes
        - L-H and H-L
        - Type III ELMy H-mode
        - H_{98}~1 H-mode

    - Isotope mass and species scaling (i.e., hydrogen and helium plasmas) of the above regimes
    - Develop strategies for minimizing the power requirements
    - Excellent edge diagnostics
        - Density, temperature and flows
        - Edge fluctuations
    - Assess different heating schemes (ICRF vs. Ohmic vs. LHRF)
- Thrust 4: H-mode pedestals
    - *What is the physics of the edge pressure pedestal in high-confinement mode (H-mode) plasmas and how does it integrate with core models of heat and momentum transport?*
    - Comparison of pedestal structure with modeling
    - Test models of the H-mode pedestal structure and of the complete ELM cycle. (including low torque)
    - Impact on the H-mode pedestal of
        - Helium or hydrogen operation
        - P_{in}/P_{th}~1
        - Near DN
        - High neutral opacity
- Thrust 6: Develop predictive models
    - Strong connection to theory/modeling
    - Essential contributions of data for model validation

Also, this proposal is well aligned with ITER R&D, 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 H_{98}=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
- PEP-26: Critical edge parameters for achieving L-H transition
- PEP-27: Pedestal profile evolution following L-H transition
- PEP-28: Physics of H-mode access with different X-point height
- 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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4. Boundary Plasma Physics

The performance of future magnetic fusion devices will ultimately depend on the degree to which plasma-surface interactions can be understood and controlled. Yet, it is widely recognized that significant gaps exist in the present knowledge base [1]; these must be addressed, both in the near term to support ITER and on a longer time-scale to establish a credible pathway forward for a fusion DEMO. C-Mod's Boundary Plasma Physics Topical Group targets research in a number of these critical areas. A central research theme is *edge plasma turbulence and transport*, since this physics ultimately controls the heat loads onto and impurity sources from first-wall surfaces. Boundary layer heat and particle flows (parallel and perpendicular to field lines) and divertor physics are important sub-themes. Understanding the detailed physics of *plasma-surface interactions* is crucial for reactors, including the topics of fuel retention in a high-Z environment, effects of RF waves on edge plasma and plasma-wall interfaces, and erosion and impurity transport effects on the core plasma. In parallel with these physics-oriented research efforts, C-Mod's Boundary Plasma Physics Topical Group also supports *research and development of first wall components* (molybdenum and tungsten tiles, divertor) with application for a fusion DEMO.

4.1. Transport Highlights from FY09

Plasma flows have been found to exhibit a remarkably rich phenomenology in the C-Mod boundary layer, setting a toroidal rotation 'boundary condition' on the confined plasma [2] as well as potentially controlling edge transport via flow-shear regulation [3]. Over the past year, the root cause of the near-sonic parallel flows that are seen on the high-field side (HFS) scrape-off layer has been investigated in detail using C-Mod's unique arrangement of three scanning Langmuir-Mach probe diagnostics, including a novel scanning-probe on the HFS midplane [4]. Through these measurements, the combined roles of anomalous cross-field transport and neoclassical drifts have been unambiguously unfolded for the first time (Fig. 4.1.1). Turbulence-driven radial fluxes on the LFS are shown to be the principal driver of the parallel flows which peak in magnitude on the HFS; turbulence-driven radial fluxes on the HFS are essentially zero, within experimental uncertainties. This work, which was the topic of a recent PhD thesis [5], also indicates that the

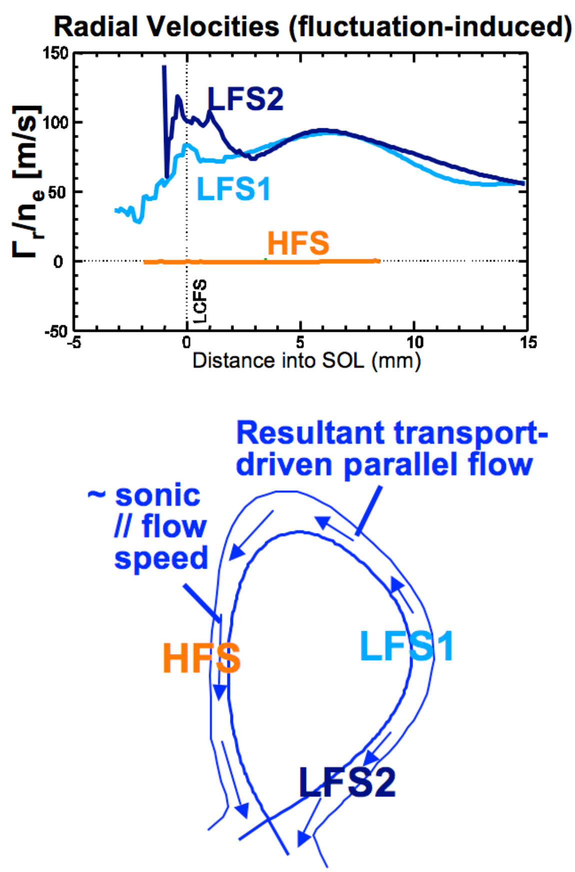

Fig. 4.1.1: Measurements of fluctuation-induced radial particle fluxes at three poloidal locations (top) are found to be quantitatively consistent with the measured parallel flows at those locations and particle continuity [5] (bottom).

resultant mass flow loop does not reconnect to the confined plasma via an inward radial pinch on the HFS midplane, as some have speculated. The research has also uncovered evidence that volume recombination near the inner divertor leg may play a role in closing the mass flow loop. This latter topic remains open for further investigations.

Over the past year, gas-puff imaging (GPI) of plasma turbulence at the outer midplane has been greatly enhanced with the installations of a new 2-D fiber array (10 x 9) coupled to an array of 90 avalanche photodiode detectors and a new fast-framing camera (390 kHz maximum frame rate for a 64x64 pixel frame) that also views this region using a new coherent quartz fiber bundle.

With the 2 MHz system, we examined the turbulence in the edge and SOL, distinguishing three connected, but distinctly different regions: 1) *inside* the LCFS - the *edge* - where the dominant broadband turbulence propagates poloidally in the electron diamagnetic drift direction, 2) in the far SOL, where the turbulent structures (blobs) typically move radially out and move poloidally in the ion diamagnetic drift direction, consistent with the E × B direction and magnitude there, and 3) a crossover region around the separatrix. This is illustrated in Fig. 4.1.2, where the conditional (k_{pol},f) spectra of L-mode plasmas are shown [6]. We find that it is primarily the turbulence of the *edge* region that is closely linked with cross-field particle fluxes in the far SOL. In the large-gradient edge region, the poloidal wavenumber-frequency spectra of the electron-drift-directed L-mode turbulence show a clear break in slope of power vs k_{pol} (Fig. 4.1.3), with $k_{pol}^{break\text{-}in\text{-}slope}$ ρ_s = 0.09 − 0.17 over a range of magnetic fields and densities. Below the characteristic $k_{pol}^{break\text{-}in\text{-}slope}$ there is a clear dependence of the spectra upon Greenwald fraction,

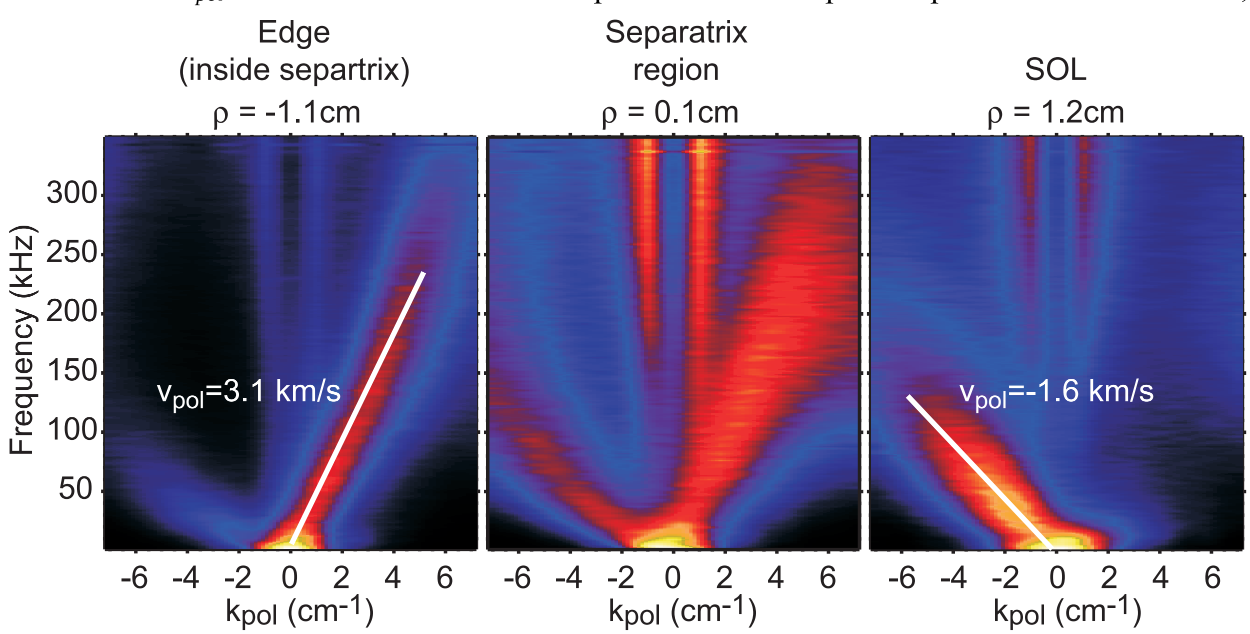

Fig. 4.1.2: Frequency and poloidal-wavenumber resolved fluctuation spectra of plasma turbulence taken simultaneously at three radial locations across the boundary layer plasma Since typical turbulence spectra fall off rather rapidly with both k_{pol} and frequency, we normalize the spectra to every frequency band, forming the conditional spectra $S(k_{pol}|f) = S(k_{pol}, f)/S(f)$.

with the power of the turbulence increasing with n_e/n_G, and the power filling in as the edge becomes more turbulent and spectral transfer is enhanced. In ELMfree H-mode plasmas, this electron-drift-direction-propagating turbulence shows a large (>10x) drop in power and speeds up to ~25km/s, approximately consistent with a concurrent increase in the E_r-well depth there. If the ELMfree H-mode evolves into an H-mode with a QCM, the QCM develops in this region with $k_{pol}^{QCM}\rho_s = 0.1$, suggesting a possible connection with the turbulence input scales seen in L-mode. On the other hand the *SOL* turbulence is largely unaffected (when normalized to average brightness) by the changes brought about by the L-H transition. This observation ties in well with previous results from the SOL in ref. [7], where the far-SOL density is shown to systematically fill in, as the Greenwald fraction is increased. Indeed, the relative intensity of the *edge* turbulence increases for n_e/n_G>0.35, approximately the same value where the relative SOL-to-edge density increases, indicating an enhanced particle transport into the SOL. If we hypothesize that the far-SOL plasma is fed primarily by the blobs born in the near SOL and separatrix region, then it should follow that the SOL turbulence stays fixed relative to the SOL density. Even though it is not possible to know from these measurements the exact physics of the underlying instability for blob generation, the picture that is starting to take shape from these findings and those of [3] seems very similar to the one presented in [8], and is roughly the following. The edge density and temperature profiles are set by gradient-limiting transport mechanisms. This manifests itself as the QCM in the EDA regime, and as broadband turbulence in L-mode (via spectral transfer from this instability to larger features. This development towards larger features appears to be hindered if: a) n_e/n_G is small (peaked spectra), or b) there is a strong edge flow-shear as is presumed in ELM-free and EDA discharges. These edge turbulence features are wavelike and their amplitude may become large enough to go beyond a critical gradient at which plasma is pushed across the field, at which point, instead of a wave motion, separate blobs of plasma travel through the SOL following v_E of that region. Of course this picture is inferred from the measurement and remains conjecture. Thus further investigations of these ideas are the subject of our research plans (Section 4.3).

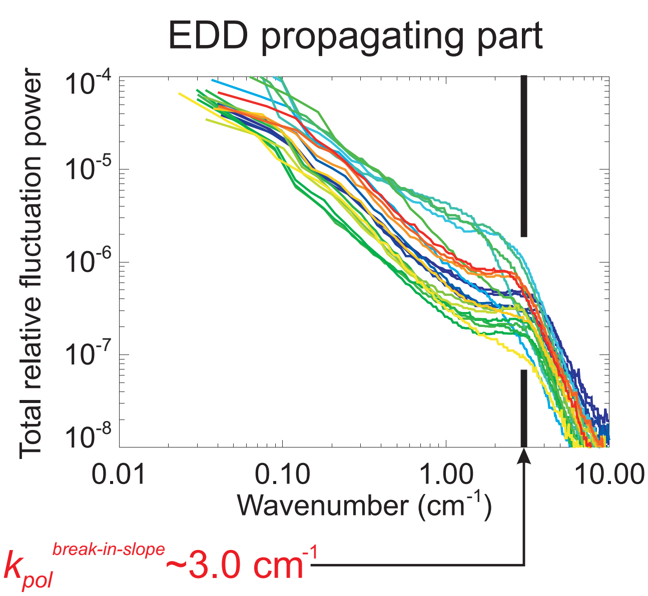

Fig. 4.1.3: The emission fluctuation power spectra (relative to the DC emission level) vs k_{pol} for the edge turbulence propagating in the electron-diamagnetic-drift direction. Note a break-in-slope of the power typically occurring around k_{pol}~3 cm^{-1}. The data are for range of plasma conditions, spanning $0.4<I_p<1.0$ MA, $2.6<B_t<6.8$ T, $0.16<n_e/n_G<0.45$. The high-resolution k_{pol} values have been calculated from $k_{pol}=2\pi f/V_{pol}$, since V_{pol} is seen to be constant over the measured range of k_{pol} and f.

The new fast camera views the same region as the 2MHz system with lower time resolution, but with higher spatial resolution (~2mm) and with viewing chords approximately aligned with the local magnetic field. Research using this fast camera has centered on SOL turbulence. The primary concern has been the role of turbulence in setting the perpendicular heat flux and density

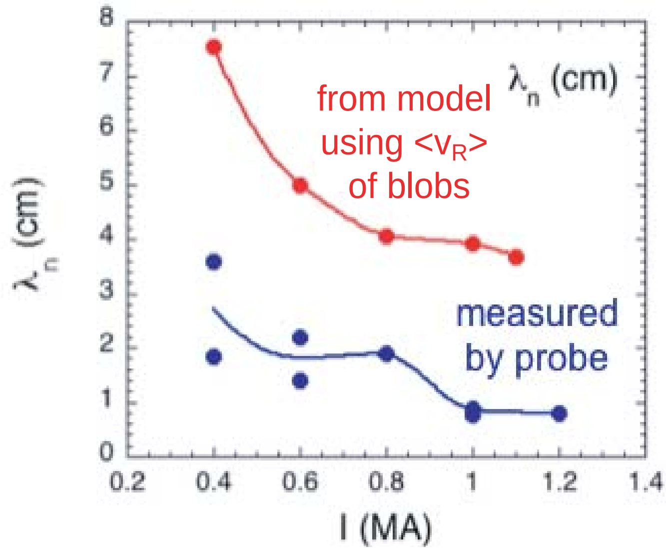

Fig. 4.1.4: The density profile scrape-off-layer widths as a function of plasma current, where the widths calculated using the radial velocity of the emission structures (see text) are compared with the widths measured with a scanning probe. The SOL region investigated is r=1-2 cm.

scrape-off widths. This is in support of the DoE Joint Facilities Milestone for FY2010 [9], discussed in more detail below. In this analysis we use a highly simplified model for the SOL density widths, assuming the perpendicular transport is purely convective. Then $\lambda_{SOL} \sim\langle v_r\rangle\ \tau_{||}$, where $\tau_{||}$ is the parallel transit time, which for particles is assumed to be $\sim L_{||}/v_{||}$, with $v_{||}{\sim}0.5\ \zeta c_s$, i.e. half the warm ion sound speed. Assuming further that $\langle v_r\rangle$ is the average radial velocity of the *emission blob structures* in the camera images, we compared $\lambda_{SOL}{}^{ne}$ with that determined using probes, with the result is shown in Fig. 4.1.4. Given the uncertainties in both determinations are at least a factor of 2, the ~x2 disagreement is not surprising. Further refinements of this model are being implemented.

Alcator C-Mod has presently begun a series of joint experiments with DIII-D and NSTX to measure “divertor heat flux profiles and plasma characteristics in the tokamak SOL…to investigate the underlying thermal transport processes.” This activity is in support a DoE Joint

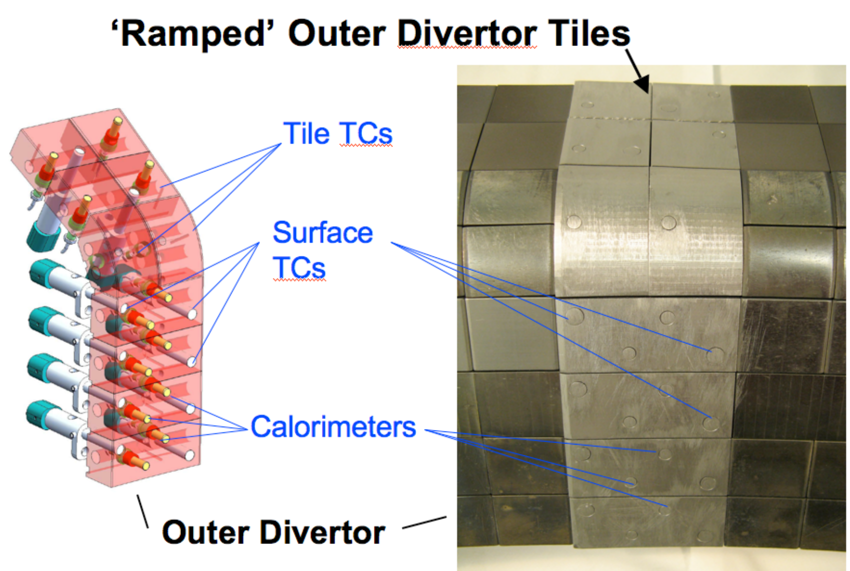

Fig. 4.1.5: A series of instrumented ‘ramped tiles’ were installed in the J-divertor module during the 2009 maintenance break. Embedded sensors include 14 calorimeters, 10 tile thermocouples and 10 surface temperature sensors.

Facilities Milestone for FY2010 [9]. In preparation for this research, a significant investment in divertor heat-flux instrumentation was made during the 2009 maintenance period. This is because C-Mod had not been equipped to make direct measurements of divertor heat flux profiles. An array of embedded heat-flux sensor probes, including tile thermocouples, calorimeters, surface thermocouples, was installed in an outer divertor module (see Fig. 4.1.5) combined with an improved IR thermography system (ElectroPhysics Titanium 550M camera), see Fig. 4.1.6. A new cPCI-based data acquisition system was installed to support the new sensors and to upgrade the old Langmuir probe data acquisition system as well. In addition, a

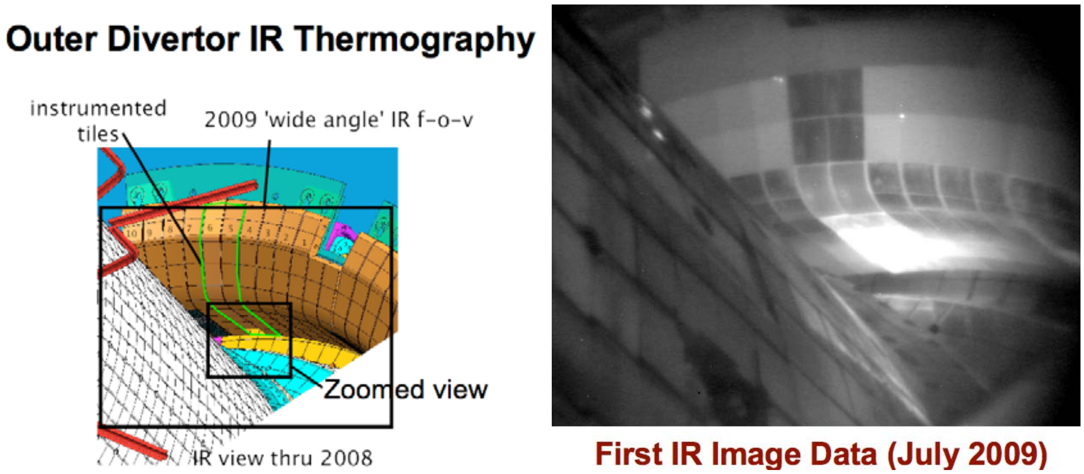

Fig. 4.1.6: A new IR camera (ElectroPhysics Titanium 550M, purchased by LANL) was installed to view the instrumented ramped tiles.

new divertor bolometer system was installed to assess radiative contributions to the divertor power balance.

IR thermography is particularly challenging in C-Mod with its shiny, low emissivity molybdenum tile surfaces, with changing coatings on these surfaces, with no normal-incidence light-of-sight viewing of the target surfaces, and with shallow field line angles onto the targets. It is important to note, however, that this is very similar to the situation faced when doing IR thermography on ITER. To improve the situation, the new set of instrumented tiles on the outer divertor were installed as two columns of ‘ramped tiles’, tilted in the toroidal direction by ~2 degrees relative to standard tiles. This ensures that the tile surfaces are not shadowed toroidally by misalignments in adjacent tiles. It also increases the thermal load to the local tile surface, improving signal-to-noise for IR and sensor-based diagnostics.

A new ElectroPhysics Titanium 550M camera was installed to view the J-port divertor module via the A-port top IR periscope. The resultant movies of IR emission allow the heat-flux footprint to be monitored as it evolves during a plasma. The camera can resolve ~1 mm scale features on the outer divertor surface. Since the camera views the tile sensors directly, in-situ cross calibrations can be performed on a shot-to-shot basis.

In conjunction with the hardware installations, a suite of support software was written to handle a range of issues associated with the new embedded sensors as well as the IR camera images, including image data archival, 3D image modeling, between-shot image stabilization and display and *in-situ* IR temperature calibration. Additional software for thermal heat transport modeling and for determining the heat flux density profile from the surface temperature measurements is presently under development.

4.2.1. Highlights: Hydrogenic Retention

Alcator C-Mod has carried out several sets of experiments to explore the issue of hydrogenic fuel retention. This is strongly motivated by the fact that C-Mod is the only divertor tokamak worldwide to use exclusively bulk refractory metals (molybdenum and tungsten) as opposed to coatings. These metals are expected to have more favorable (i.e. less) H retention than the more commonly used low-Z graphite plasma-facing components, and as a result there is a significant interest in providing experimental demonstration of low fuel retention for both ITER and future reactors.

Alcator was a major participant in the FY09 Joint Facility Milestone on H fuel retention and control. Over the last 7 years Alcator pioneered a very accurate global particle balance method which closed all external pumping during a discharge and allowed the released H gas from the walls to reach pressure equilibrium several minutes after the plasma termination [10]. This "static" particle balance provides very high accuracy, but lacks time resolution during the shot. It was found that in the absence of disruptions deuterium (D) fuel retention is constant from shot-to-shot, i.e. the Mo/W wall continues to absorb D fuel particles and not release them. The rate of retention scales ~ 1% of the total D ion fluence incident on the divertor and main wall plasma-facing components (PFCs). The major success of the Joint Facility work was to export and combine this technique with "dynamic" particle balance, i.e. a method that computes in real-time the in-vessel particle balance from the difference of particle input and exhaust rates. On C-Mod retention we were able to exploit the upper divertor cryopump design that allows for an accurate accounting of pumped particles after each shot (this method was adopted to DIII-D as well).

A surprising result is that in many discharges the real-time retention rate of D drops from large values in plasma rampup, to a constant and near zero value in the steady-state portion of the

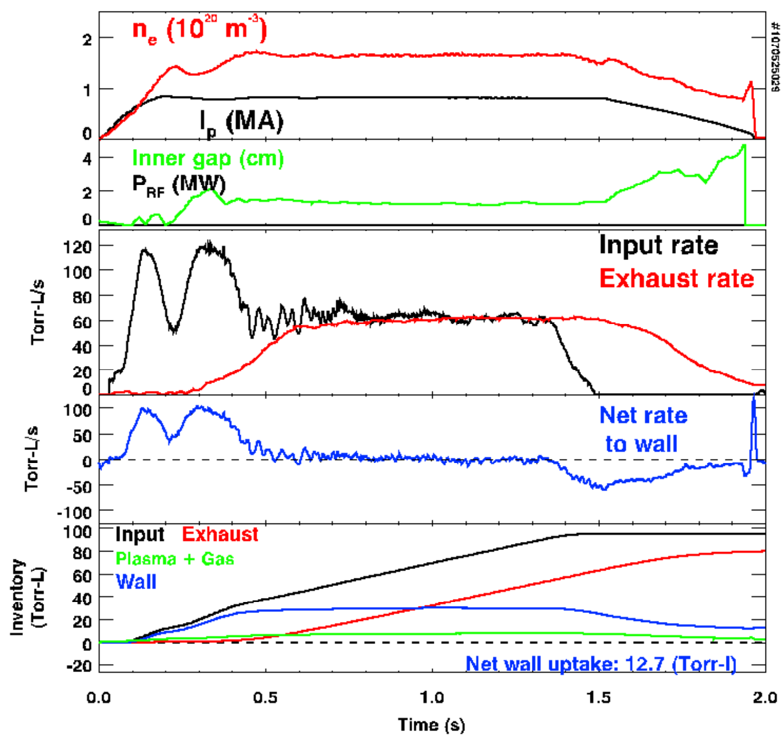

Fig. 4.2.1: Example of dynamic particle balance for a C-Mod shot with stationary balanced double-null plasma. The shot-averaged static particle retention is 12.7 Torr-L obtained from cryopump regeneration (bottom panel).

discharge (Fig. 4.2.1). This would seem to imply that the D uptake is dominated by the early phases of the discharge. However this is inconsistent with the previous 'static' particle balance experiments in that the total retention variations with strikepoint position, divertor film thickness, etc. Furthermore, a constant (or zero) fuel retention rate is surprising because the wall temperature is varying throughout the discharge. A further surprise is that the dynamic behavior is nearly identical to the results of DIII-D with a graphite wall - a material with completely different properties for retention and hydrogen diffusion. Several other puzzles have emerged: Plasmas can have near-identical core plasma parameters yet small changes in the magnetic topology (e.g. magnetic balance between upper and lower null) can affect the global static balance and the dynamic retention in the stationary part of the shot. For example the upper divertor (cryopoump) pressure, and thus pumping, changes with null balance, yet the global fueling requirements remain the same. Also unclear is the relative importance of transient effects in startup and shutdown. Dynamic balance would indicate these as dominant, yet the retention can be reproduced shot after shot (if no disruptions are present), contraindicating a "temporary" reservoir for particles. Furthermore, despite the very constant core plasma conditions the edge recycling evolves throughout the discharge (indicated by evolving exhaust rate).

A significant fraction of the effort in the areas of fuel retention and surface studies has been on

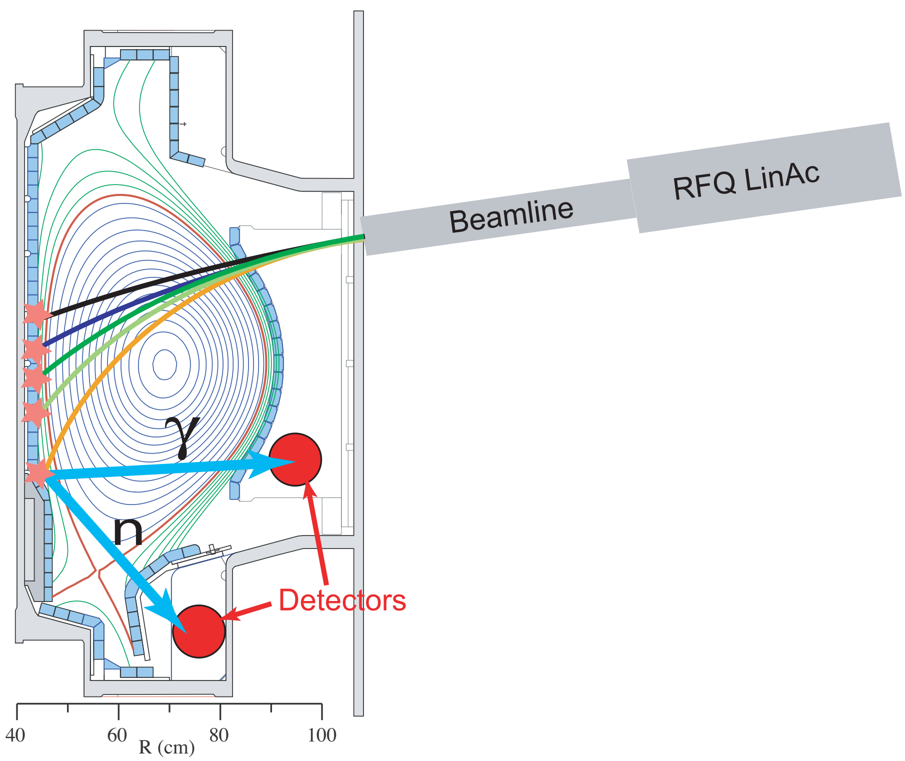

Fig. 4.2.2: The proposed position of the RFQ ion beam accelerator is shown. The beam is steered poloidally by varying the toroidal field and steered toroidally with the poloidal field. The nuclear reactions of the 1 MeV D ions with the D and impurities on/in the surfaces result in neutrons and gammas that will be detected by in-vessel diagnostics.

developing the new RFQ (Radio Frequency Quadrupole) diagnostic. This 1 MeV ion beam will be used for in-situ surface interrogation using ion beam analysis techniques (Fig. 4.2.2). There was an existing RFQ ion beam at MIT that is in the process of being refurbished this year. A remote, digital control is being added as well. In parallel beam dynamics simulations have continued with the goal of playing a role in the decision of which port the RFQ will be installed on as well as the required magnetic fields for steering it to individual tiles. As part of the general design effort synthetic diagnostics have been developed to simulate emission and detected spectra (inorganic scintillators for γs coming, for example, from D-B^{11} reactions). Additional simulations of neutron products from D-D fusion reactions (beam ions with D in the tiles) are being modeled for simulating deuterium depth measurements.

4.2.2. Highlights: ICRF Sheath Rectification

Alcator C-Mod relies primarily on Ion Cyclotron RF heating. Previous C-Mod experiments identified a critical issue for ICRF (also planned for ITER); namely the formation of enhanced voltage sheaths at plasma-facing surfaces. This process of "sheath rectification" produces a greatly enhanced DC potential drop between the plasma and surface. This is highly detrimental since it "turns on" sputtering of high-Z plasma facing materials such as Mo and W, leading to degradation of the core plasma performance through line radiation losses. To date the solution to this problem on C-Mod has been the application of boron films to cover up the Mo/W. However this is not a long-term solution for ITER or reactors. Eliminating or controlling ICRF sheath rectification appears to be critical for the successful use of high-Z refractory metals in tokamaks.

Solving the sheath rectification problem has faced the major problem of direct diagnosis of the plasma potential over a wide range of ICRF conditions and plasma positions. The emissive probe is a well-understood method to measure plasma potential, however it suffers from not being able to function in high-density plasmas of the SOL due to its current emission limits and from its short lifetime to filament burnout. We have recently developed a new robust diagnostic for plasma potential diagnosis based on the ion sensitive probe (ISP) using an RF laboratory plasma (to be presented at the Plasma Surface Interactions meeting in San Diego, May 2010). These ISP probes have only cold, solid components and are therefore as robust as a regular Langmuir probe. In 2009 a test stand was positioned on C-Mod limiter to test a prototype of the ISP against an emissive probe at the same location. The results of this test were positive (Fig. 4.2.3) with the ISP showing the same trends and potential values as the emissive probe. Simultaneously the local magnetic fluctuation levels were measured. Surprisingly, little to no correlation between magnetic fluctuations and enhanced sheath voltage was found. Rather a correlation between local plasma density and sheath voltage was found, in contradiction to present sheath rectification theories [11].

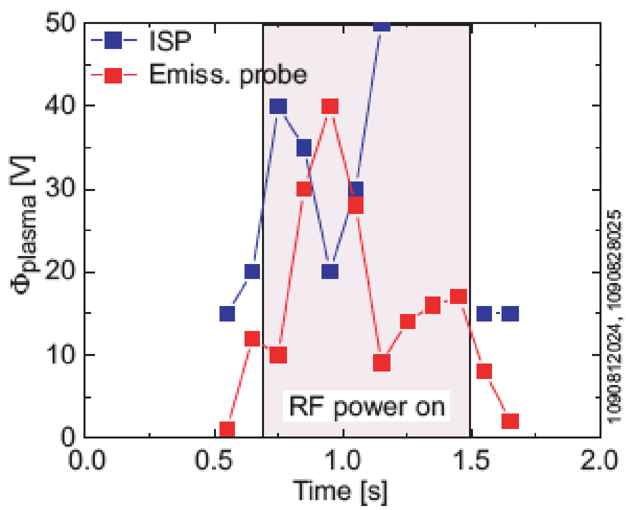

Fig. 4.2.3: Comparison of Ion Sensitive Probe (ISP) and emissive probe plasma potential in the C-Mod edge plasma.

4.2.3. Highlights: Material Migration

Last year we initiated a collaboration with J. Brooks of Purdue to model material migration in the outer divertor based on measurements made in C-Mod of molybdenum influx (gross), Mo erosion (net over the run period) and local plasma parameters (Langmuir probes). Erosion/redeposition of Mo for a set of 4 discharges were modeled by Brooks et al., corresponding to the typical plasma currents in that run period, and using detailed coupled codes for sputtering, erosion/redeposition, and related phenomena. Each discharge has an OH and RF heating period, and these were modeled in detail. Also included was the effect of a hypothetical RF induced sheath, as well as the "conventional" dual-structure sheath corresponding to C-MOD, near tangential magnetic field incidence.

The code/data comparison shows generally good agreement for gross Mo sputter erosion rates, but poor agreement for net erosion. This disagreement[11] points towards either some unknown physics e.g., carrying the eroded Mo away, or issues with the measurements. The study will be presented at the May 2010 PSI-19 meeting in San Diego.

In parallel with the modeling effort we also embarked on a new study of global migration: A poloidal set of tiles in the lower divertor were removed in 2008. 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 gives us the campaign-integrated migration of eroded W from the W tiles. The results, shown in Fig. 4.2.4, indicate that some fraction of the eroded W is immediately ionized near the divertor plate and then flows back down

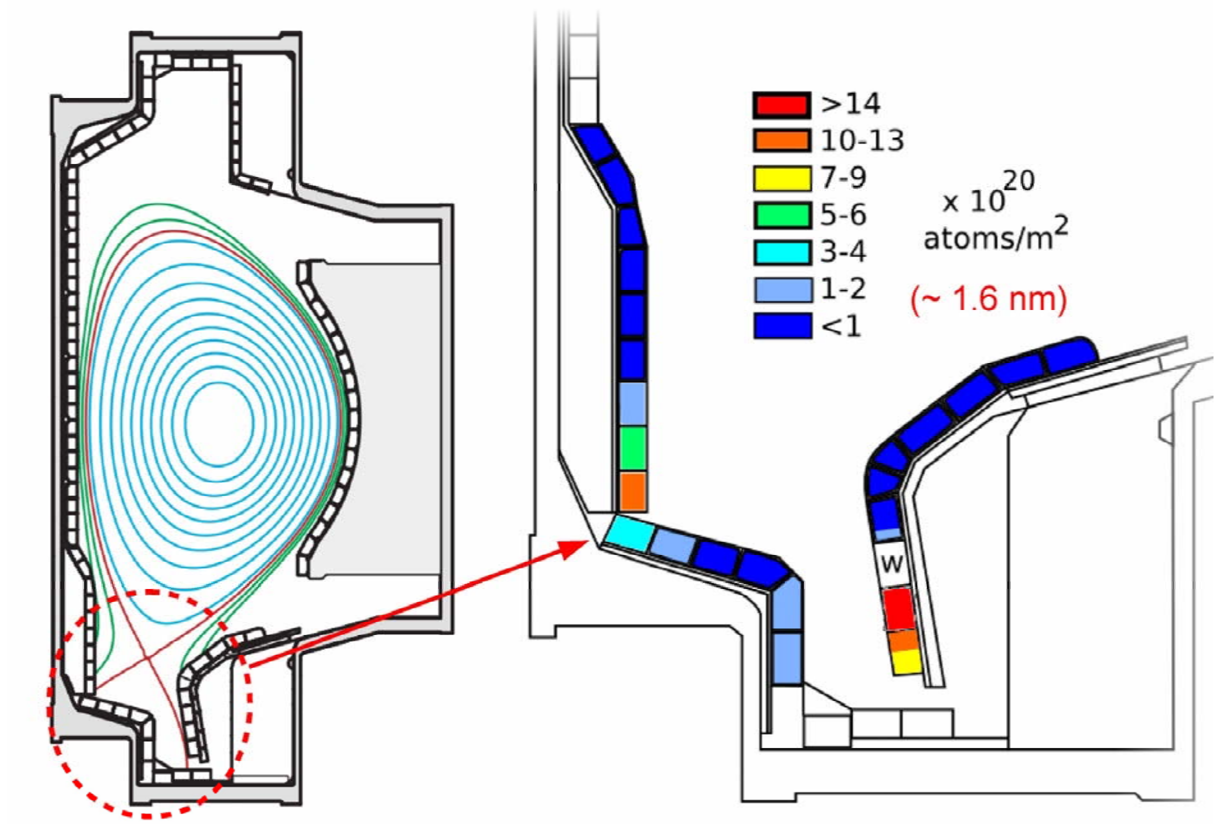

Fig. 4.2.4: A real density of tungsten atoms on the molybdenum tiles surrounding the toroidal row of tungsten tiles.

to the plate, the redeposition occurring closer to the strike point for each erosion/re-deposition cycle until the W is beyond the strike point, in the private flux region, where it is no longer eroded. A significant fraction of the eroded W that is ionized flows around the plasma to the inner divertor. As can be seen through examination of the figure a smaller fraction is not ionized at all after the erosion and is sprayed on the tiles between the divertor covering the equilibrium field coil (EF1).

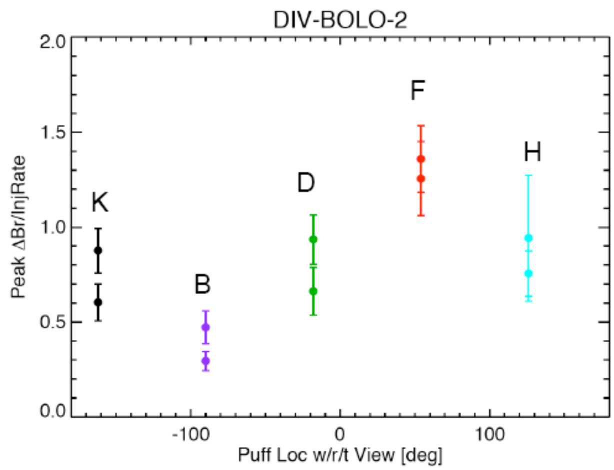

Fig. 4.2.5: Change in bolometer brightness, normalized to injection rate as the puff was moved around the machine. The bolometer was located at B port.

4.2.4. Highlights: Divertor Physics

During the past year Richard Pitts, of the IO (ITER Office), requested an experiment to test the toroidal uniformity of impurity gases when injected into the divertor. This is a time-sensitive issue for ITER given the present design of the divertor impurity injection system calls for equi-spaced valves and that coarseness of toroidal locations might lead to local power peaking on the divertor. C-Mod has 5 independent puff locations in the divertor and the experiment with N2 injection showed (see Fig. 4.2.5) that even for the toroidal circumference of C-Mod there were toroidal asymmetries.

4.3. Results: High Heat Flux Divertor

Early in FY10 we began engineering work on the new outer divertor for C-Mod (Fig. 4.2.6). The goals of this new divertor are multiple - 1) remove large leading edges (e.g. as currently at ports now for diagnostic access) in the high heat flux region to reduce heat loads; 2) Convert the entire high heat flux region to tungsten lamellae tiles; 3) Upgrade the divertor design to handle high temperatures (500-600C) that could be experienced in the long pulse Lower-Hybrid current-driven plasmas presently envisaged; and 4) design in the capability to hat the divertor to the same temperatures to allow the capability to study fuel retention as a function of the tile temperature. The initial work (also see operations section of this proposal) has centered on developing divertor plate support methods capable of withstanding the high temperatures while allowing the divertor plate to expand ~ 12 mm in diameter. To assist the engineering group we have studied the level and time dependence of currents induced in, and running through (halo currents), the divertor. That information is required for the engineering design.

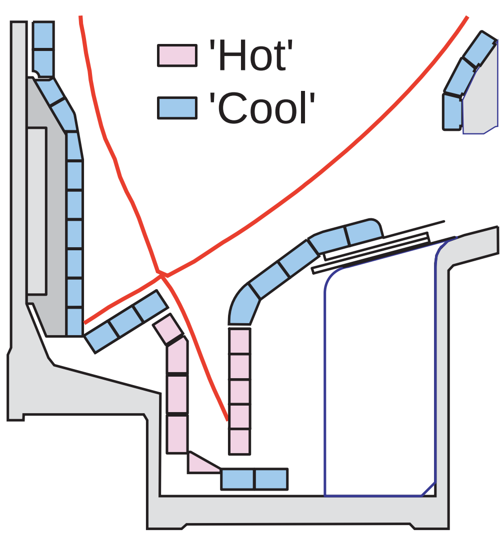

Fig. 4.2.6: Areas of the outer divertor that are designed to be heated.

We have also studied the range of equilibria in C-Mod with the goal to determine what shape a 'dome' between the inner and outer divertor could be to block the transfer of heat to the inner divertor through IR emission.

4.3. Edge Transport Plans for FY10 through FY12

Research in the area of *edge plasma turbulence and transport*, is organized into two sub-topical areas: (1) time-averaged measurements of plasma profiles, flows and particle/energy transport and (2) spatial/temporal measurements of the underlying turbulence. We seek to develop descriptions of boundary layer transport that can be scaled to reactor conditions. Empirical scalings are the first priority; physics-based scalings and understandings are the ultimate goals. The later requires a close-coupling among experiment, theory and numerical modeling.

4.3.1. Boundary Layer Heat Flux Experiments

Heat flux footprint experiments will remain the primary effort in the *edge plasma turbulence and transport* topical area for FY10, in support of the FY10 Joint Facilities Milestone. The research will explore divertor heat flux 'footprints' and its mapping to the heat flux channel widths at the outer midplane for a variety of plasma conditions. In addition, turbulence imaging diagnostics will be employed to follow systematic trends in the fluctuation spectra. The parameter space for H-mode discharges is indicated in Fig. 4.7, which highlights the exploration of 'similarity discharges' that match shape and dimensionless plasma physics parameters (ν^*, ρ^*, β) in the pedestal region of DIII-D [12]. DIII-D has already run a set of companion discharges for the comparison. In addition, the effect of magnetic connection length on power e-folding widths will be explored by comparing lower single null versus near double null discharges under otherwise identical conditions. This experiment will help differentiate among the roles of plasma current, safety factor and connection length in empirical scalings.

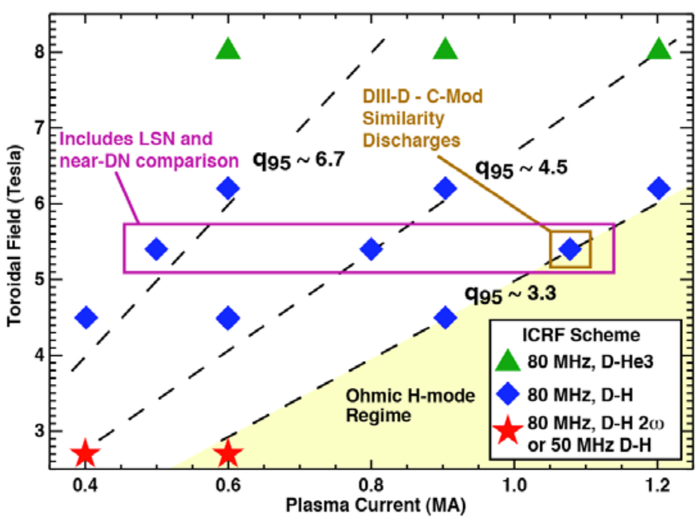

Fig. 4.3.1: Plasma current and toroidal field combinations to be investigated for boundary layer heat flux experiments. ICRF power scans would be performed at each combination. Plasma density will be chosen so as to produce stationary H-modes (EDA or ELMy).

An upgrade to the instrumented divertor tiles is planned for the April 2010 vacuum break. At that time, a fresh set of ramped tiles will be installed along with a new and improved set of embedded heat flux sensors. The ramp will also be extended by one tile column so that an array of

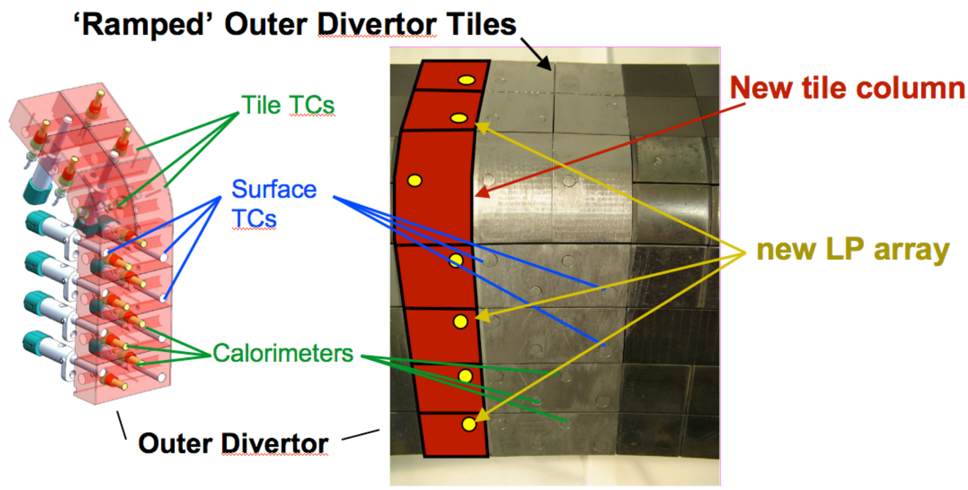

Fig. 4.3.2: Embedded sensors will be repaired and upgraded during a vacuum break scheduled for April 2010. An array of Langmuir probes will also be added on a separate column of tiles to allow a direct measurement of plasma conditions on the ramped tile surfaces.

embedded Langmuir probes can be installed. These will enable a direct measurement of the plasma conditions at the ramped tile locations.

Boundary layer heat transport experiments will continue into the FY11 run campaign and beyond as we refine the experimental measurements and broaden the scope of the heat flux investigations. Topics of interest include radiative and detached divertor regimes, lower-hybrid current drive discharges, ELMy H-modes and enhanced performance L-mode discharges, and continued study and analysis of the connection between SOL widths and turbulence.

4.3.2. Edge Turbulence Studies

With the increased capability for diagnosing edge and SOL turbulence, we will continue to connect the turbulence measurements with the time-averaged observations.

In low-field-side pedestals of EDA H-mode modes and improved L-modes (I-modes), the Quasi-Coherent Mode (in EDA) and a broader (in frequency) fluctuation feature (in I-mode) are observed. Continued characterization of these modes is planned. How they develop, and what their connection is to edge turbulence are questions that we will investigate.

As highlighted in section 4.2, edge turbulence dynamics are intimately connected to plasma density (or perhaps collisionality) and are thought to play an essential role in the physics that sets the robust Greenwald density limit. Since this clear dependence of edge turbulence on the Greenwald fraction is seen, we have a unique opportunity to explore important connections among the density limit, the steady-state profiles, and the turbulence at large Greenwald fraction.

We will also exploit our increased measurement capability to continue our characterization of the edge and SOL turbulence at different poloidal locations. We will do wavelet analysis on the images to follow the connection between edge turbulence and SOL blobs/filaments. To explore the connection further we hope to compare the observed scalings with a theoretical beta scaling of Krasheninnikov [13]for blob-generating turbulence inside separatrix. Using the fast cameras and GPI, we will analyze blob dynamics at both outboard midplane and X-point regions, and compare results with analytic blob models and SOLT and/or BOUT simulations and continuing the present studies of the effects of SOL turbulence on SOL profile widths.

4.3.3. Empirical Power E-folding Widths and 'Critical Gradient' Dynamics

Previous experiments in C-Mod have uncovered clear correlations between pressure gradients near the last-closed flux surface (LFCS), the plasma current in the discharge, the local collisionality and the magnitude of flow shear near the separatrix [7, 14]. These observations point to a 'critical gradient' behavior near the LCFS, i.e pressure gradients are 'clamped' at a ~fixed value of α_{MHD}, dependent on collisionality (principally), but also affected by flow shear. We wish to explore if boundary layer heat flux widths and their observed scalings can also be understood in terms of this physics. This topic will come into focus in the FY11 time scale, and will overlap nicely with C-Mod's participation in a planned joint facilities investigation of the H-mode pedestal region. In particular, we wish to map out separately the scalings of density gradient and electron temperature gradient (which is thought to set the power e-folding length) and their connections to the observed power e-folding widths and the 'critical pressure gradient' behaviors near the LCFS. An important related question is how the pressure gradients of the LCFS region (α_{MHD} ~ 1) smoothly connect to the pressure gradients further up the H-mode pedestal (α_{MHD} ~ 8).

4.3.4. Role of Boundary Layer Plasma Flows, Flow Shear on 'Critical Gradients'

As mentioned above, we have found experimental evidence that plasma flow shear plays an important role in setting the 'critical gradients' that are observed in the plasma boundary, perhaps even dominating over magnetic shear in some cases [3]. C-Mod's complement of scanning Langmuir-Mach probe diagnostics combined with gas-puff turbulence imaging (e.g. Fig. 4.1.2) is well-suited to study plasma flow shear in the boundary layer plasma. Simulations of the C-Mod edge turbulence performed by M. Umansky using the BOUT turbulence code [15] have reproduced many interesting features that we see with our diagnostics: wave-like electron-diamagnetic propagation feature in the closed field line regions, which grows in amplitude and intermittently spawns blob-like propagation events in the open field line regions. These simulations are particularly telling because they are done with the time-averaged radial electric field forced to zero; ExB therefore does not play a role in the simulated poloidal phase velocities, exposing the electron-diamagnetic source. We will continue to work closely with M. Umansky to extend these simulations to explore the role that E_r and E_r shear has in the turbulence dynamics, including the influence on macroscopic features such as radial correlation lengths, transport levels and poloidal phase propagation. In particular, new BOUT simulations will be constrained to have time-averaged radial electric fields that correspond to the values measured from our array of scanning Langmuir-Mach probes. From recent work [5], we now have a better idea on the reliability of inferring E_r from probes and can therefore specify target E_r profiles with a level of confidence. To further constrain the problem, and to attempt to validate the turbulence simulation, we also have measurements of poloidal phase velocity profiles of density fluctuations

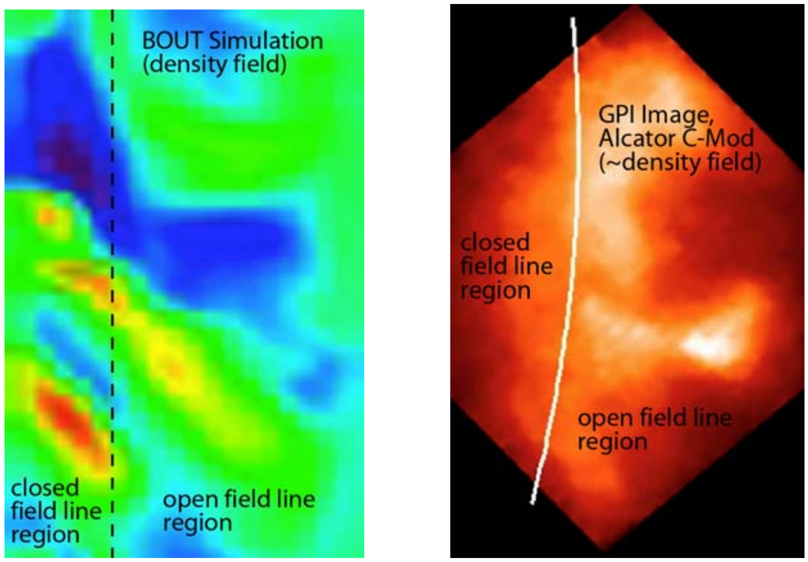

Fig. 4.3.3: Plasma response changes from an electron diamagnetic drift-like behavior inside closed field line regions to an intermittent, 'blob'-like propagation behavior on open field lines, as simulated by the BOUT transport code [15] (left) and measured by gas-puff turbulence imaging (right).

at the three scanning probe locations and also from the turbulence imaging diagnostics. These exhibit a clear 'shear layer' near the LCFS, with shearing rates (i.e., gradients in ExB, Vphase) are comparable to the ideal ballooning growth rate – a strong hint that flow-shear is a player in the turbulence dynamics. A direct comparison of phase velocities between simulation and measurement will be an important test moving forward.

4.3.5. Further Validation Efforts of Boundary Layer Turbulence Models

Finally, we aim to further elucidate the role of electromagnetic effects in the plasma turbulence by direct measurement. Using a novel scanning probe head with embedded coils, we will build upon our database of poloidal magnetic field fluctuations as a function of distance into the SOL. These will be compared with output from first-principles numerical computations (e.g., BOUT). In addition, with the help of a new 'Mirror Langmuir Probe' technique developed at MIT [16, 17], there now exists the possibility of simultaneously measuring plasma density, electron temperature and plasma potential fluctuations, their k-spectra, and their correlations as a function of distance into the SOL using C-Mod's standard 4-electrode scanning probes. If ballooning mode physics (resistive or ideal) does indeed play a prominent role in the near SOL, as suggested by the pressure gradient scaling with α_{MHD}, then the local phase delay between pressure and potential fluctuations should tend towards ~π/2 in this zone, instead of ~0 for drift-wave turbulence [18]. In any case, we will test the viability of this diagnostic technique and develop it further, if necessary, to more generally interrogate plasma density, electron temperature and plasma potential fluctuations in the C-Mod SOL. This will provide valuable information on the dynamics of blob and ELM transport physics.

4.3.6. Upgrades to Turbulence Imaging

Depending upon allocation of manpower resources, as part of our collaboration with PPPL, we plan to develop a more collimated gas-jet source and improved optics for GPI to increase spatial resolution of the fast cameras from ~2 mm to better than 1 mm. The collimated gas jet will probably require a test stand at PPPL or MIT.

4.4. Future Plasma Surface Interaction Plans

4.4.1. Hydrogenic Retention

Further experiments on hydrogenic fuel retention will try to address the present knowledge gaps and inconsistencies in our understanding of the controlling processes underlying fuel retention in Alcator C-Mod's metallic walls. These experiments will build on the extensive diagnostic tool set which has already been developed on Alcator. The recent improvements of PFC temperature will also be critical, since surface temperature evolution is perhaps the most important parameter controlling H fuel retention.

Perhaps the most surprising result to emerge from the C-Mod experiments is that under stationary core plasma conditions, the edge recycling (gas pressure, ion flux) continue to evolve. This is contrary to the typical view of the boundary plasma where the plasma conditions are set by the dual requirements of steady-state heat and particle exhaust from the core. This issue is critical to the fuel retention issue, since it is in the end the physics of the recycling requirements locally at the material surfaces that sets the particle retention or release.

A linked question is whether strong divertor cyropumping in C-Mod is itself affecting the boundary plasma conditions and thereby forming a self-regulated particle fuelling control loop with the core plasma (Fig. 1). Previous C-Mod studies discovered main-chamber recycling, which overall indicated that the fuel recycling picture is more complicated than the standard picture of divertor recycling and neutral leakage to the core. It would seem that the link between core plasma requirements and edge plasma parameters is also more complicated than believed. C-Mod is an ideal experiment in which to carry out experiments in this area due to its highly opaque SOL and lack of particle fueling with heating neutral beams.

In this area, experiments will focus on careful parameter scans with simultaneous full diagnosis of dynamic and static particle balance. In addition, new diagnostics (RFQ in-sit D measurements, divertor tile temperatures) will be exploited to better diagnose the material properties throughout the shot and after. The parameter scans will particularly take advantage of being able to modify the relative amounts of divertor recycling to pumping by changing magnetic topology from upper null to lower null. In this way the governing factors (heat flux, D thermal release, etc.) can be extracted to better understand what is controlling the fuelling and recycling requirements.

In addition, the new temperature diagnostics will be used to more directly correlate fuel release following disruptions, with the actual surface temperature evolution. These data can be more directly compared to numerical modeling of the H retention previously developed.

During FY11 the upgrade/refurbishment of the RFQ will be completed and the port selected. The RFQ along with its control electronics and software will be fully developed offline along with the data acquisition equipment/method. The simulations will continue of both the synthetic diagnostics (gammas and neutron measurements) that will allow determination of the minimum measurement capabilities. In FY12 the RFQ is scheduled to be installed on C-Mod and operated.

4.4.2. Sheath Rectification

Given the successful testing of the ISP probe, we will deploy these into more locations on Alcator to better diagnose the sheath rectification problem.

4.4.3. Material Migration

We plan to continue the collaboration with J. Brooks of Purdue in the upcoming two year period, using new upgraded diagnostic measurements of C-MOD divertor erosion. Brooks et al. will continue numerical simulations of sputter erosion, transport and redeposition. We (Purdue, MIT) also plan to assess models/code upgrades, in particular, for mixed-material surface evolution (e.g., B/Mo surfaces), and RF sheath phenomena.

During the 2009-2010 run period we found that one of the tungsten tiles had loosened leading to a major leading edge and melting. During the remainder of this campaign the plan is to place the strike point in the area of the missing tile and determine, through slowly increasing the RF power, whether the W impurity ions reaching the plasma are due to evaporation of atoms or due to droplets of W - the signature of the former would be for the W level in the core to be exponentially increasing in time corresponding to the exponentially increasing evaporation rate (if we can slow it down through judicious use of RF power). At the end of this campaign, near the end of FY10, we will enter the machine and determine whether a toroidal and poloidal set of tiles surrounding the melted W tile can be removed and studied for the migration of W. If the tiles are removed the analysis work will be performed in FY11 and FY12.

Prior to the next run campaign, which will be primarily in FY11, our intention is to remove 2 tiles form the divertor floor where the strike point is rarely located. The missing tiles will be located at sections of the divertor where there are IR temperature and spectroscopic views, the latter to track the impurity influx. The missing tiles will allow us to run well characterized melting tile experiments to study the source, surface temperatures and the penetration of the impurity to the core plasma.

In FY12 we plan to the have the new RFQ diagnostic on C-Mod. This will allow discharge to discharge measurements of the impurities on surfaces. This will be a quantum increase in capability to follow material migration from and to surfaces. Ideally the RFQ measurement will be made at surfaces that are also monitored for impurity influxes during discharges such that we can track the gross erosion of materials and compare that to the net change in surface impurities from the RFQ.

4.4.4. High Heat Flux Divertor

The conceptual design review is slated to occur in early 2010. After that we in the boundary group will continue to work with the engineering group to find solutions that lead to a new divertor to be installed in 2012. That will include work specifying the forces during disruptions (including halo currents), plasma heat footprints on the divertor, and temperature ranges. Most of that work will peak in FY10 and FY11.

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5. ICRF

C-Mod provides a unique opportunity to explore wave propagation, absorption, and mode conversion physics in the ion cyclotron range of frequencies. 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. In core ICRF physics, simulation validation of wave propagation and absorption, characterization of flow and current drive, and RF and plasma physics integration are areas of emphasis. In many present, 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 to potential improvements: coupling physics; compatibility with high performance discharges and metallic plasma facing components; maintenance of coupled power despite load variations; and availability to deliver ICRF power on demand without burdensome antenna conditioning.

5.1. Highlights

In an effort to better understand mode conversion flow, extensive parametric scans were performed. In Figure 5.1, the change in central rotation velocity (flow) scaling with RF power, plasma temperature, plasma current and plasma density is shown. The toroidal flow velocity is derived from the Doppler shift of the x-ray spectra of Ar impurity measured by high resolution x-ray spectroscopy (HIREX). We found that the velocity change, ΔV_ϕ, scales approximately linearly with injected power, indicating a mechanism dependent on power dissipation rather than one directly related to the RF field strength. The ΔV_ϕ scales roughly as $T_e^{3/2}/n_e$, suggestive that wave momentum is playing a role in the generation of flow (momentum is randomized on a collision time scale). In contrast to intrinsic rotation, the ΔV_ϕ increases with plasma current. The ΔV_ϕ dependence on antenna phasing appears to be complicated. At low power (< 2 MW), the magnitude of ΔV_ϕ is approximately constant for co-current, heating and counter-current wave injection phases. At high power, the ΔV_ϕ for counter-current wave injection saturates, while for heating and co-current wave injection, it continues to increase. Additional experiments also showed that the peak in the flow drive occurs at moderate ^{3}He levels, 8-12%, and is largest when the cyclotron resonance and mode conversion layer are near the plasma center. The dependence on ^{3}He fraction suggests the interaction of the mode converted wave with ions is important for flow drive. Experimentally, no flow drive has been observed when a majority of ICW power is absorbed by electrons, and TORIC indicates significant power to the ions when rotation is observed.

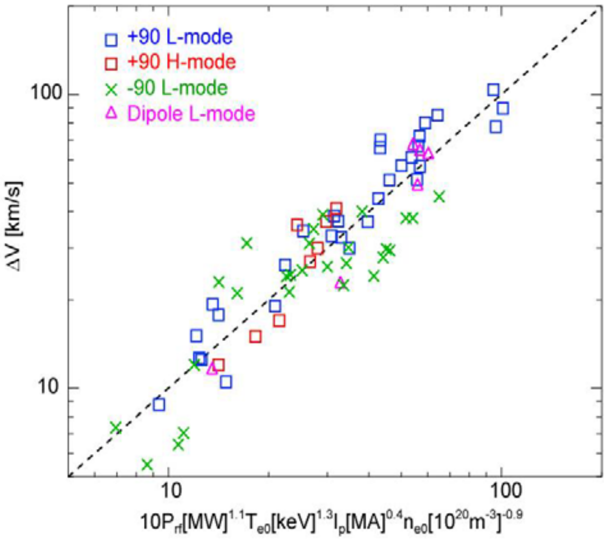

Fig. 5.1: Mode conversion flow drive empirical scaling on RF power, plasma temperature and density, and plasma current. The scaling is notably different from intrinsic scaling which scales inversely with plasma current.

As part of ITPA TC-14, experiments were conducted on JET to investigate mode conversion flow drive. Flow drive was observed; however, the flow was in the counter current direction. Further analysis is required to understand the origin of the sign change in comparison with the C-Mod results.

To inject ICRF power, antennas are situated near the plasma edge, and one of the primary ICRF utilization challenges is to reduce/eliminate specific ICRF impurity production. A previous prescription to ameliorate impurity production was developed for experiments with carbon plasma facing components (PFCs).[16] 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. To identify and ameliorate ICRF-related impurity sources, the molybdenum tiles on the outer divertor shelf, plasma limiters, and RF limiters on C-Mod were vacuum plasma coated with 100 μm of boron, where the B density was ~50% of nominal density, see Figure 5.2. In contrast to operation with bare molybdenum tiles, the core molybdenum brightness no longer scales with RF power and is controlled for a significantly increased number of RF joules following a boronization, as shown in figure 5.3. The combination of low Z impurity seeding and boron coating enabled high power operation without significant impurity influx and with near fault-free operation. This result was somewhat surprising since one might expect sputtering to increase with light impurities present. The improved antenna operation suggests that localized heating is contributing to antenna faulting.

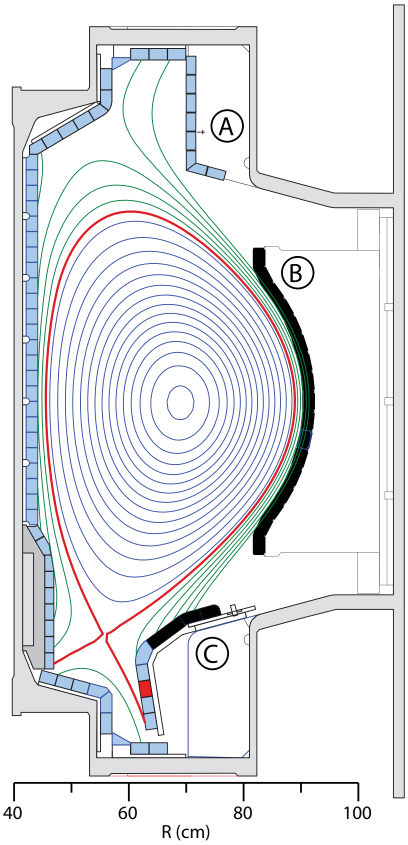

Fig. 5.2: Poloidal schematic showing the location of the boron coated tiles.

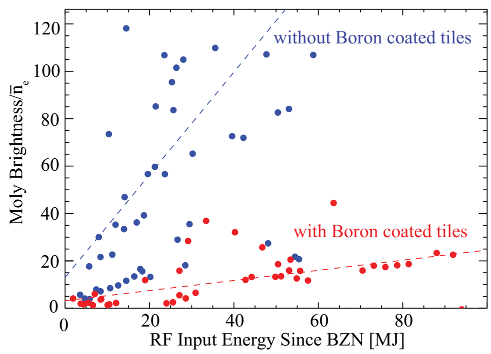

Fig. 5.3: Molybdenum brightness normalized to the plasma density as function of injected RF Joules since last boronization.

5.2. Plans

5.2.1. 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 upgraded ICRF simulation capability that includes non-Maxwellian ions, AORSA-CQL3D. A newly implemented 8 channel Compact Neutral Particle Analyzer (CNPA) and the original 3 channel CNPA are the primary diagnostics to measure this fast ion distribution function, and a fast ion charge exchange (FICX) will be available, through collaboration with the University of Texas FRC, to measure the fast ion spatial distribution. Experiments where the minority tail energy and spatial distribution are varied, by injected power for example, combined with plasma parameter scans, will provide experimental benchmarks for the simulations. In particular, the impact that finite banana width and ion minority tail energy have on absorption will be assessed. This work will be done in conjunction with RF-SciDAC

In ICRF mode conversion, the long wave length fast wave mode converts to two short wavelength modes, ion Bernstein and ion cyclotron waves. Using the Phase Contrast Imaging (PCI) diagnostic (DoE Diagnostic Initiative), the fluctuation profile of the mode converted waves can be measured. Since the original PCI implementation[1], the system has been upgraded to measure higher wavenumber (*k*) and its calibration improved. Furthermore, the synthetic PCI diagnostic in TORIC has been upgraded and compared with one implemented in AORSA in collaboration with the RF SciDAC Group. An outstanding issue regarding amplitude discrepancies can potentially be resolved because of these diagnostic and code improvements. Including toroidal effects, the remaining differences between the simulation and experiment may be resolved. These studies will initially focus on $D(^3He)$ 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 as a function of minority concentration and compare with the simulations. This is enabled through collaboration with the RF Sci-DAC Initiative, and enhanced by local access to the MARSHALL and LOKI Beowulf clusters to perform full-wave simulations routinely.

Realizing high heating efficiencies in $D(^3He)$ 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. In recent 8 T experiments, significant heating effectiveness has been observed at low density and high power, with central T_e reaching 9 keV. This may result from a positive feedback loop where, as the plasma temperature increases, the drag on the minority tail decreases. This results in an increase in tail energy and increases the single pass absorption. Additional high power experiments are planned to investigate the possibility of bootstrapping to higher single pass absorption. We also plan to continue experiments where a direct comparison of D(H) and $D(^3He)$ can be done in the same discharge using the D and E antennas at 80 MHz and J antenna at 50 MHz. 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 planned utilization in ITER and theoretical calculations which suggest that it may have stronger damping than previously thought. With the compact neutral particle analyzer[2], 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 C-Mod 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, which is predicted by the codes and could have significant effects on impurity generation.

5.2.2. Mode Converted Flow Drive (MC)

Rotation control offers an opportunity to increase plasma stability and improve plasma confinement; thus, development of an ICRF flow actuator is a high priority. Since the initial observations of driven toroidal flow[3], we have characterized flow drive over a large plasma and RF parameter space and we will seek to expand this parameter space and optimize the flow drive. We will investigate flow drive in low density H-modes and He and H majority discharges (proposed ITER non-activation phase discharges). We can also utilize a narrow magnetic field scan to vary the power partition between IBW and ICW waves and utilize 40 MHz at 4 T to characterize flow drive with lower frequency and higher momentum content. In addition we will also investigate flow drive at 60 MHz and 70 MHz to determine the scaling on wave momentum. 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. Another test is to investigate off-axis flow drive in an effort to evaluate the flow drive efficiency and the ability to modify the rotation profile. 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.

5.2.3. ICRF Sawtooth Destabilization/Stabilization

In ITER, the expected monotonic q-profiles of the baseline ELMy H-mode plasmas have low shear, s_1, at the q=1 radius, and the q=1 radius, r_1, is relatively large. These plasmas are expected to be unstable to the internal kink mode. Furthermore, the energetic trapped fusion-born α-particles are predicted to lead to significant stabilization of the internal kink mode [4,5], resulting in very long sawtooth periods. However, such long sawtooth periods have been observed to result in triggering NTMs at lower plasma β [6,7], which in turn can significantly degrade plasma confinement. For C-Mod, large sawteeth, partially stabilized by fast ions produced by ICRF heating, are realized under conditions with high confinement and high ICRF power. The associated sawtooth crash can terminate the high performance phase of a discharge.

A novel mechanism proposed for sawtooth pacing is based on kinetic effects resulting from an asymmetry in passing fast ions near the q=1 surface. [8,9] For highly energetic ions, the radial drift motion becomes comparable to the radial extent of the kink mode. In this regime, the kinetic contribution to the mode's potential energy (together with a non-convective contribution to the fluid part of δW) becomes increasingly important. When the passing fast ion population is asymmetric in velocity space, finite orbit contributions to the mode stability can be either stabilizing or destabilizing, and the effect is enhanced for large effective orbit widths, which is to say, for highly energetic ions (like ICRH). Passing fast ions can destabilize the internal kink mode when they are co-passing and the fast ion distribution has a positive gradient across q = 1, or when they are counter-passing, but the deposition is peaked outside the q = 1 surface. The effect of passing fast ions has been confirmed in NBI experiments in JET [10,11] and ASDEX Upgrade [12] and using ^{3}He minority ICRH in JET [13]. In support of ITPA WG3, we plan to investigate sawtooth stabilization/destabilization via this novel fast ion kinetic mechanism. In C-Mod, we can utilize one RF system to create low frequncy sawteeth via central heating, and then scan the other system's resonance across the q=1 surface while monitoring the sawtooth period. The primary knobs for varying the

ICRF pacing mechanism are the resonance position (high/low field side, inside/outside q=1), antenna phasing, RF power, minority concentration and plasma density.

5.2.4. 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 a 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 whereby the local current profile is modified to destabilize the sawteeth.[14] 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 can 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 alternately be shortened or lengthened by changing the antenna phasing or deposition location. A principal 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 (reduce) the sawtooth period. These experiments will be performed within the framework of the ITPA joint experiment MDC-5.

For a reference C-Mod advanced tokamak discharge, a central seed current of approximately 20 kA is required.[15] 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 important additional 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 impact on instabilities, like sawteeth. Experimental time devoted to this current drive mechanism is resource dependent.

5.2.5. Antenna coupling and Antenna/Plasma interactions

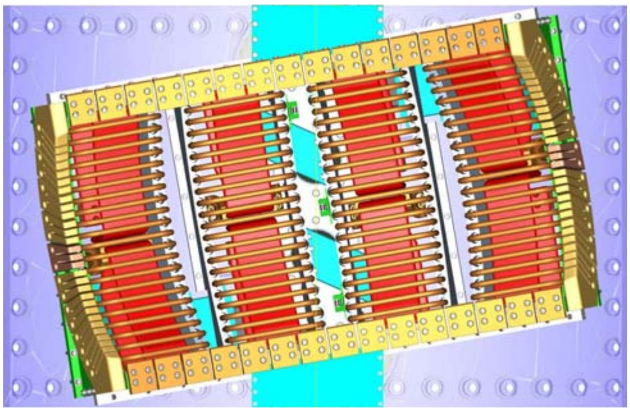

Fig. 5.4: Rotated antenna at 10 to reduce integrated parallel electric field with the aim to reduce the RF enhanced sheaths.

A number of physics and technological issues relevant to ITER and future devices can be addressed at C-Mod: coupling physics; compatibility with high performance discharges and metallic plasma facing components; reliable maintenance of 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 in excess of 10 MW/m^2 has become routine. We plan to install a new four strap antenna, shown in figure 5.4, to replace the J antenna that is designed to reduce RF related impurity production. If successful a second antenna will be built to replace the D and E antennas. To reduce impurities, we would like to reduce the magnitude of the parallel electric field, $E_{\parallel}$, and reduce the integrated $E_{\parallel}$ along a field line. The latter is achieved by rotating the antenna to follow a field line and by using symmetry to reduce the integrated $E_{\parallel}$. The experimental field line angle has a range from 7-13 degrees and we choose 10 degrees for the design. Simulations show that the integrated $E_{\parallel}$ is dependent on antenna phase; for [0,pi,pi,0] (heating) phasing, it is reduced to near zero. To reduce the magnitude of local $E_{\parallel}$, heating phase also results in the lowest local $E_{\parallel}$ magnitude. A further reduction in $E_{\parallel}$ magnitude can be obtained through reducing the current (and power) for the end straps and increasing it for the middle strap pair, and we plan to investigate this approach.

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, including ITER. Recent ASDEX-U ICRF operational experience with tungsten plasma facing components has been largely negative and resulted in modification of the ITER ICRF antenna design to avoid use of tungsten near or on the antenna. In C-Mod, we have successfully operated with molybdenum RF limiter tiles following boronization and the boron coated tiles have decreased the core molybdenum concentration. We have also re-instrumented the plasma limiter with plasma potential and RF probes to characterize RF sheaths. In addition, a scanning probe with plasma potential probes has been fabricated and will allow investigation of the plasma potential profile. 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 100 μm of boron on the outer divertor shelf, plasma limiter, and RF limiters. After the campaign we will inspect and assess the boron coating, and thus should be able to identify the important areas of erosion and impurity production. 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 of a new prototypical ITER ICRF antenna reflectometer (in collaboration with ORNL), we can measure the local density to determine the extent of the up-down density asymmetry near the antenna and infer the strength of convective cells through modeling. We will continue to develop techniques to improve the boronization lifetime or reduce the impurity influx. Although unlikely to be applicable in a reactor, vacuum plasma sprayed low Z-coatings are one possibility that can be explored. Another approach 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 and antenna loading in C-Mod are relatively robust, the development of tools and techniques to manage or control coupling could greatly improve antenna performance in future experiments. The new scrape-off layer (SOL) reflectometer will allow direct monitoring of SOL density profiles. 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 position in parameter space with a SOL that 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 F2008/9 campaign, we successfully deployed a fast matching system into the E antenna matching network, injecting up to 1.85 MW. This 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. An order for additional systems, made possible by incremental ARRA funding, has been placed, and we will begin implementing first on the D antenna. 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 which had been lacking in previous comparisons. We also plan to simulate fast changes in loading to understand the antenna behavior during confinement changes associated with H-mode transitions and ELM’s.

In future fusion devices 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 neutral pressure limit for RF breakdown is significantly lowered when an externally applied magnetic field is parallel to the RF electric field. An outstanding question for this neutral pressure limit was the role of the initial fault. We have recently performed experiments where the first fault was identified with a sudden change in reflected power. The source of the impedance change is unknown. With our upgraded test facility, we should be able to replicate the neutral pressure limit condition and we should also be able to characterize the changes to the material surface as the structure is conditioned.

5.3. Connections to ReNEW:

Thrust 4: Qualify operational scenarios and supporting physics basis for ITER

- ICRF compatibility (impurity production) with high performance plasmas.
- Flow drive (MC Flow Drive).
- Antenna operation: load tolerance and voltage/power handling.
- Sawtooth destabilization/stabilization with RF.

Thrust 5: Expand limits for controlling and sustaining fusion plasmas

Demonstrate integrated ICRF and LHRF.

Develop and test temperature, current, density, and rotation velocity profile control methods in DEMO-relevant conditions.

Thrust 6: Develop predictive models

We have a strong emphasis on experimental validation of simulations.

Thrust 10: Science and technology of plasma-surface interactions

Evaluate refractory metal RF armor and antenna.

Support for ITER and Connection to ITPA Activities

Although no dedicated ITPA organization exists to address heating and current drive, C-Mod contributes to multiple joint experiments utilizing RF for localized tailoring of temperature, momentum, or current profiles.

5.4. ITPA-ITER High priority Research Tasks

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

5.5. 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.
ITPA WG3: Assess the power requirements for ICRH and ECCD for control of sawteeth in ITER.

5.6. References

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6. Lower Hybrid Range of Frequencies

The main motivations for Lower Hybrid 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 and investigate steady-state and/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 drive to ITER's HCD portfolio. In this regard it is worth noting that the operational parameters of the C-Mod LHCD system (1) (f_0 = 4.6 GHz, $n_{||} \approx 2$, $B_\varphi \approx 5.5$ T, $n_{e0} \approx 10^{20}$ m^{-3}) are similar to the proposed LHCD system on ITER.

Lower Hybrid current drive research on Alcator C-Mod is highly relevant to the Research Needs for Themes I and II and Thrusts 4 and 5 described in the ReNew Report. In particular, the research plan directly addresses the development of LHCD-assisted hybrid and steady-state scenarios for ITER. The goal is to provide the necessary physics understanding and simulation capability to reliably predict the integrated performance of an LHCD system, not only for ITER, but also for the steps needed beyond ITER for developing practical fusion energy.

Lower Hybrid Experiments with up to 1.2 MW of absorbed RF power were stopped at the end of the 2008 campaign in order to concentrate effort on the design and fabrication of a new coupler, referred to as Lower Hybrid Coupler II (LH-II). As reported last year, the experiments with Coupler I were highly successful with up to 1 MA of current driven by the LH waves. MSE-measured current profiles were substantially broader than those resulting from inductive drive, with indications of possible shear reversal and suppression of sawteeth. These results agreed at least qualitatively with expectations based on extensive modeling carried out by the wave-tracing code GENRAY together with the Fokker-Planck code CQL3D, although the comparison was made difficult by the fact that the driven current was inadequate to fully eliminate the toroidal electric field.

Although there have not been new LH experiments during the past year, considerable progress has been made in four highlighted areas: 1) analysis of data taken during the 2008 campaign, including the elaboration of a so-called “density limit” above which current drive efficiency decreases rapidly, falling well below the level prediction by the conventional GENRAY-CQL3D simulation; 2) development of a new simulation tool based on Finite Element Modeling (FEM) of the propagation of LH waves essentially from the source to the region of absorption in the plasma; 3) upgrade and development of new diagnostics, which will be useful in challenging the results of LH wave simulations; and 4) design and fabrication of a novel power splitter and grill, the so-called LH-II, which we expect will have superior power handling capability and performance relative to the splitter and grill used in our LH experiments to date. These results are briefly summarized in the following sections, followed by a section on the research plans for 2011 and 2012.

6.1. Density Limit for LHCD on C-Mod

Lower Hybrid Current Drive (LHCD) is an attractive option for non-inductive tokamak operation due to its high current drive efficiency and ability to drive current off axis. The operational parameters of the C-Mod LHCD system (1) ($f_0 = 4.6$ GHz, $n_{||} \approx 2$, $B_\varphi \approx 5.5$ T, $n_{e0} \approx 10^{20}$ m^{-3}) are similar to the proposed LHCD system on ITER. Based on results from previous LHCD experiments on other tokamaks, LHCD is expected to drive significant current with C-Mod parameters ($\omega/\omega_{lh} > 2$, $n_{||} > n_{||crit}$) (2) (3). However recent experiments on C-Mod have shown that the population of fast electrons generated by LHCD, as measured by fast electron bremsstrahlung (4), is substantially reduced in L-mode at line averaged electron densities above 10^{20} m^{-3} for 5.4 T $< B_\varphi <$ 7 T and $n_{||} \approx 2$ (see Fig. 6.1). H-mode discharges with LHCD exhibit enhanced fast electron signatures as compared to L-modes at the same line averaged density. In addition, LHCD causes substantial changes in the character of the H-mode temperature and density pedestals, thereby raising T_e while lowering n_e. This lower density may have critical implications for the use of LHCD on future high density non-inductive tokamaks.

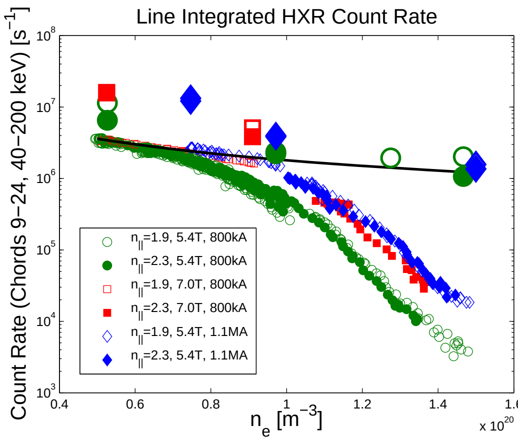

Fig. 6.1: Bremsstrahlung from fast electrons measured by the Hard X-Ray (HXR) diagnostic as a function of line averaged density. Experimental data (small symbols) deviates from the $1/n_e$ curve (black line) above 0.8-1.0x10^{20} m^{-3}. Data from the GENRAY/CQL3D synthetic HXR diagnostic (large symbols) roughly follows the $1/n_e$ curve.

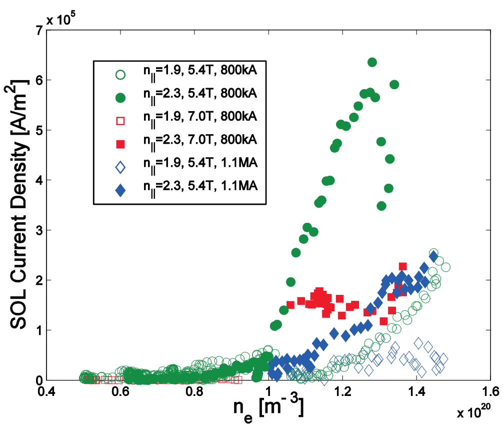

Fig.6. 2: SOL current density measured on divertor Langmuir probes during LHCD as a function of line averaged density.

The data from C-Mod show that discharges with higher plasma current and magnetic field exhibit somewhat stronger fast electron bremsstrahlung emission at the same density as compared to lower current and magnetic field. Simulations using the GENRAY/CQL3D ray tracing/Fokker-Planck package (5) (6) predict a decline in fast electron bremsstrahlung similar to $1/n_e$, in line with simple theoretical estimates. Experimental results deviate significantly from these predictions above 10^{20} m^{-3}, with a discrepancy of two orders of magnitude at 1.5x10^{20} m^{-3}, as shown in Fig. 6.1. Wave accessibility and parametric decay into ion-cyclotron quasi-modes do not explain the lack of fast electron generation at high density. Electric currents flowing between the inner and outer divertors in the scrape off layer (SOL) increase substantially as signs of fast electrons in the core plasma decrease (see Fig. 6.2), suggesting that absorption of the lower hybrid waves moves outside the separatrix at high density. The SOL currents during

LHCD flow in the same direction as the main plasma current regardless of the launched $n_{||}$ direction (see Fig. 6.3), which suggests that the current is not due to a Landau interaction between the LH waves and electrons in the SOL.

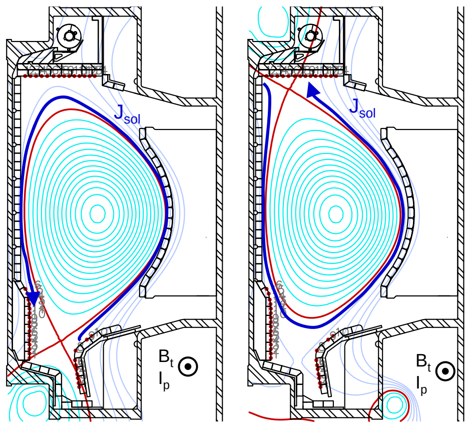

Fig.6.3: Direction of electric currents in the SOL during LHCD for lower and upper null configurations.

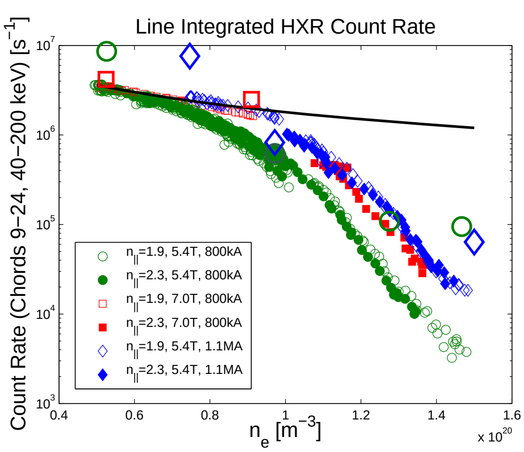

Fig. 6.4: Bremsstrahlung emission predicted by the GENRAY/CQL3D diagnostic including a SOL with collisional absorption (large symbols) shows a dramatic improvement in agreement with experimental data (small symbols) across a wide range of densities as compared to the model without a SOL (see Fig. 1).

By adding a SOL including collisional absorption to the numerical model, the agreement between model predictions and experimental results improves above 10^{20} m^{-3}, as shown in Fig. 6.4. Increasing the temperature of the SOL has been identified as a possible mitigation strategy to reduce absorption outside the separatrix. These results show that strong absorption of lower hybrid waves in the SOL of a high density diverted tokamak is possible and must be considered in computational modeling of future LHCD experiments.

6.2. Simulation Results

For the first time, a finite element method has been used to solve lower hybrid wave propagation for realistic fusion plasma parameters (7,8). For our study we used COMSOL Multiphysics, a commercially available FEM package which allows the definition of the full 3D dielectric tensor of a spatially varying media, which is necessary to model the harmonic propagation of waves in a cold magnetized plasma. With COMSOL we have modeled the exact shape of the tokamak first wall and of the antenna launching structure. Also, the magnetic field topology and the plasma density are directly input from experimental measurements.

With the FEM approach we have modeled in a single self-consistent simulation the behavior of a realistic antenna geometry when facing a (cold) plasma. In particular we have focused on the propagation of waves in Alcator C-Mod for the Lower Hybrid (4.6GHz) and also Ion Cyclotron (80 MHz) frequency ranges. In addition we developed a new approach to do single mode analysis in a 3D FEM solver, and for the LH frequency range we were able to include the effect of electron Landau damping by means of an innovative iterative routine developed in MATLAB (LHEAF code). The LHEAF code has been successfully verified with the TORIC-LH code for a simplified geometry and a Maxwellian plasma. The non-Maxwellian electron velocity distribution arising from the interaction of the LH waves with the plasma is self consistently taken into account by coupling LHEAF to a 1D Fokker Plank code.

The FEM approach has been proven to be extremely efficient and scales more favorably than full wave spectral methods for large simulations. Simulations of LH waves in ITER requiring more than $4x10^7$ unknowns were solved by the aid of the MUMPS library using the massive parallel computing resources at NERSC.

The results of a simulation of a simplified LH launcher as it radiates at 4.6 GHz in a cold magnetized plasma are shown in Figure 6.5. The launcher is composed of an array of 8 phased waveguides. Each waveguide has dimensions 60x5.5mm, and is separated from the adjacent waveguides by a 2mm septum. These dimensions are similar to those of the C-Mod Launcher I. For this simulation the plasma is modeled as a slab, the magnetic field is purely toroidal, and a linear density profile is used. Wave absorption was introduced by adding a finite conductivity in the core of the plasma.

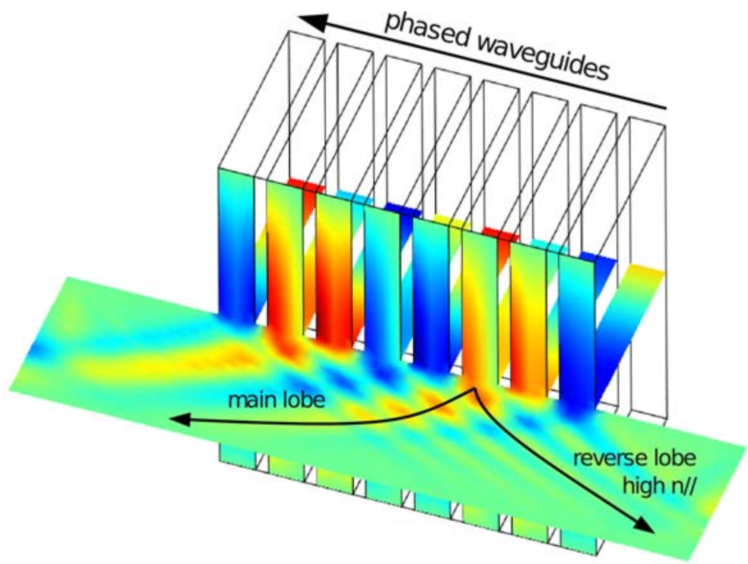

Fig. 6.5: Snapshot of parallel electric field of LH waves. The resonance cones which are characteristic of LH waves are clearly visible.

Figure 6.5 shows a snapshot of the parallel electric field when the phase difference between adjacent columns is 90 degrees. Figure 6.6 shows the corresponding launched parallel power spectrum, as evaluated by taking the two dimensional Fourier transform of the parallel electric field in front of the antenna. The spectrum is peaked at $n_{||}=2$. Figure 7 presents a comparison of the coupling coefficients as computed by COMSOL and TOPLHA for three different phasings, showing very good agreement between the two codes.

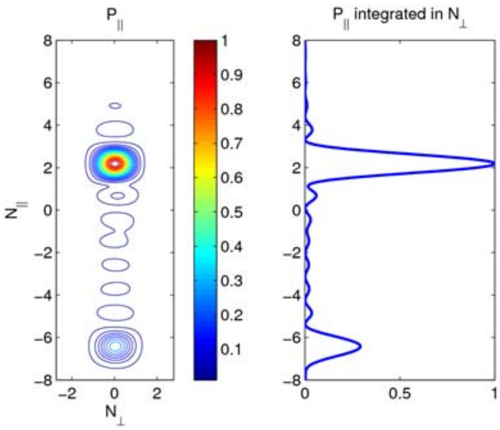

Fig. 6.6: *Normalized parallel power spectrum launched by the eight waveguides LH launcher structure*

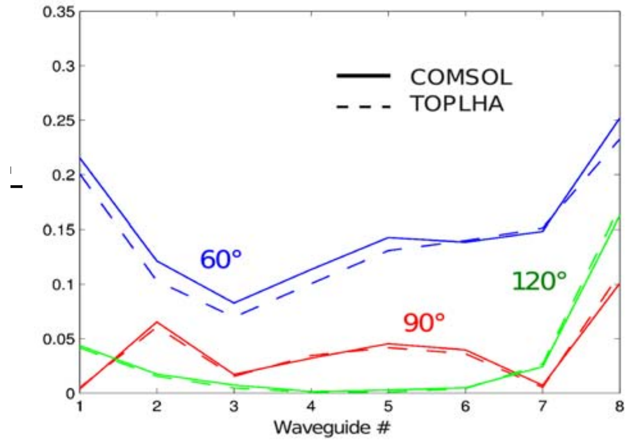

Fig. 6.7: Power reflection coefficient at the input of each of the eight waveguides, for progressive phasing of 60, 90 and 120 degrees

Figure 6.8 shows a simulation of an Alcator C-Mod LH discharge (1080320017). The computational domain includes the scrape-off-layer region, which is critical for several processes involved in the LH wave physics. The results presented take into account the effect of the LH waves on the electron distribution function. The mesh used to resolve the short perpendicular wavelength resulted in 15 million degrees of freedom.

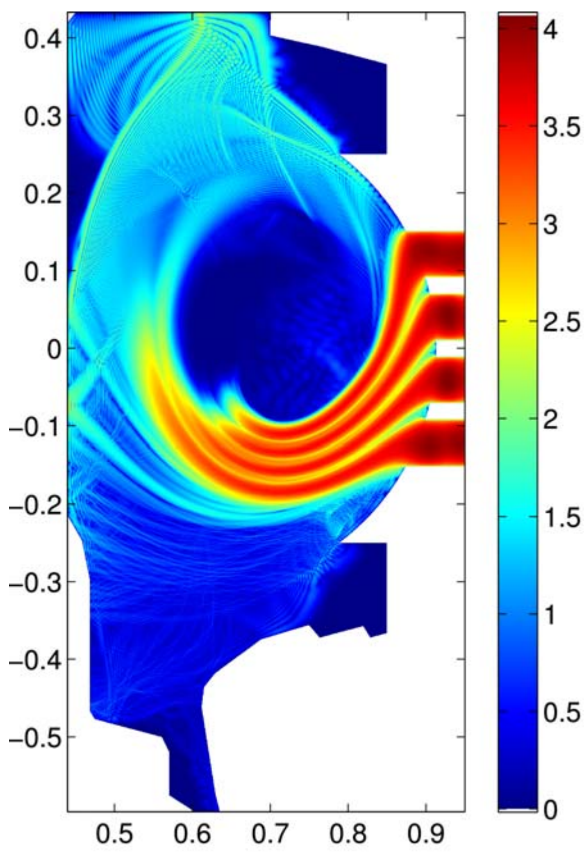

Fig. 6.8: Logarithmic plot of the parallel electric field resulting from COMSOL full wave

The magnetic field topology is based on the EFIT reconstruction of the experimental data. The on-axis toroidal field is 5.4 T, and the plasma current is 800 kA. The density and temperature profiles are based on the measurements from the Thomson scattering diagnostic. The central electron density is 8×10^{19} m^{-3} and the central electron temperature is 2 keV. The density and temperature decay exponentially in the scrape-off-layer region. Four waveguide rows located on the low field side of the torus couple 600 kW of microwave power to the plasma at 4.6 GHz with a toroidal refractive index ($n_\varphi \approx n_\|$) of 2.3.

The figure shows a logarithmic plot of the parallel electric field in the poloidal cross section of the Alcator C-Mod tokamak. In the full wave simulation, the waves propagate through the waveguide structure as TE10 modes. The seamless handling of the coupler and the plasma regions allows one to take into account the reflection from the plasma, which for this simulation is ~5%. The logarithmic plot clearly shows how the LH waves are not confined in the hot plasma region, but can propagate throughout the scrape-off-layer region as well.

6.3. Diagnostic Upgrade and Development

a) *MSE Upgrade*. Current profiles of plasmas with high power LHCD have been analyzed by constraining the EFIT equilibrium code using the internal magnetic pitch angles measured by MSE. Broadening of the current profile with LHCD was confirmed by this analysis. Further analysis including kinetic constraints was carried out in FY2009 and suggested that the shear reversal may have occurred in some LHCD driven discharges (see Figure 6.9). However, all of the MSE data to date is based on changes of pitch angle between a reference OH phase and the LH phase instead of the pitch angle itself, in order to avoid issues caused by thermal birefringence of the MSE in-vessel optical system. Consequently, there is some ambiguity due to an assumption made about the OH phase current profile. Eliminating this ambiguity is crucial for reliable detailed analysis and, in 2009, a new heat shield was installed to thermally isolate the MSE optics from plasma. Significant suppression of temperature variation around the MSE optics has been confirmed, increasing the possibility of obtaining reliable pitch angle measurements in future campaigns.

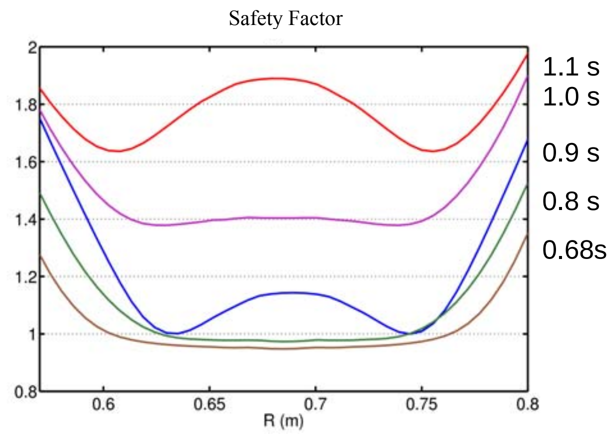

Fig. 6.9: Safety factor profile as a function of time from MSE constrained EFIT with kinetic constraints. The LHCD pulse starts at 0.8 s in this discharge.

b.) *LH wave reflectometer*. The LH wave is a short wavelength electrostatic wave which produces density perturbations typically in the order of 0.01- 0.1 % as it propagates through the plasma. We have investigated the possibility of detecting this density perturbation using a reflectometer in order to directly measure LH waves propagating inside a hot plasma. In the 2008 run campaign, a prototype experiment was carried out on C-Mod and modulation of the reflectometer signal at 4.6 GHz was successfully observed.

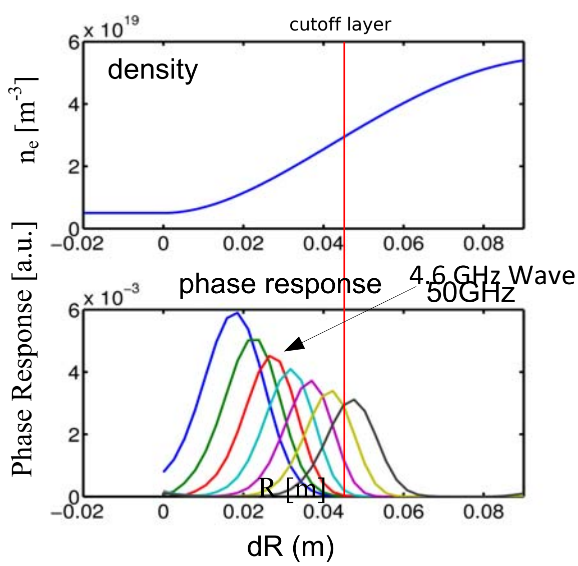

Fig. 6.10: Simulation of reflectometer response to density fluctuations caused by LH waves.

Moreover, the spectrum of the received signal contained several peaks attributed to parametric decay of the LH waves.

Since the LH wave is a short wavelength mode, the interaction between the LH wave and the reflectometer probe beam can not be explained by a simple "reflection" at the cut-off layer. Numerical simulations of the interaction between the LH induced density perturbation and the reflectometer wave were carried out. Results show that the peak of the reflection is not at the cut-off layer, but in front of it, suggesting a contribution due to Bragg scattering of the probe beam (see Figure 6.10). The times of flight of up-shifted and down-shifted signals were also consistent with the selection rule expected from the scattering with a backward LH wave.

6.4. The LH-II Launcher

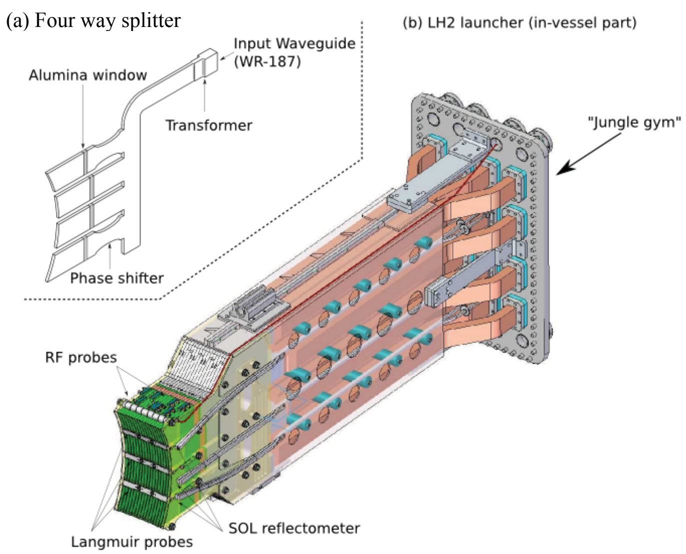

Fig. 6.11: The LH-II lower hybrid antenna. (a) A single four way splitter. (b) The fully assembled vacuum side components of the launcher.

Figure 6.11 shows a schematic of the LH-II launcher. A major change from the previous launcher is that it employs a novel four way splitter. The splitter evenly splits the microwave power from klystrons in the poloidal direction. 16 splitters are stacked in the toroidal direction, for a total of 16 columns and 4 rows of active waveguides. A total of 8 passive waveguides (one on each end of each row) are installed to reduce the reflection on the edge columns. The splitters are connected to standard WR-187 waveguides via a transformer, and a new "jungle gym" connects the launcher with the transmission line from the klystrons. The phase difference between the columns can be controlled by phase shifters and by independent phasing of the klystrons, allowing for $n_{||}$ to be adjusted from -3.8 to 3.8.

For better prediction and understanding of the LH-II launcher performance, the modeling of the four-way splitter was conducted by using TOPLHA and CST microwave studio. TOPLHA solves the antenna-plasma coupling problem assuming a stratified plasma, and CST microwave studio solves the EM problem of the vacuum region. The S matrixes of the two codes are then cascaded to obtain the overall self-consistent solution. In these simulations a linear density profile, defined by $n(x)= dn/dx\ (x-x_0) + n_0$ (for $x > x_0$) and $n(x) = 0$ (for $x < x_0$), where n_0 is the minimum density, x_0 is the vacuum region thickness and x is the distance from the launcher, was used. By changing these parameters, we tested the robustness of coupling.

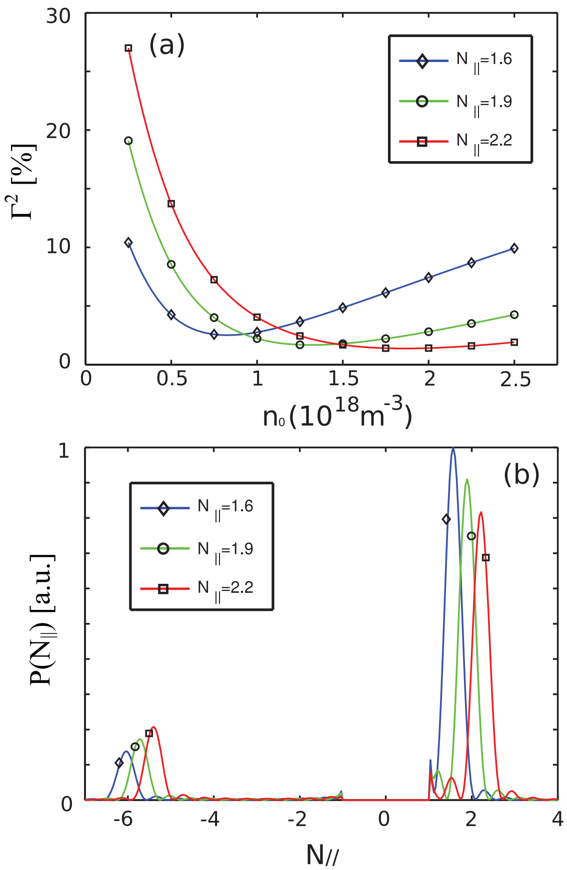

Fig. 6.12: (a) Reflection coefficient as a function of edge density. (b) Launched $n_{||}$ spectra for different waveguide phasings.

Figure 6.12(a) shows the predicted reflection coefficient as a function of n_0 for different launched $n_{||}$. In this case, $dn/dx = 10^{20} m^{-4}$ is used. It can be seen that good coupling (less than 10% reflected power) can be achieved over a wide range of n_0. Figure 6.12(b) shows the associated $n_{||}$ spectrum for the n_0 of 1.25×10^{18} m^{-3}. In Fig. 6.12, the density profiles in front of all four rows of launcher are assumed to be same. In real experimental conditions, the density in front of the launcher was often observed to be not uniform in the poloidal direction. Hence, we tested this effect by assuming different density profiles in front of different rows. A typical case is shown in Figure 6.13. $n_0 = 0.5\times10^{18}$ m^{-3} for top and bottom rows and $n_0 = 1.25\times10^{18}$ m^{-3} for the middle rows were used. Very little adverse effect was found on the reflection at the four way splitter input (Fig. 6.13 (a)), and the $N_{||}$ spectrum was also not affected. The major impact of the poloidal asymmetry of the density is the uneven power splitting in the poloidal direction (Fig. 6.13 (b)). In this case, more power was injected from the middle rows. This suggests the importance of measuring the forward and the reflected power for each row.

In order to characterize its performance experimentally, the LH-II launcher is equipped with rich diagnostic tools. 32 RF waveguide probes are installed on a carefully selected set of active and passive waveguides to measure the forward and reflected power in each of these waveguides. These measurements are also planned to be used as a part of a new coupler protection system. A total of six Langmuir probes and three X-mode reflectometer waveguides are also installed to measure the density profile in front of the launcher. The X-mode reflectometer covers the frequency range from 100 GHz to 140 GHz, providing density profiles with the sweep time of 0.01-1 ms.

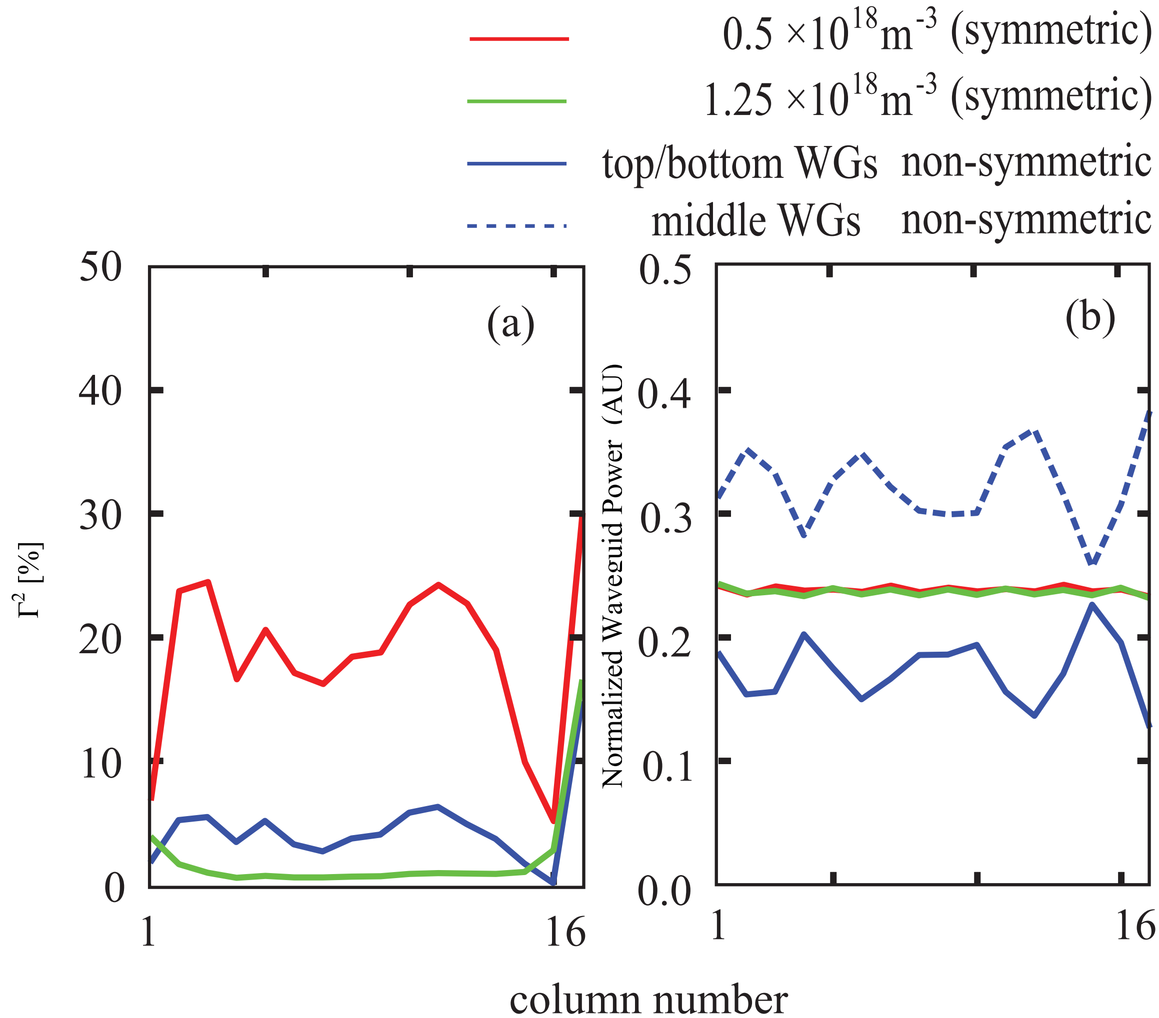

Fig. 6.13: (a) the power reflection coefficient (%) at the input port of four way splitter on each column, and (b) the normalized forward power on rows, where the perfectly even power splitting without absorption corresponds to 0.25 (N||=1.9)

6.5. Research Plans for 2011-2012

6.5.1. Coupling Studies:

The coupling performance of the LH-II launcher will be evaluated under a variety of conditions. Good coupling performance at large launcher to separatrix distances is of critical importance to the proposed ITER LHCD system and will be explored with the new antenna. Compatibility of the LH-II launcher with the various ICRH antennas on C-Mod, including the new tilted four strap antenna, will also be investigated.

A precise analysis of the physics of LH wave coupling can be pursued utilizing the 32 RF probes installed on the LH-II launcher. In particular, the diagnostic allows a detailed study of the coupling pattern at the grill mouth, which is key for the validation of LH coupling models which are currently used for the design of LH experiments. With the aid of six Langmuir probes and an X-mode edge reflectometer system (installed on LH-II) to provide density profile measurements in front of the launcher, the validity of LH linear coupling theory can be assessed and the effect of nonlinearities with respect to LH power and of transient events such as edge-localized-modes (ELMs) can be studied. Finally, the microwave probe measurements are also part of the coupler protection system, allowing for reliable arc detection and subsequent shutdown of the high power klystrons.

6.5.2. Density Limit:

The research plan for further exploration of the density limit will build upon experiments conducted on C-Mod during 2008. These experiments used fast electron bremsstrahlung and electron cyclotron emission as proxies for the non-thermal part of the electron distribution function. In order to scan a large parameter space in the fewest number of discharges, the plasma density was intentionally ramped over the range of $n_e = 0.5 - 1.5\text{x}10^{20}$ m^{-3}. This makes it very difficult to determine a quantitative measure of the current driven in the plasma since the density was changing on a timescale short compared to the current redistribution time. Repeating these experiments with steady density and temperature profiles will require additional experimental time but will allow for measurements of the driven current in addition to X-rays generated by non-thermal electrons.

The proposed experiments will make use of current profile diagnostics to directly measure the current profile as a function of plasma density, toroidal magnetic field, plasma current, plasma temperature, parallel refractive index of the wave, and plasma topology. Both the Motional-Stark-Effect (MSE) and Faraday Rotation Polarimeter systems are receiving significant upgrades for the 2010-2011 run campaigns which should allow for more sensitive measurements of the plasma current profile.

In addition to determining the driven current by means of current profile measurements, the total driven current can also be obtained by allowing the plasma current to relax, over the course of several hundred milliseconds, to a steady state under conditions of zero loop voltage which should be achievable with the improvements made to Coupler II. This will allow a simple measurement (via Rogowski coils) of the plasma current without the need to subtract the Ohmic component, thereby giving a clearer indication of the current drive efficiency for a set of plasma conditions. Observations of SOL currents during LHCD at high density, along with the presence of localized RF wave fields in the outer SOL, suggest that at high density the LHCD power not absorbed by the core is deposited near the edge or outside the plasma. Current data on the SOL RF wave fields exists for only a few shots where a horizontal reciprocating Langmuir probe on the low field side of the tokamak was connected through a band-pass filter to a rectifier diode. A better diagnosis of the SOL RF wave fields using both vertical and horizontal scanning Langmuir probes on both the low field and high field sides should be conducted along with the current drive efficiency measurements.

In previous experiments there has been no direct measurement of the presence of LH waves in the core plasma. Above the density limit there are indications that the fast electron population disappears although it is not known if the waves penetrate into the core plasma. Installation of a 60 GHz and 75 GHz O-mode reflectometer will make it possible to detect 4.6 GHz oscillations near the location of the cutoff layers ($n_e = 4.5\times10^{19}$ m^{-3} and 7.0 $\times10^{19}$ m^{-3}). The phase contrast imaging system is scheduled to receive an upgrade which would allow for heterodyne detection of LH waves in the core plasma. An X-mode reflectometer system will be attached to the LH II antenna for measurement of the SOL density profile. This new capability will be a significant benefit to understanding the density limit.

Modeling of the experimental results has given some indications of how the density limit can be overcome. Increasing single pass absorption on closed field lines is likely to decrease the parasitic losses in the SOL. This may be achieved by applying LHCD to hotter L- and/or I-mode targets with auxilliary heating provided by ICRH. Recently developed 8 T discharges with ^{3}He minority heating, central electron temperature up to 9 keV and line-averaged density of 8×10^{20} m^{-3} are especially attractive for this purpose. Also, the use of compound spectra will be investigated. These options are discussed in greater detail in Section 9.

This new physics is of high importance not only for C-Mod advanced scenarios but potentially for ITER and for proposed future steady state devices (for example the Fusion Development Facility) and for DEMO. In recognition of this, a new ITPA Joint Experiment 5.3: *Assessment of lower hybrid current drive at high density for extrapolation to ITER advanced scenarios* was started in 2009 which will be led by MIT and will also involve experiments on FTU, Tore-Supra, JET and EAST, and modeling from the associated groups. As outlined above, C-Mod experiments on this topic will begin in FY10 and extend into FY11.

6.5.3. Simulation and Modeling:

In the near term we plan to validate LHEAF through comparison with experimental observations. We are now in the process of coupling LHEAF to a bounce-averaged zero-banana-width 2D Fokker Plank code. This is a crucial step to be able to accurately predict the electron distribution function which results from the interaction of the electrons with the LH waves. The integration with the new Fokker-Plank solver is also important for accurately evaluating the power deposition profile and the driven current. A synthetic hard X-ray diagnostic is under development and will be extensively used for comparing simulation results with experimental observations. We expect the validation process to extend throughout FY2010 and well into FY2011.

After completion of the verification stage, we plan to start using LHEAF as a predictive tool for studying LH physics problems which are still unanswered. Among these issues, the most immediate application for LHEAF is the study of the LHCD density limit observed on Alcator C-Mod and the physics associated with a cold plasma edge. Also, the capability to accurately evaluate the amplitude of the wave electric field inside of the plasma should enable the study of high energy electron production in the SOL region by near field effects. Similarly, collisionless damping of the high frequency components of the $n_{||}$ spectrum on thermal electrons will also be evaluated with LHEAF. We expect this stage to take us through FY2011.

LH induced plasma rotation is another physics problem of particular interest for LHCD and the fusion community in general. Using LHEAF the electric field profile in the plasma is calculated, so in principle RF generated radial diffusion can be evaluated by first principles. However the overall picture cannot be obtained by LHEAF by itself, but further physics must be included in the modelling, for example by coupling to an external transport code. Recently developed transport codes which do not rely on the charge neutrality assumption may be used for this purpose. This is a long term project for the FY2012.

6.5.4. Use as a Tool for Profile Control:

During the latter part of the 2008 campaign the LHCD system progressed from an extended checkout phase to a reliable tool for profile control. The checkout phase for the new LH-II launcher is expected to require much of the remaining 2010 campaign, after which it is expected that LHCD system will again be used as a tool supporting other research areas (MHD, rotation/transport, advanced scenarios). With the addition of an additional coupler and the increased source power from new klystrons procured in 2010 (made possible by incremental ARRA funding) this capability will increase moving into 2011 and 2012.

6.5.5. Hardware Upgrade:

LH current drive experiments carried out with the Coupler I have revealed that more power will be needed to control the current profile in Alcator C-Mod discharges, particularly at moderate densities where current drive effciency falls off as $1/n_e$. A key motivation for the development of the new coupler LH II was in fact to decrease the losses in the power splitter and make more power available for delivery to the plasma. Nevertheless, even accounting for this power upgrade modeling results indicate that still more power will be needed to achieve our goal of full non-inductive current drive by applying LHCD to regimes with high bootstrap current.

Our plan is to increase the LH source power by 1) installing a fourth cart of klystrons, bringing the total source power to 4 MW, and 2) adding a second coupler which will increase the projected power delivered to the plasma to ~ 2.5 MW. Thanks to ARRA funding, 7 new klystrons are being procurred this year (FY 2010) and they, together with 9 klystrons from Alcator C, will be sufficient to populate the 4 carts with 4 klystrons each, bringing the total source power to 4 MW. Work has already begun on the fabrication of the 4th cart, also ARRA funded, and will be completed in 2011. As for the additional coupler (LH-III), the plan is to build it along the lines of Coupler II, perhaps with a few modifications to improve the fabrication process. Naturally, the plan is dependent on a full evaluation of Coupler II's performance, which will only be possible toward the end of FY 2010 and early FY 2011. The plan is to fabricate LH-III in FY 2011-2012 and install it in FY 2012. Having two couplers will greatly enhance the flexibility of the LH system, for example by allowing investigation of synergies possible with two independent launched spectra.

6.6. References

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7. 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. Much of the MHD research on C-Mod involves 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.

7.1. 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[1,2]. Over the most recent couple of years our efforts have been 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[3]. Currently the ITER approach is to quench the avalanche process by massive gas injection to reach the Rosenbluth density (~10^{22} m^{-3})[4], 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[5]. 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 previous run campaign, which used the C-Mod lower hybrid (LH) system to generate a large seed population of runaways, 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, nearly 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 2010-12 time frame we plan to test this both experimentally and theoretically once the LH system is reinstalled on C-Mod, 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 is also being carried out. Assuming that significant runaway populations will be found in low-elongation current quenches opens up additional experimental opportunities for research on controlling and mitigating the

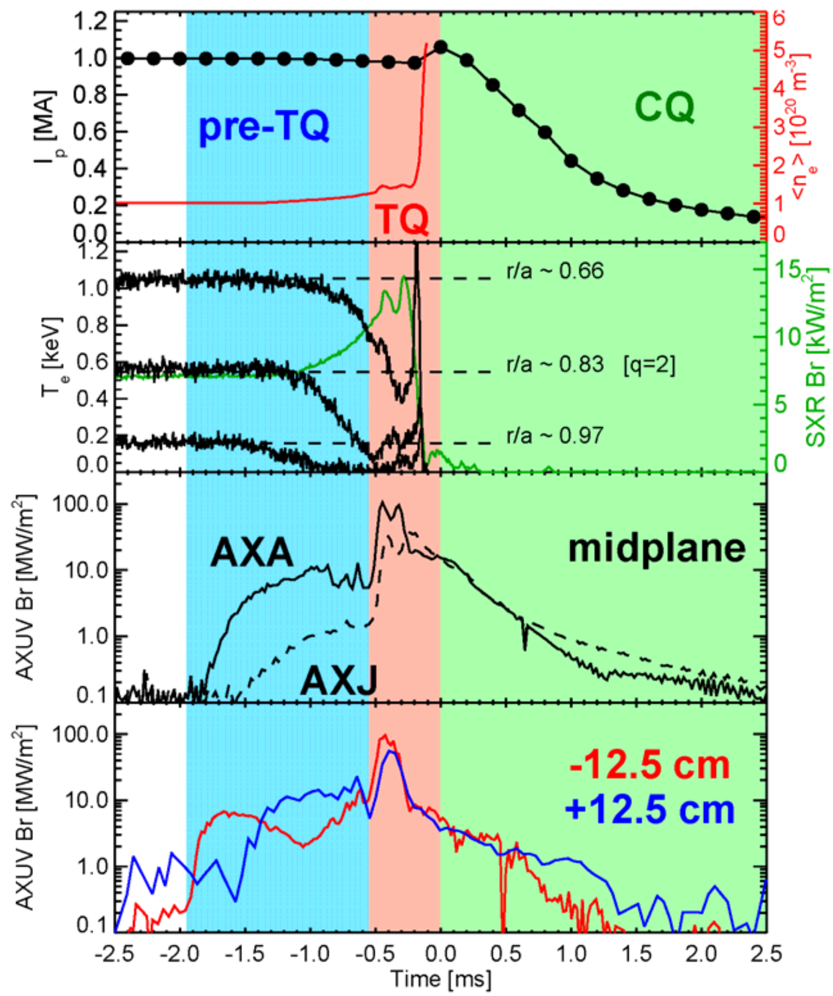

Fig. 7.1: Gas jet mitigated disruption, showing in panel 3 a large toroidal asymmetry of Prad during the pre-thermal-quench, which becomes nearly symmetric during the thermal quench (TQ) and current quench (CQ)

runaways. Specific research topics will include pre-programmed and feedback control of the position of the RE channel, control of deposition location, controlled inversion of the OH power supplies to decelerate the REs, and possibly testing the effects of impurity or pellet injection on avalanching and/or deconfinement of the RE population.

In addition to the runaway electron issue, recent work has also concentrated on studying the toroidal variation of radiated power in gas jet mitigated disruptions. In ITER, a factor of 2 toroidal variation of the radiated power during a disruption risks melting the

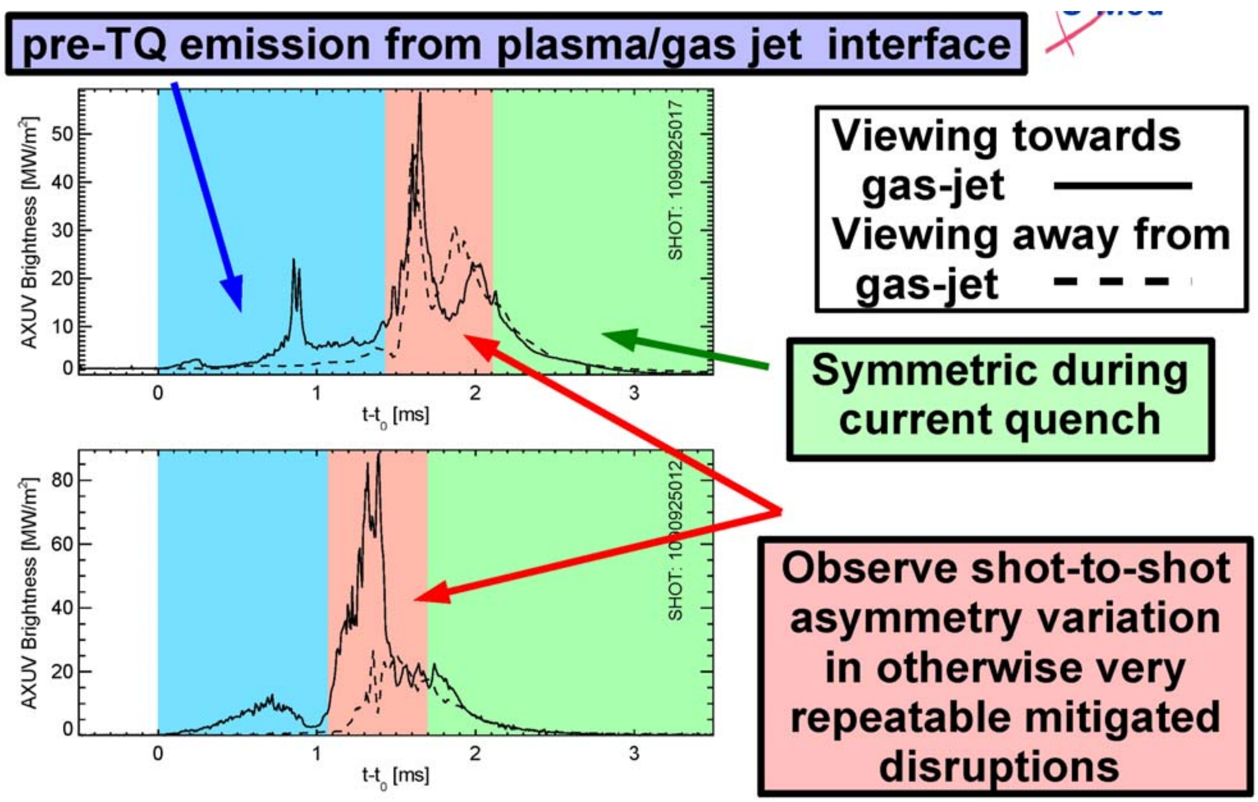

Fig.7. 2: A large shot-to-shot variation is observed in the toroidal asymmetry of the radiated power during the thermal quench in otherwise identical mitigated disruptions.

beryllium plasma facing components[6]. An understanding of this asymmetry is also required in order to specify the number of gas jet valves needed on ITER. Radiation asymmetries during gas jet mitigated disruptions are studied primarily with a number of solid-state diode (AXUV) arrays at various locations around the torus. These studies also employ the soft x-ray arrays and the T_e and n_e diagnostics. Initial results from last year showed that the radiated power was strongly peaked (about an order of magnitude) around the location of the gas jet during the pre-thermal quench (i.e. the period of time

when gas from the jet is affecting the plasma, but before the disruption begins), but that the radiated power became more-or-less symmetric during the thermal quench (TQ) and continuing throughout the current quench (CQ), as shown in panel #3 of Fig. 7.1. However, there is a very large shot-to-shot variation in the asymmetry during the TQ, as shown in Fig. 7.2. This variation is observed even though the mitigated disruptions are otherwise very repeatable, and the reasons are not understood yet. Integrating the instantaneous radiated power over the disruption gives the total radiated energy, and this also exhibits large shot-to-shot variation in toroidal asymmetry, ranging up to a factor of 2.5, which is high enough to cause beryllium melt damage in ITER. The asymmetry in total radiated energy does not show any clear correlation with target plasma conditions.

Our plans for future research on gas jet disruption mitigation include understanding the cause of the toroidal asymmetry and variability of P_{rad} in the thermal quench. This will involve more detailed measurements of P_{rad} at the gas jet/plasma interface using an expanded diagnostic set that will include fast visible spectroscopy, Lyman-α detectors, and more AXUV coverage. We will also be installing a 2nd gas jet system (supplied by ORNL) on the opposite side of the torus from the existing gas jet, and exploring disruption mitigation effectiveness and asymmetries using these multiple jets.

Further development of real-time disruption prediction, and possibly automated mitigation and/or avoidance activation, are planned. The ultimate goal is to be able to recognize and act on impending disruptions due to a number of different causes, such as locked modes, impurity injections, ß-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(s) 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.

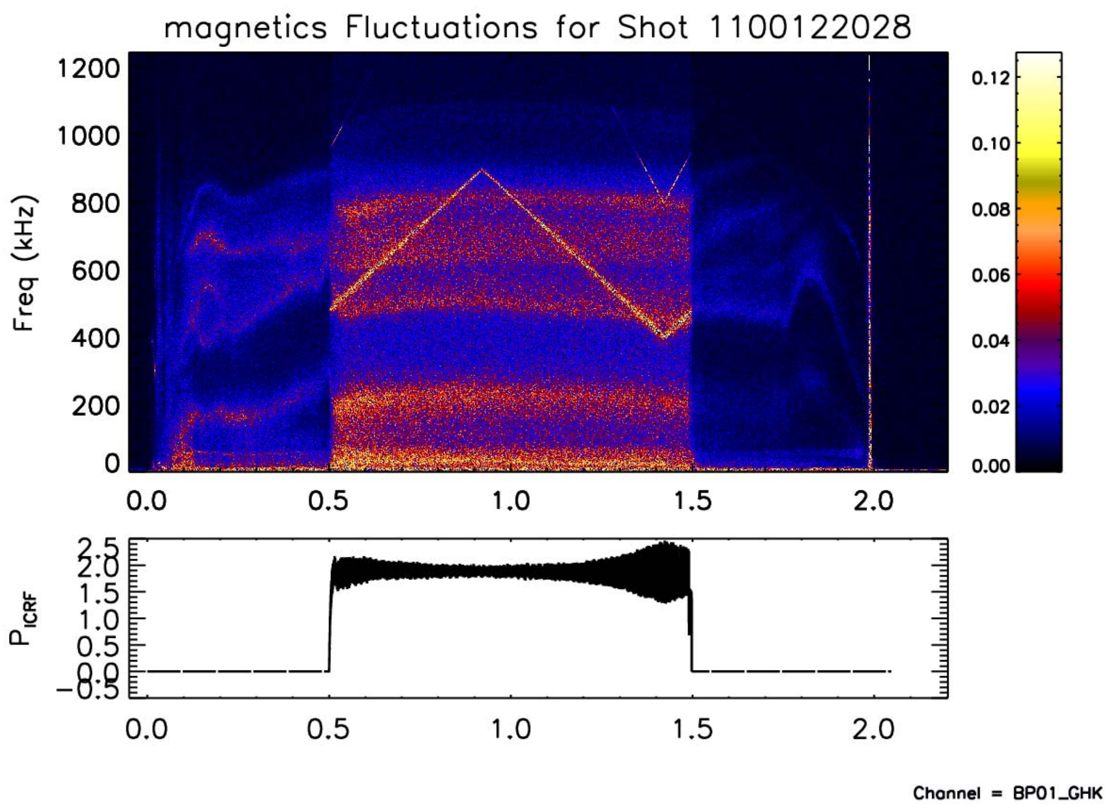

Fig. 7.3: Initial test of using amplitude modulation of the ICRF power at AE-relevant frequencies to produce an AE drive wave in the plasma core. In this example the power modulation was swept through a range of AE-relevant frequencies (400-900 kHz).

7.2. AE Modes, Alfvén Cascades, and Fast Particles

Alcator C-Mod has a number of resources for studying Alfvén instabilities and fast particle interactions. These include a pair of active MHD antennas, which can excite stable ITER-relevant moderate-*n* Alfvén eigenmodes (AE) and measure their damping rates as a function of plasma parameters. Large populations of fast particles are generated by the ICRF heating system on C-Mod, which produces primarily trapped fast ions, and the LHRF current drive system, which produces energetic electrons. Diagnostic capabilities include a large set of toroidally and poloidally arrayed Mirnov coils, phase contrast imaging (PCI) to measure core density fluctuations, newly upgraded arrays of compact neutral particle analyzers (CNPA) to measure the energy and spatial distribution of confined energetic ions, and a hard x-ray camera for measuring suprathermal electron spatial and energy distribution.

Recent work in this area has concentrated on studying reversed-shear Alfvén eigenmodes, both during I_p rampup and also during sawtooth cycles [8,9]. The observed frequency up/down chirping and other details of the frequency evolution provide 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.

Plans for continued research in this area include studying the interaction between ICRF and AEs, including AE-induced effects on ICRF heating efficiency through mode-induced losses of the ICRF fast hydrogen minority population. Schemes for generating larger amplitude AEs using the ICRF have been proposed. One method involves using the 500 kHz beat frequency between the D and E antenna/transmitter systems (80.0 MHz and 80.5 MHz respectively) to drive Alfvén modes. Another recently suggested technique involves modulating the power of one of the transmitters at AE-relevant frequencies (300-900 kHz at typical C-Mod densities and B-fields) to generate an AE drive frequency in the ICRF spectrum. The power modulation can even be swept during the plasma shot to cover a range of AE frequencies, as shown in Fig. 7.3. In principle, both of these techniques could provide much larger amplitudes to drive AEs compared to the active MHD antennas, as well as placement in the core rather than the plasma edge. The larger modes that would be generated should be more precisely measured with the C-Mod diagnostic set, and could provide high-quality datasets for benchmarking codes (NOVA-K, AORSA/CQL3D) against experiment. An additional goal would be to directly study AE-induced losses of fast ICRF-generated minority ions using the CNPAs, and possibly even by monitoring sawtooth reheat rates and fusion neutron production. Future proposed upgrades to the diagnostic set include a fast ion H_α diagnostic to look at confined ICRF ions, as well as an array of fast ion loss diagnostic, which could be either 2-D scintillators, or arrays of Faraday cups.

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8. 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 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.

In addition to the nominal Q=10 Scenario, a focus of the near-term research program is to support preparation of scenarios for the ITER "Pre-nuclear commissioning phase". During the first two to three years of ITER operation, prior to issuance of a nuclear license, operation will be restricted to hydrogen and helium plasmas. It is envisioned that a substantial fraction of this operation will be carried out at reduced parameters, at about half the design field and current, and initially with less than the full complement of heating and current drive systems. This regime of operation would also require the application of alternate RF heating scenarios. A key issue for the pre-nuclear phase is accessibility to H-mode and the opportunity to assess ELM mitigation techniques.

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.

8.1. Recent Research Highlights

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, 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.

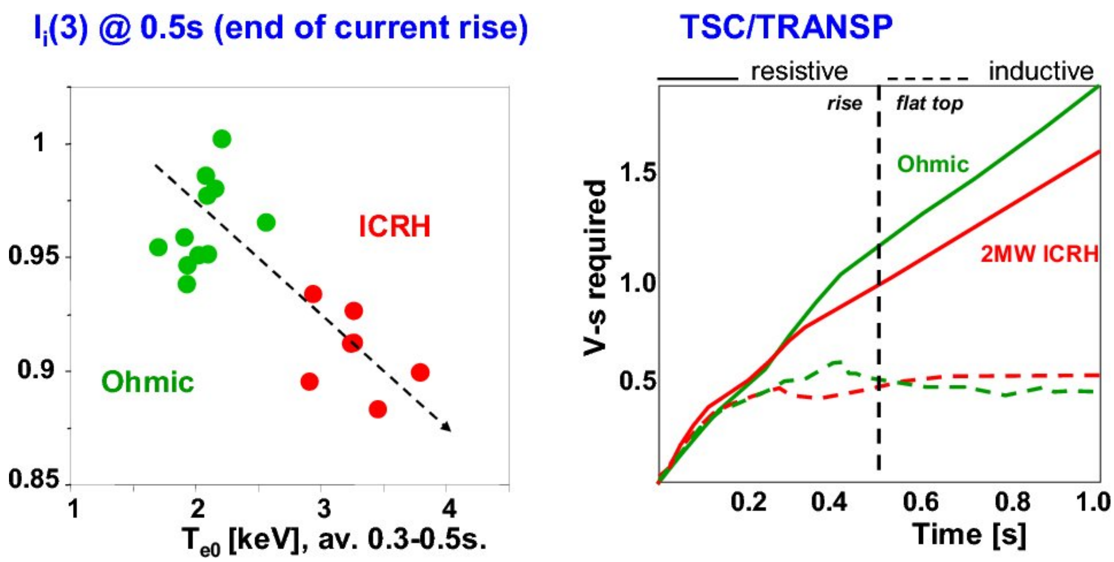

Fig. 8.1: Result of heating during diverted phase of ITER-like ramp-up scenario. Reduction in l_i (left) is modest. TSC analysis (right) indicates main saving is in resistive flux consumption.

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. During 2009-10 we improved the control of the early ramp-up to allow earlier X-point formation, achieving the ITER target of divertor operation at < ¼ of the flattop current. The earlier X-point scenario resulted in lower Z_{eff} and facilitated application of ICRF heating during the current ramp.

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}. The resistive volt-second consumption was reduced by the ICRF heating. 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.

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} ~3.2 . The plasma remained diverted throug 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. Increases in the OH1 current (comparable to CS coil in ITER) at the H-L transition, predicted in TSC simulations, were observed in the C-Mod experiments.

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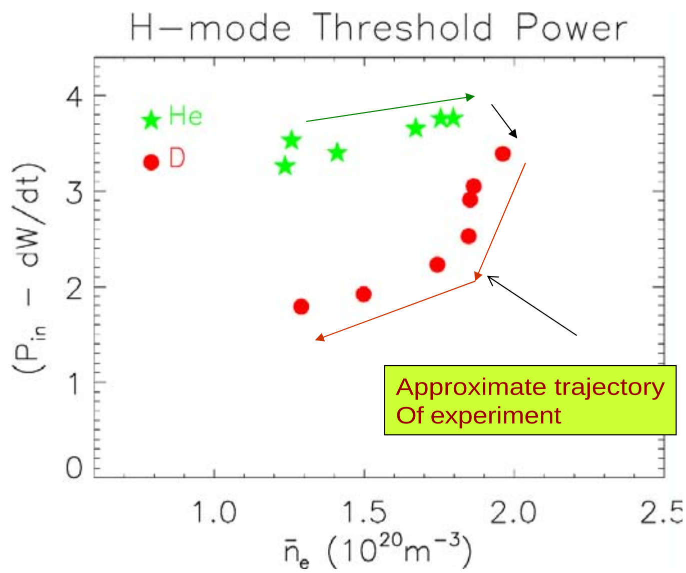

Fig. 8.2: Threshold power in helium and deuterium plasmas.

A comparison of the L-H transition threshold in He and D plasmas was carried out in support of planning for the ITER pre-nuclear phase. In rhe C-Mod experiments, which were conducted in collaboration with Joe Snipes of the ITER Organization, the power threshold in helium was found to be higher than in deuterium by a factor of 1.2 to 1.8, with the higher ratio observed at the lowest density and lowest absolute threshold power. The C-Mod experiments were part of a multi-machine program (ITPA Joint Experiment TC-4), which resulted in inconsistent comparisons of the He and D thresholds, ranging from observations of similar threshold powers to up to a factor of two higher threshold in helium. Implications for helium operation in ITER are therefore inconclusive, and further investigations, emphasizing the underlying physics rather than empirical scalings, are required

Power requirements for access to good quality (H_{98y2}~1) H-modes are an important issue for ITER. In 2009-10 we have carried out a series of experiments, in collaboration with Alberto Loarte of the IO, to examine pedestal and confinement characteristics as a function of margin above the threshold power (P_{LH}). Using impurity seeding with different radiating species (Ar, Ne, N_2) we also investigated the role of the localization of core and edge radiation on the power requirements. Experiments were carried out in seeded and unseeded stationary EDA and ELMy H-mode plasmas, with the ICRF power

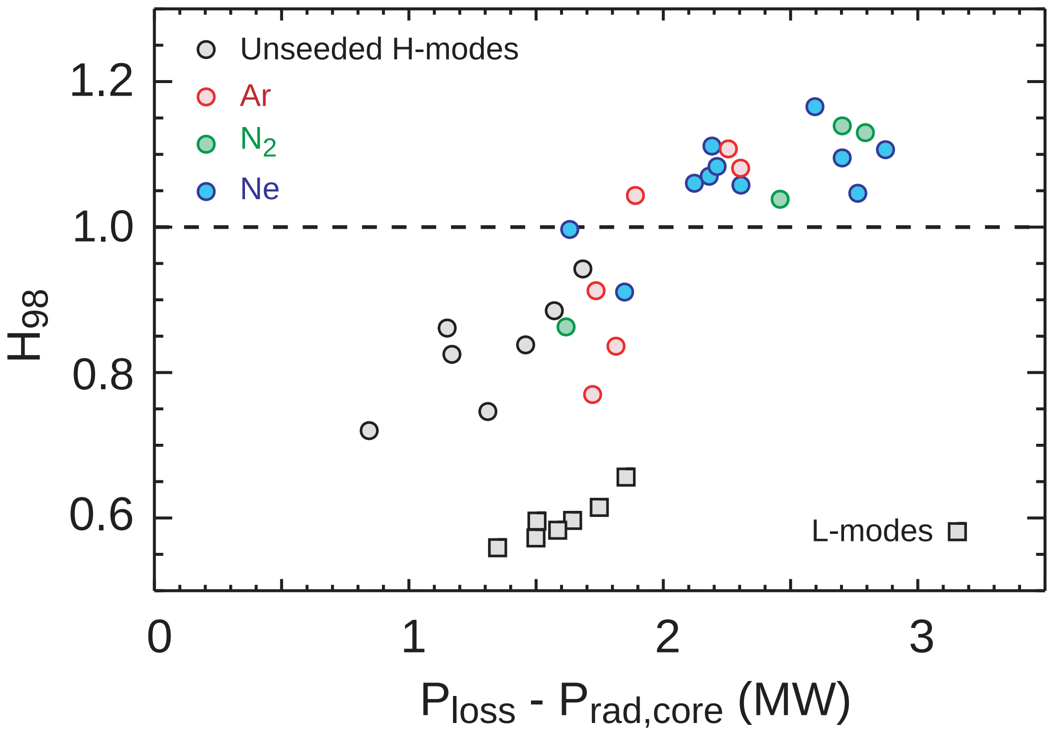

Fig. 8.3: Good H-mode confinement is observed at with P_{net} comparable to the L-H threshold.

adjusted to allow the net power conducted through the pedestal to be matched. In seeded EDA H-modes, we have obtained good H-mode confinement with P_{rad} sufficient to substantially reduce the divertor heat load. Degradation of confinement at high values of P_{net}(core) is correlated with a decrease in the pedestal temperature, consistent with the usual picture of profile stiffness. Pedestal cooling is more pronounced with Argon seeding than with Neon or Nitrogen, as expected from the radiation profiles. It was also found that impurity seeding reduces the incidence of high Z impurity injections and also reduces ICRF trips. Neon seeding has been found to lead to higher performance H-modes than obtained in unseeded conditions.

8.2. 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.

8.3. 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[1], 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.

8.4. ITER H-mode Operational Scenarios

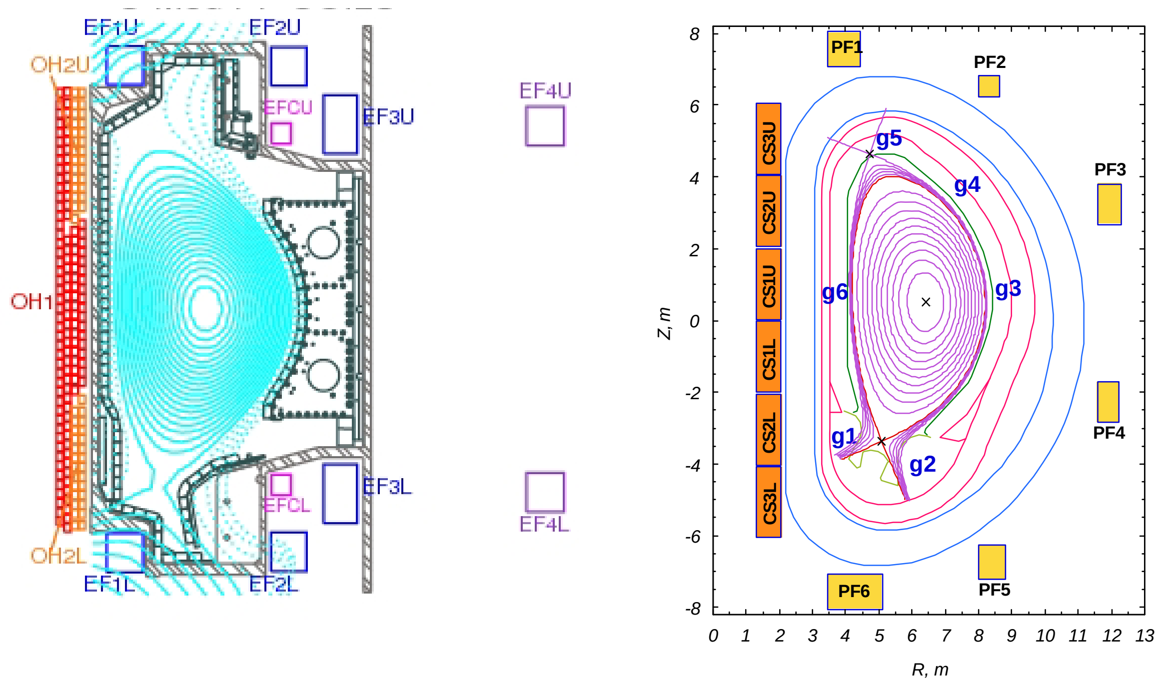

Fig. 8.4: C-Mod (left) and ITER (right) poloidal field coil sets have similar arrangement and functionality. Shape control depends on combinations of coils to maintain a small number of gaps, with minimal null space.

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.

8.5. 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 has addressed 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. 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.

8.6. 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 $q\leq3.2$ 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 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 q95 for which this regime can be attained; our highest pressure H-modes to date were at q95 = 3.9, while the ITER reference scenario is at q95 ~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 $\tilde{\nu}\omega$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 4 shows example parameters of C-Mod discharges which match the ITER H-mode operating point in β_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 $5\times10^{20}\,m^{-3}$, corresponding to the upper two collisionality matching points. High power H-mode operation in C-Mod at q_{95}~3 and a density of $2\times10^{20}\,m^{-3}$, 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

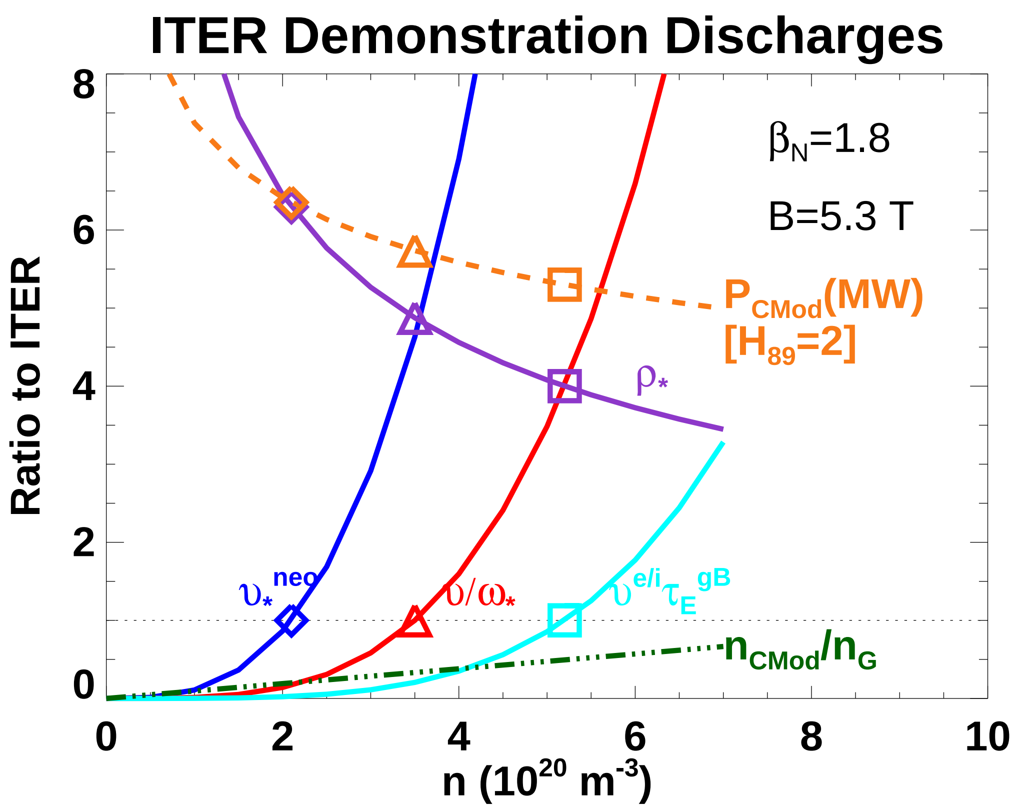

Fig. 8.5: Parameters of ITER H-mode demonstration discharges in C-Mod which match ITER in terms of three dimensionless measures of collisionality. Vertical axis corresponds to the ratio of the C-Mod parameters to the ITER nominal values for ρ^* (purple) and each collisionality parameter. Also shown are the power (MW) required for C-Mod (yellow) and the Greenwald density parameter of the C-Mod discharge.

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.

8.7. 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 extended C-Mod H-modes to lower $\square^*$ 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.

8.8. 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 2012 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[2] 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.

8.9. 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[3] 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[4]. 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[5,6], 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[7] 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 $\beta_N \sim 1.8$, 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.

8.10. 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
Power ratio – hysteresis and access to H-mode with H~1	TC-2	Assess existing database; propose additional expt if warranted
L-H threshold power at low density	TC-3	Initial C-Mod expt. completed X-point scaling 2010-11
H-mode transition and confinement dependence on ionic species	TC-4	for ITER pre-nuclear phase begun 2009
He profiles and transport coefficients	TC-11	Evaluating diagnostic capability 2010
H-mode access with different X-point height	PEP-28	New – related to TC-3
ITER demo at q95~3, β_N=1.8,n_e=0.85n_G	IOS-1.1	Now includes He & H studies 2010...
Study seeding effects on ITER demo discharges	IOS-1.2	Initial studies 2009
Rampdown from q95=3, demo conditions	IOS-2.2	Continue experiments begun under SSO-5
Control of experimentally simulated burning state	ISO-6.3	New DPCS enhancements, 2011...

H-mode Scenarios research on C-Mod directly supports ITER short term needs[8]

- 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 scenario

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9. 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. This research is closely aligned with Theme II of the 2009 Research Needs Workshop, "Creating predictable, high performance, steady state plasmas". 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. However, we also use the flexibility of C-Mod and its unique parameter space to explore new operational regimes which are not currently in the ITER plans but may prove attractive for later phases of operation or for future facilities. In response to the ReNeW recommendations, we place strong emphasis on integrating all of our scenarios with high heat fluxes (~ 1 MW/m^2), since this will be a key requirement for high performance steady state operation and is a unique capability of C-Mod.

9.1. Research Highlights from FY09

Experimental demonstration and assessment of advanced scenarios with active current profile control was deferred due to unexpected delays in the fabrication and installation of the new LHCD launcher, as discussed in Section 6. However, considerable progress was made in developing target plasmas which will broaden the range of potentially attractive scenarios, and in LHCD and integrated modeling which are critical to planning and interpreting future experiments.

Plasma breakdown and ICRF-heated rampup were developed at reduced field (3.5 T). While most present and planned C-Mod operation is at higher field (5.4-8 T), lower B could have some advantages for advanced scenarios by enabling lower q_{95}, and β_N, at a given plasma current and β_{pol}. This must be balanced against reduced LHCD accessibility. In past campaigns, plasma breakdown was always near 5.4, and B_T ramped to the desired target. This uses a significant portion of the discharge and, more importantly, prevents use of the current ramp phase for ICRH or LH tailoring of low B discharges. In FY09, reliable breakdown at 3.2 T was developed, and ICRF at 50 MHz was coupled into the current ramp, heating the plasma and producing H-modes. As anticipated, l_i was reduced for given I_p with respect to 5.4 T discharges, and β_N increases at given input power (Figure 9.1). It should be noted that currently only half of the installed ICRF sources are tunable, limiting P_{RF} to about 2.5 MW and reducing β. An upgrade of the remaining sources is proposed in FY12, following further experiments and integrated simulations discussed in the next section. The improved capability for reduced B_T discharges proved useful for a number of experiments in the Topical Science and H-mode scenarios areas.

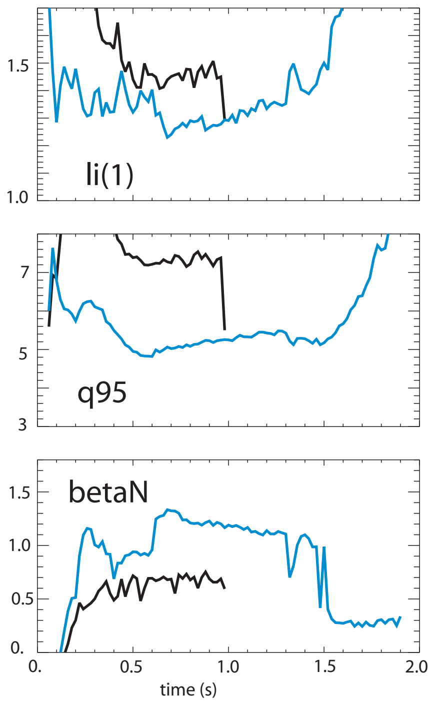

Fig. 9.1: Comparison of 450 kA H-mode discharges with 2.5 MW of ICRH. The 3.5 MW discharge in blue used 50 MHz heating, applied at 0.6 seconds. The black traces are from an earlier 5.4 T comparison discharge. In other discharges, at both fields, RF has been applied as early as 0.1 seconds, during the current rampup.

Advanced scenarios should ideally combine high self-driven “bootstrap” current, which requires high input power and plasma confinement, with significant external current drive for profile control. This implies tradeoffs, since the greatest energy confinement has generally been obtained in H-modes. These also have a strong particle barrier and high density, which reduces external current drive for all techniques, including lower hybrid current drive. Great progress has been made in FY09 and 10 in developing and extending the I-mode regime, discussed in detail in the Pedestal section, demonstrating steady, high performance discharges with high input power (up to 6 MW), energy confinement up to or exceeding H-mode levels, but without a particle barrier. As shown in Figure 9.2, H_{ITER98} is up to 1.1 and increases with input power. The low densities, readily controllable with our cryopump, and high temperatures (up to 6 keV) obtained are ideal for strong LH absorption and current drive, one of several motivations for exploring this new regime. The positive results open up new avenues for development of integrated scenarios, which will be assessed both in simulations and experimentally using the new LHCD launchers. Very recently, D (He^3) heating in 8 T L-Mode plasmas, with low density and high power, has achieved even hotter plasmas (with peak T_e 8-9 keV) (Section 5), suggesting new directions to extend the I-mode regime.

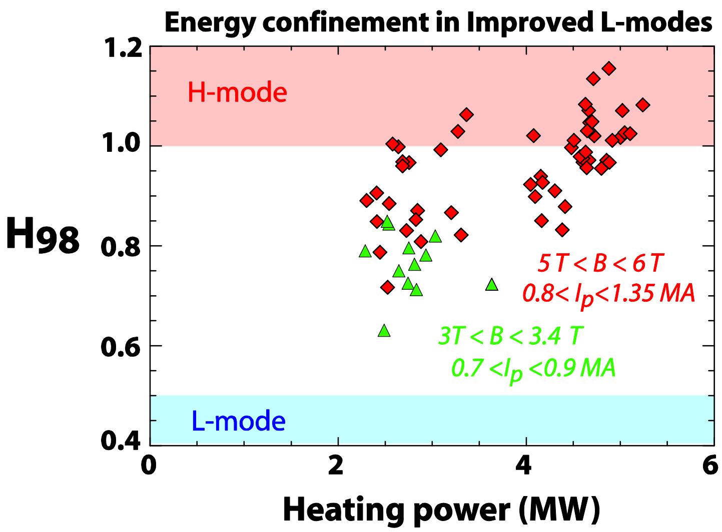

Fig. 9.2: Energy confinement normalized to the H-mode confinement scaling ITER98y2 for a broad set of I-mode discharges with varying B, I_p, density, shape and ICRF input power. The increase in H_{98} with heating power is encouraging and indicates less degradation with power than is typical of L or H-modes.

As input powers (6 MW ICRH and up to 2.5 MW LHCD) and pulse lengths (up to 5 s) increase in C-Mod advanced scenarios, power handling, already a significant issue, will be an increasing challenge. Integrating means of controlling and mitigating heat loads into these scenarios is not only extremely relevant for future devices, it is becoming necessary in C-Mod experiments. Without due care, Mo or W tiles and other components can and do melt. Experiments in 2009 with impurity seeding are extremely promising in this regard. Argon seeding has been used in both H-mode and I-mode discharges, initially motivated by ITER requests, since ITER intends to rely on this technique to reduce divertor heat loads and needs to know the effect on performance. It was found that not only did this technique notably reduce divertor tile heating, in both regimes, it had beneficial effects for high power ICRF operation, reducing impurity injections and faults. The net effect was to reduce plasma performance very little, or in some cases even improve the plasma stored energy. This technique will certainly be assessed and exploited in future advanced scenarios.

In parallel with the experimental program, significant progress has been made in simulation, both in advancing the state of the art in RF modeling and in enhancing capabilities for integrated scenario modeling. The extension of the GENRAY ray tracing model to include a more realistic scrape-off layer model, discussed in detail in Section 6 is particularly significant. Refraction of waves before entering the core plasma, or on multiple reflections, combined with collisional absorption, could explain the observed decrease in non-thermal signatures at high density, and suggests several avenues to increase the efficiency. It also, however, presents a concern and requires revisiting and potentially modifying integrated scenario projections and targets. The rapidly evolving state-of-the art in LHCD modeling increases the priority on coupling the latest codes to integrated models. An important step in this direction in FY09 was linking GENRAY to TRANSP. The next step will be to include the CQL3D Fokker-Planck code for more accurate prediction of driven current. Coupling TRANSP to TSC, which is used for most C-Mod scenario predictions and for experimental interpretation, is also highly important.

9.2. Research Plans for FY10-12

Commissioning of the new LH-II launcher in FY10 will be critical for the experimental program on advanced scenarios. As discussed in Section 6, assessing the power limits with the novel design will be highly informative. It is anticipated that reduced system losses and increased power handling will allow higher coupled LH powers than in past C-Mod campaigns. Equally important for advanced scenarios is assessing the efficiency at high density (> 10^{20} m^{-3}), in higher temperature and confinement plasmas which are more relevant to advanced scenarios. These experiments will require combining with significant ICRF heating, and will include density and field scans in L-mode, H-mode and I-mode, the latter building on the results described above. In particular, we will assess the efficiency in conditions with high single pass absorption, where it is expected that any rays which reach the core plasma should damp and drive non-thermal electrons and current. This new physics is of high importance not only for C-Mod advanced scenarios but potentially for ITER and for proposed future steady state devices (for example the Fusion Development Facility) and for DEMO. In recognition of this, a new ITPA Joint Experiment 5.3: *Assessment of lower hybrid current drive at high density for extrapolation to ITER advanced scenarios* was started in 2009 which will be led by MIT and will also involve experiments on FTU, Tore-Supra, JET and EAST, and modeling from the associated groups. C-Mod experiments will begin in FY10 and extend into FY11, with new physics results directly influencing the scenarios program. Because of this, the details and priority of FY11-12 experiments need to remain flexible.

Integrated scenario modeling to date, which has been used to prepare, plan, and interpret experiments at C-Mod, has not included the SOL refraction effects which are now thought to be important at high density. The main codes employed, which are also used for ITER and other devices, are ACCOME and TSC, which uses LSC for lower hybrid. As improved LH models become available, it is very important that they be coupled to integrated models. In the near term (FY10), we plan to couple GENRAY-CQL3D directly to TRANSP and link TRANSP to TSC, as part of the PPPL collaboration. C-Mod modelers at MIT and PPPL are also very active in the SciDAC and SWIM projects. In FY11 and beyond, the Integrated Plasma Simulator should provide an excellent platform for coupling various RF codes, as well as equilibrium codes and physics based transport models.

The improved integrated models will be used to revisit planned advanced scenarios. In particular, H-mode plasmas on C-Mod have densities exceeding 10^{20} m^{-3}, largely determined by pedestal particle transport, so it is likely that current drive efficiency will be reduced to some extent. We now have a much better experimental basis for the plasma profiles (n, T, etc) to expect for given global plasma parameters, and will use these to make more realistic predictions for a range of potential scenarios. Efficient and flexible simulation tools will be important. In addition, we will simulate scenarios based on adding LHCD to I-mode plasmas, which as noted have high temperature and confinement but much lower and easily controlled density, to compare the expected non-inductive current drive. In each regime it is expected that there will be tradeoffs between bootstrap and non-inductive current drive as density, field, lower hybrid $N_{//}$ spectrum and other parameters are varied. Since experimental time is limited, scenario modeling will play a key role in guiding parameter selection.

While the details remain to be determined, we propose in FY10- FY12 to carry out experiments using LHCD in both H-mode and I-mode targets. The goal of H-mode experiments will be to assess whether the so-called 'hybrid' or 'advanced inductive' scenario can be created using LHCD to modify j(r), maintaining central safety factor near or slightly above one. This is of high importance for ITER and is a C-Mod Research Goal. Experiments will build on encouraging results obtained in FY08, in collaboration with AUG researchers, in which LHCD was applied in the current ramp and ICRH later used to trigger H-modes. It was found that sawteeth could be reliably suppressed until the L-H transition, demonstrating q(r) modification. In addition, edge pedestals and global confinement were increased, transiently, in the early phase of H-modes with the LH-modified targets. It is not yet clear whether the q(r) modification can be sustained into the H-mode.

The C-Mod goal dovetails very well with current priorities of the ITER project and ITPA, in particular 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 – in absence of MHD instabilities, the effect should be much more clearly discernable. Another reason for the high interest by ITPA in the C-Mod contributions is that of "ITER relevant restrictions", by which they principally include low torque and equilibrated ions and electrons. While other experiments will have to gradually move towards such conditions by making use of non-standard heating mixes, C-Mod naturally operates torque-free and with T_i/T_e near one in all H-mode experiments. Successful demonstration of improved confinement regimes in these conditions would increase confidence that they can indeed be achieved in ITER. These experiments will also contribute to related ITPA experiments PEP-20: *Documentation of the edge pedestal in advanced scenarios* and TC-5: *Determine transport dependence on T_i/T_e ratio in hybrid and steady-state scenario plasmas*

A closely related avenue of research will be studying the interaction of LHCD with H-mode pedestals. These experiments will follow highly interesting results in FY08 in which application of lower hybrid power to established H-modes, which had steady densities as low as could readily be achieved using shape and cryopump control techniques. Surprisingly, a further reduction in pedestal and core density occurred, and pedestal and core temperatures increased. This effect, which we seek to understand, offers potential synergism in that the LH waves modify the plasma in such a manner as to increase the accessibility and efficiency of LH waves, with very positive implications for scenarios.

In other experiments, LHCD will be added to high performance I-Mode targets, with higher edge and core temperatures and lower densities. These conditions should be conducive to maximum LH absorption, efficiency and driven current, simulating more closely the plasma conditions expected on ITER and providing a key test of LH models in this regime. A goal will be to combine strong lower hybrid current drive with significant bootstrap current. While simulations remain to be carried out, it is possible that magnetic shear can be sufficiently modified to affect core plasma transport.

Other proposed experiments will explore the generation of "current holes" by LHCD in the current rise, as was done for example on JT60-U. Models show this should be feasible with moderate LH power, consistent with one launcher, using variable phasing to produce a compound $N_{//}$ spectrum (Fig. 9.3). As noted in Section 6, new analysis indicates that shear reversal may already have been obtained in some prior LHCD-driven

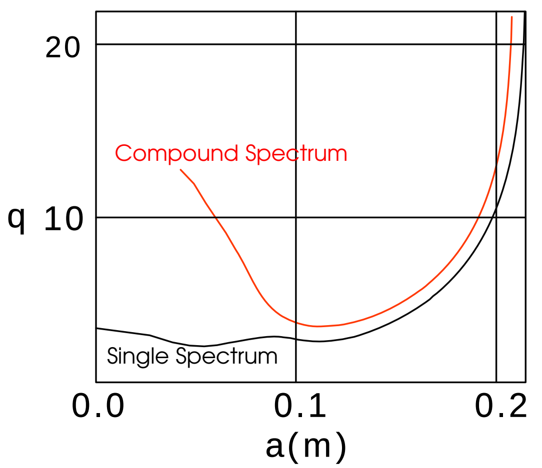

Fig. 9.3: Comparison of calculated current profiles produced by a compound spectrum ($N_{//}$=2.3 and $N_{//}$=3.1) and a single spectrum ($N_{//}$=2.3). LHCD power of 1.3 MW was assumed. (from S. Shiraiwa, APS 2008)

discharges.

The relative advantages of operation at different magnetic fields will be assessed in both simulation and experiment. LH accessibility is optimum at high B, up to 8 T on C-Mod, which can be heated with 80 MHz ICRH using $D(He^3)$ minority or mode conversion heating. For high bootstrap fraction, one would need to reduce I_p and increase q_{95}, a potentially attractive mode of operation for a reactor well away from stability limits. At the other extreme, as demonstrated in 2009 experiments, operation at 3.5 T and lower q_{95} would enable increased β_N and could allow assessment of stability limits. TSC-LSC modeling indicates that efficient off-axis current can be obtained with $N_{//}$=3.0. This will be assessed experimentally and using improved LH models. Results and simulations will inform the decision to upgrade remaining ICRF source to be tunable from 50-80 MHz in FY12. This would be necessary to allow operation with full power and high β at 3.5 T.

While, clearly, many important experiments will be enabled by the upgraded LH launcher, which should couple up to 1.5 MW power at least for shorter pulses, it must be recognized that increased LH power (4 MW source) will be required to achieve the fully non-inductive advanced scenarios which are the main aim of this C-Mod integration thrust. The planned additional klystrons are essential, and based on power handling in other LH launchers it is expected that a second launcher will be needed to reliably couple this power for multi-second pulses. Its design will be informed by the performance of LH-II, and its installation is planned for FY12. Once commissioned, coupled powers of 2.5 MW are expected. This will open up opportunities for strong current profile modification and high non-inductive fraction. At this point emphasis on active control of current, density and other plasma profiles will increase. Development of control actuators, sensors and algorithms suitable for reactor-relevant parameters, in particular high density and field, is the subject of ReNeW Thrust 5, to which we expect to make key contributions.

Given C-Mod's unique, DEMO level heat fluxes and world-class divertor research program, development of compatible solutions to plasma sustainment and edge power handling will continue to be a major component of our scenarios research program. The importance of this cross-cutting challenge is increasingly recognized in the world and US fusion programs, as evidenced by attention in the Research Needs Workshops. Beginning in FY11, we expect to begin integrating advanced scenarios with radiative divertors, following the promising experiments in baseline H-mode and I-mode scenarios. The upgraded DEMO-like divertor in FY12 will open up new opportunities, in particular enabling longer pulse durations at full input power. This research directly informs and supports ReNeW Thrust 12.

Overall, the C-Mod research in advanced scenarios is well situated to help develop and validate the tools and scenarios which will be needed in ITER as well as for any future steady state device at high density and high power, conditions essential to fusion energy and even for a fusion nuclear science facility. We will, though development of and comparison with integrated scenario models, place a strong emphasis on developing the predictive capability to give confidence in the needed experimental steps.

10. Theory and Simulation Support for Alcator C-Mod

10.1. Introduction

Theory and simulation support for Alcator C-Mod research consists of active collaborations through the International Tokamak Physics Activity (ITPA), the Burning Plasma Organization (BPO), formal collaborations with Princeton Plasma Physics Laboratory (PPPL), University of Texas at Austin, Asdex Upgrade, the various SciDAC Centers, and the Prototype Fusion Simulation Projects. There also exist important collaborations between individual C-Mod personnel and theorists within MIT (PSFC Theory Group) and outside MIT. An overarching goal of theory and simulation research on C-Mod is to provide support for interpretation and guidance of experiment as well as for program and experimental planning.

10.2. Research Highlights from FY09-10

10.2.1. Loki Computing Cluster

A two stage upgrade of the Loki parallel computing cluster was completed in 2009 which increased the number of cores from 260 to 600 and tripled the memory per node from 4 GB to 12 GB. The cluster has been a valuable tool for the interaction of theorists and experimentalists at the PSFC with near-continuous availability, excellent vendor support, 3.2 million CPU hours used in the past two years by 20 active users, and 49% of total available CPU hours utilized. Over half (55%) of the CPU usage has been for research directly in support of C-Mod. An array of advanced parallel simulation codes are run routinely on Loki and the cluster has been the workhorse computational resource for several PhD theses in the past two years including Liang Lin, Leonardo Patacchini, and Brock Bose.

10.2.2. Turbulent Transport Theory and Simulation

During FY09 members of the PSFC Theory Group engaged in an effort to verify and improve the physics in the gyrokinetic code GS2. Dr. D. Ernst and co-workers carried out a comparison of the particle-in-cell code GEM and the continuum GS2 code [10.1]. This work was primarily a study of zonal flows in TEM turbulence, but was also the first comparison of nonlinear particle and continuum gyrokinetic approaches with full electron and ion dynamics. The ion and electron heat and particle fluxes, fluctuation spectra, and zonal flow shearing rates were compared from the two codes as function of the parameter η_e, where $\eta_e = d(\ln T_e) / d(\ln n_e)$ The fluxes and wavelength spectra for pure TEM turbulence were found to largely agree. Interestingly, it was observed that convergence was poor for values of $\eta_e > 4$ suggesting that larger simulations (box sizes) are needed for the large η_e values found in typical H-Mode plasmas. This type of verification activity is considered to be an important step before validation against experiments can be trusted. Dr. Ernst also implemented a new collision operator in GS2, which physically accounts for energy scattering and finite Larmor radius effects [10.2]. Simulations with the new collision operator were found to damp short wavelength TEMs in C-Mod [10.2]. New conserving terms, that preserve the H-theorem, were then added to the same operator in work by Barnes, including Ernst [10.3]. Dr. Ernst and Dr. P. Catto developed a new Monte Carlo implementation and more elegant formulation of the H-theorem preserving operator, suitable for use in gyrokinetic particle codes [10.4].

Professor M. Porkolab led an extensive validation activity in which his student Dr. L. Lin performed nonlinear GYRO simulations to investigate the role of ITG, TEM and ETG fluctuations on turbulent transport in L- and H- mode plasmas and to investigate the role of the ITG mode in the formation and control of internal transport barriers (ITB's) [10.5]. This work also included a comparison of measured fluctuation spectra with simulated spectra from a synthetic diagnostic code for Phase Contrast Imaging (PCI). Dr. Lin and Professor Porkolab worked in collaboration with Drs. R. Waltz and J. Candy from General Atomics and Dr. D. Mikkelson from PPPL. The importance of ITG turbulence in saturated ohmic and H-mode plasmas was verified; where it was found that the absolute fluctuation intensity agrees with simulation within experimental error. Simulated profiles of χ_{eff}, χ_e, and χ_i were found to agree with experiment after varying the ion temperature gradient scale length parameter (a/L_{Ti}) by no more than 20%. In the area of validation, an important disagreement between experiment and simulation was found in low density C-Mod plasmas in the linear ohmic regime, where electron transport is known to dominate ($\chi_e >> \chi_i$) from transport analyses of those discharges, yet GYRO simulations indicate that $\chi_i > \chi_e$. It was found that TEM turbulence was unlikely to be important given the measured density and temperature profiles. Also, inclusion of high-k ($k_\theta \rho_s \leq 8$) turbulence was not found to raise χ_e to the experimental level. Furthermore, GYRO simulations showed that the contributions from the electromagnetic fluctuations are negligible in low-β C-Mod Plasmas. At present, Prof. Porkolab is collaborating with Prof. P. Diamond at UCSD to investigate whether or not this discrepancy can be understood in terms of the impact of the ohmic electron drift.

10.2.3. Macroscopic MHD

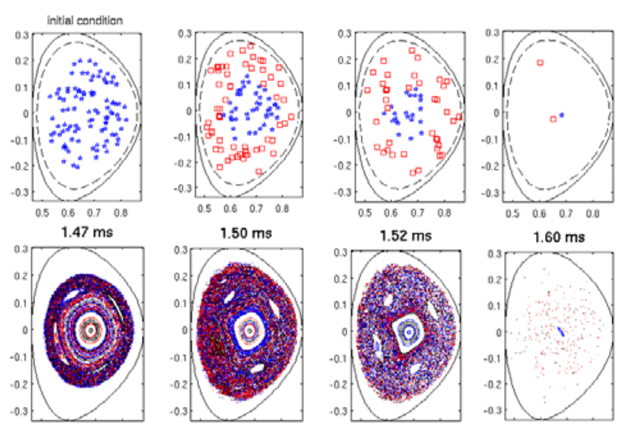

Fig. 10.1: NIMRAD simulation of runaway electron suppression in C-Mod using MGI. Initial 100 keV electrons shown in blue and relativistic electrons shown in red.

During the past year, Dr. V. Izzo from UCSD in collaboration with Prof. D. Whyte succeeded in simulating experiments on Alcator C-Mod where massive gas injection (MGI) was used to terminate a runaway population of electrons that was seeded by driven lower hybrid (LH) waves [10.6]. This work provides a new set of experimental and modeling tools to quantify the dynamics of stochastic transport of runaway electrons.

Simulations of the C-Mod experiments with the nonlinear MHD code NIMROD are in agreement with predictions of the electron Fokker Planck code CQL3D, where it was found that rapid loss of the relativistic electrons due to stochastic transport at $\delta B_r/B \sim 10^{-3}$ effectively suppresses runaway growth, obviating the need to attain the Connor-Hastie-Rosenbluth collisional limit to suppress runaways. The evolving, robust MHD driven by massive edge radiation is found to be the same process that de-confines the electrons due to growing stochasticity, thus linking disruption mitigation effectiveness to runaway suppression. A simulation of the MGI of LH seeded runaway electrons using NIMRAD (NIMROD plus the radiation code KPRAD) is shown in Fig. 10.1. Here NIMRAD was used to evolve the 3-D magnetic field topology and trace electrons were followed in the changing magnetic field geometry to study electron transport. The runaway population corresponds to 100 kA of LH current and electrons are considered “lost” when their position exceeds r/a=0.95 (by stochastic transport only). The timescale for the loss agrees well with experiment and the stochastization of field lines in Fig. 10.1 is apparent as time evolves.

10.2.4. Pedestal and Plasma Boundary

An active collaboration exists between Dr B. Lipshultz at C-Mod and Dr. P. Catto and co-workers that focuses on radial electric field and flow behavior in the pedestal of C-Mod. The experimental observation on C-Mod [10.7] is that the ion diamagnetic flow in the pedestal of density width of $\sim \rho_{\theta i}$, is $V_{dia}/v_{ti} \sim \rho_{\theta i} n^{-1}(dn/dr) \sim 1$. Thus one expects the $\mathbf{E} \times \mathbf{B}$ and diamagnetic terms to cancel to lowest order since the ion flow is subsonic in C-Mod. Measurements on C-Mod are consistent with this theoretical picture where $E_{radial} \approx (en)^{-1}\, dp_i/dr$, and the pedestal is held in equilibrium by electrostatically confined ions and magnetically confined electrons.

More recently Dr. Catto has worked with graduate student K. Marr to interpret his experimental measurements of poloidal ion flow on C-Mod in terms of neoclassical theory. The result of their analysis is that the measured poloidal flows are consistent with the Pfirsch-Schluter expressions, whereas standard theoretical predictions for the banana regime disagree with the measured flows. Graduate student G. Kagan working with Dr. Catto has found that the radial electric field can cause finite orbit effects that can change the sign of the poloidal flow, making banana predictions consistent with the measurements. In addition they found quite interestingly that the bootstrap current density in the pedestal can also be modified as well by this effect. Graduate student M. Landreman and G. Kagan working with Dr. Catto have found that the electric field and orbit squeezing can modify zonal flow behavior and neoclassical ion heat flux in the edge.

10.2.5. Wave-Plasma Interactions

Accurate calculation of 3-D ion tail distributions is important for several key physics areas on C-Mod: MHD studies of Alfven cascades, sawtooth modification experiments via energetic minority tail generation, and also for transport analysis of standard and improved confinement (ITB) regimes, and analysis of ICRF heated plasmas with significant toroidal rotation. During FY09 significant progress was made in the development of a simulation capability to include finite ion orbit width effects in energetic minority ion heating by ICRF fast waves. Through research conducted in the RF SciDAC Center by Drs. M. Choi, D. L. Green, and L. A. Berry an advanced full-wave solver (AORSA) was combined with two Monte Carlo orbit codes – ORBIT RF [10.8] and sMC [10.9]. The full-wave solver uses the full particle lists from the Monte Carlo codes to re-evaluate the plasma response through a module named "p2f" [10.9]. ORBIT RF employs a simplified RF operator that is approximately valid for minority heating, whereas the sMC code uses the complete 4-D RF diffusion coefficient to push resonant ions, thus providing an exact coupling between the particle dynamics and full-wave solve. Plans to verify and validate this capability will be discussed in Section 10.2.5. Another area of significant progress was in the benchmarking of the AORSA all-orders spectral solver and the ion FLR semi-spectral solver TORIC. A long standing error in the electric field normalization of the TORIC code was resolved and the new wave field amplitudes and patterns from TORIC and AORSA have been shown to be in close agreement for minority heating and mode conversion schemes in Alcator C-Mod [10.10].

During FY09 major progress was made in developing and applying an advanced LH simulation capability consisting of a full-wave spectral solver (TORIC-LH) and the electron Fokker Planck code (CQL3D) [10.11]. This work was conducted in collaboration with Dr. R. Harvey at CompX and Drs. E. Valeo and C. K. Phillips at PPPL.

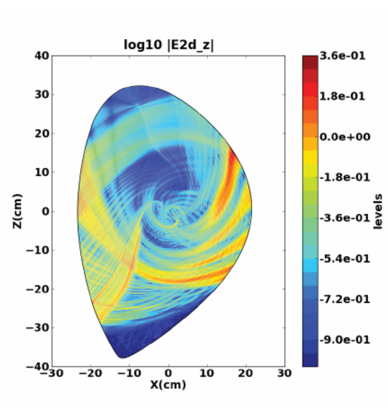

Fig. 10.2(a): LH wave fields from a TORIC-LH simulation for Alcator C-Mod [$n_{//}$ = 1.55, $n_e(0) = 7 \times 10^{19}$ m^{-3}, $T_e(0)$ = 2.33 keV, f_0=4.6 GHz, N_R = 980 N_m = 1027].

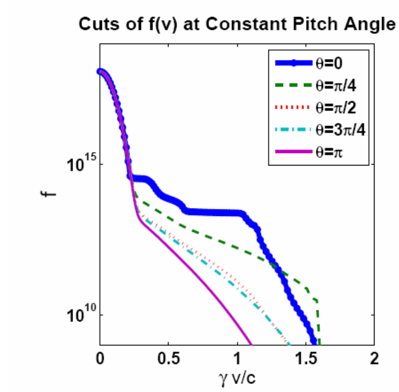

Fig. 10.2(b): Electron distribution function versus relativistic velocity at various pitch angles at a radial location of r/a=0.5, based on the quasilinear flux evaluated from the wave fields in Fig. 10.2 (a).

Extensive verification of the full-wave solver was carried out using ray tracing calculations and spectral convergence studies. This resulted in the discovery of a long standing error that was corrected in the cross-term of the electron FLR current that represents the combined effect of transit time magnetic pumping and electron Landau damping. First ever self-consistent nonthermal electron distributions were computed using this combined model in the weak and strong single pass damping regimes. Shown in Figs. 10.2(a) and 10.2(b) are the wave fields and electron distribution function for a TORIC-LH / CQL3D simulation of weak damping in Alcator C-Mod [$n_{//}$ = 1.55, $n_e(0)$ = 7×10^{19} m^{-3}, $T_e(0)$ = 2.33 keV, f_0=4.6 GHz, radial resolution of N_R = 980, and poloidal mode resolution of $N_m = 1027$]. The results shown in Fig. 10.2 are at the fourth iteration between the full-wave code and the Fokker Planck solver. At each iteration, the non-thermal electron distribution from CQL3D is used to re-evaluate the plasma response in TORIC-LH and the RF diffusion coefficient based on the most recent wave fields from TORIC-LH is used in CQL3D to advance the distribution function.

Another significant advance made in the area of LH simulation was the incorporation of a scrape-off-layer (SOL) model in the GENRAY ray tracing code that is used in conjunction with CQL3D. This physics improvement was made possible through a collaboration with Drs. R. Harvey and A. P. Smirnov at CompX. Lower hybrid ray tracing has traditionally entailed launching rays inside the last closed flux surface (LCFS) and calculating power absorption and current drive from ray increments, as the rays propagated in the plasma and reflected from the region of the LCFS. This model was proving inadequate for C-Mo as the combined CQL3D Fokker-Planck/GENRAY ray tracing model was grossly overestimating current drive and hard x-ray spectra as the density was increased [10.12]. Stopping the rays after one pass of the plasma brought the model into much better agreement with experiment. Consequently, the facility for LH ray propagation and collisional damping in the scrape-off-layer outside the LCFS was added to GENRAY. The main components of this model enhancement were (1) rays propagate outside the LCFS, (2) density and temperature profiles are specified by poloidally dependent exponential decay lengths as a function of perpendicular distance from the LCFS, and minimum density/temperatures, and (3) detailed specification of the vacuum vessel with reflection near the vacuum by additional reduction of the density. Figure (10.3) shows LH rays including propagation in the SOL. The result, with CQL3D modeling of the distribution function inside the plasma is that the hard x-ray signal and driven current density are reduced by orders of magnitude above $\bar{n}_e \approx 1 \times 10^{20}$ m^{-3}, bringing the simulation into substantial agreement with experimental observations [10.11], thus providing an explanation for the LHCD density limit observed in C-Mod.

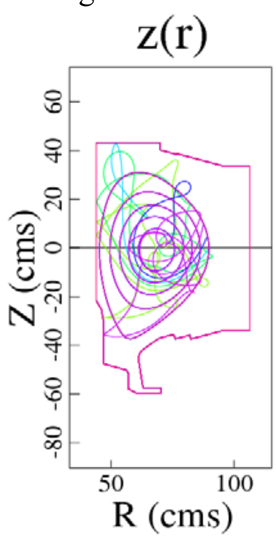

Fig. 10.3: LH rays launched from the plasma periphery and undergoing collisional absorption in the SOL.

10.2.6. Integrated Scenario Modeling

Time dependent integrated scenario simulations of LH current profile control experiments were on-going during FY09 and were carried out primarily by Dr. C. Kessel (PPPL). These simulations currently use TSC to perform discharge evolution using 1.5D transport equations, a free boundary MHD solver, and the LSC code for LH current drive. Model profiles are used for ICRF heating and the Tang-Coppi micro-instability based thermal diffusivity is used for transport evolution. At present, Dr. Kessel is working with Drs. P. Bonoli, D. Batchelor and other members of the SWIM Prototype FSP Project to simulate a set of LH current profile control experiments in which sawteeth were delayed for longer periods of time as the level of LH current was increased (through a reduction in the electron density at fixed LH power). After preliminary analysis of these discharges with TSC and LSC, an initial "state file" file from the TSC simulation will be used by the SWIM framework to perform the same simulations with more detailed models for LH current drive based on the GENRAY ray tracing code, an adjoint code for the LH current drive calculation, and finally with the Fokker Planck code CQL3D. This work is discussed in more detail in Section 10.2.6.

10.3. Research Plans for FY10 – FY12

10.3.1. Loki Computing Cluster and Upgrade

During FY10 and FY11 we plan to maintain stable, ongoing cluster operations. Possible upgrades of the disks and operating system will be considered. We will also begin future technology/system configuration research in preparation for the eventual replacement of Loki. Preliminary research indicates that if 12-16 core processors are available in 2012, this would allow replacement of Loki with a new 2400-2500 core cluster for approximately $480,000. Procurement of such a cluster would have to start in the FY12 time frame.

10.3.2. Turbulent Transport Theory and Simulation

During FY10 we plan to continue investigations of ITG, TEM and ETG roles in turbulent transport in L- and H- mode plasmas with equilibrated T_i and T_e. This will be carried out from the experimental side by graduate students working under Prof. M. Porkolab and by Dr. D. Ernst of the Theory Group. Both the GS2 and GYRO codes will be used, including comparisons with synthetic diagnostics for PCI that have been implemented for each code. Dr. Ernst will continue to work with Dr. C. Fiore to study ITB formation, and will conduct his own experiments on ITB control with on- and off-axis ICRF heating. These planned experiments will use vertical jogs to nearly double the effective number of channels for Thomson scattering, and modulated on-axis ICRF heating to destabilize TEMs in the ITB, allowing core and edge signals to be separated in the chord-integrated phase contrast imaging signal. Dr. Ernst will also work with Dr. M. Greenwald to study the relationship of particle and impurity transport in C-Mod to drift wave turbulence (i.e., ITG and TEM driven). This work will involve performing gyrokinetic (GS2) simulations of impurity transport and comparing with the results from laser blow-off experiments conducted by graduate student N. Howard.

During the FY11-FY12 period we expect to carry out transport and gyrokinetic analysis of discharges in C-Mod with current and pressure profile control. We plan to assess the role of LH current drive in ITB formation via magnetic shear modification as well as the role of LHCD in the modification of core plasma rotation and pedestal density and temperatures. This work will entail performing gyrokinetic analyses of C-Mod discharges with LHCD, including the local effect of the shear profile, electron/ion temperature ratio, and loop voltage, and will involve a collaboration between Dr. D. Ernst and C-Mod experimentalists [Drs. C. Fiore, R. Parker, M. Porkolab, J. Rice, and J. R.

Wilson (PPPL)]. Proper interpretation of shear flow generation by mode converted ICRF waves will also be a high priority in the FY10-FY12 period. Dr. Y. Lin at C-Mod will continue to work with Dr. E. F. Jaeger (ORNL) and Dr. J. Myra (Lodestar) to validate theories for shear flow generation, using mode converted wave fields from advanced ICRF solvers such as TORIC and AORSA.

During FY10-FY12, new plans will be implemented for understanding core transport physics on C-Mod through gyrokinetic code validation. This effort will be led by Prof. A. White in collaboration with Drs. P. Phillips (UT-Austin) and Dr. A. Hubbard. The design of a Correlation ECE (CECE) diagnostic for C-Mod will be revisited using nonlinear / linear GYRO simulations to support planning of new CECE experiments, measurements of electron temperature fluctuations, and also for eventual modeling and analysis of those experiments. The GYRO code will be tested against experimental results via a synthetic diagnostic code for CECE. A local, core T_e fluctuation measurement will be useful for thermal transport studies. Professor White also plans to implement a Doppler reflectometry diagnostic on C-Mod to measure density fluctuations at intermediate wavenumber. Results from this diagnostic can be compared directly to GYRO predictions, thus providing insight into thermal and momentum transport. It is anticipated that much of the linear and nonlinear GYRO simulation work will be carried out on the local Loki computing cluster. It is also expected that during the FY11-FY12 time frame the trapped gyro-Landau fluid model TGLF will be validated against C-Mod discharges as part of a collaboration involving G. Staebler (GA), J. Hughes, Y. Ma, A. White, and D. Ernst. These studies will involve linear stability analyses with TGLF that could also be performed on the Loki cluster.

10.3.3. Macroscopic MHD

In the near term (FY11-FY12) we plan to continue to model experiments with MGI of LH seeded runaway electrons using the combined NIMROD and KPRAD codes. Further experimental and simulation scoping should lead to a much better predictive capability for runaway electron suppression in ITER.

A second major area of emphasis in MHD for FY11-FY12 will be to understand the role of realistic ICRF-generated fast ion distributions in excitation of unstable Alfven Eigenmodes (AE's) and Alfven Cascades. This will require better implementation of fast ion distributions from the combined zero orbit width AORSA-CQL3D and TORIC-CQL3D models in the kinetic MHD code NOVA-K and will be done in collaboration with Drs. D. Batchelor and E. F. Jaeger from ORNL and Dr. N. Gorelenkov from PPPL. As fast ion distributions from the full-wave / Monte Carlo codes become available they will also be implemented in codes such as NOVA-K. Part of the validation process will be to compare synthetic diagnostic code predictions for mode stability based on NOVA-K with experimental data.

During FY10-FY11 we also plan to simulate sawtooth modification experiments on Alcator C-Mod that were performed with localized ICRF mode conversion current drive (MCCD). This will be done by Drs. P. T. Bonoli and J. C. Wright in collaboration with the SWIM FSP Project. The parallel framework developed within SWIM will be used to simultaneously execute the TSC code to evolve the background plasma and the TORIC and AORSA codes to evaluate the MCCD, based on an adjoint calculation of the current drive. The TSC code employs the Porcelli model to trigger the crash of sawteeth so that these simulations could provide a test of that model by comparing directly to the observed sawtooth period in the experiments. The simulated electron deposition profiles associated with the MCCD can also be compared with the electron deposition profiles inferred from an RF power modulation technique, thus providing a further check on the simulation predictions. In FY12 we also plan to have a simulation capability in place through the SWIM framework to assess sawtooth stabilization experiments on C-Mod through ICRF-generated minority tails (energetic particle stabilization). These simulations would again use TSC to evolve the background plasma and the AORSA/TORIC full-wave solvers will be used in conjunction with the Fokker Planck code CQL3D to simultaneously evolve the ICRF-generated ion tail in time. By FY12, the SWIM Project is expected to have some MHD simulation capability in place to test for sawtooth stability in the presence of energetic ion tails and the plan would be to test that model against the C-Mod experiment.

10.3.4. Pedestal and Plasma Boundary

During FY10-FY11 the PSFC Theory Group (Dr. P. Catto and graduate students) will continue to use measurements from C-Mod to test theoretical understanding of pedestal formation and the radial electric field. In addition we will begin to use computational tools to enhance our physical understanding of the pedestal and contribute data for validation of the newest edge codes. In particular we will continue ELM/EDA studies with the ELITE and M3D codes. In collaboration with Dr. L. Sugiyama, M3D simulations will be performed of the EDA H-mode edge in C-Mod in an attempt to identify the quasi coherent mode (QCM) and to test scalings for the QCM if the mode is seen in the simulations. In collaboration with Dr. P. Snyder from GA, C-Mod data will be used to validate the EPED and EPED1 reduced models for the pedestal height and width. Over the longer term (FY11-FY12) we plan to work with Drs. C. S. Chang and L. Sugiyama from the Center for Plasma Edge Simulation (CPES). The advanced particle-in-cell code XGC0 will be used to simulate pedestal transport in the C-Mod edge, with 3-D electromagnetic turbulence calculated by XGC1. We will also attempt to take advantage of the integrated work flow tool (the Kepler framework)) for simulating complex, time-dependent edge phenomena such as the ELM cycle and L-H transitions.

10.3.5. Wave-Plasma Interactions

During the period from FY10-FY12 we plan to conduct an extensive validation study with our modeling capability for simulating ICRF-generated ion tails. The first phase in FY10-FY11 will involve comparing the predictions from zero orbit width models such as CQL3D with experiment. Synthetic diagnostic code predictions of CNPA signals based on the CQL3D tail predictions will be compared with experiment. In FY11-FY12 we should also have synthetic diagnostic code predictions based on non-thermal ion distributions from models that include finite ion orbit width effects through the sMC and

ORBIT RF Monte Carlo codes. By comparing the simulation results with and without orbit effects included it should be possible to assess where the zero ion orbit width approximation breaks down and also assess the importance of these effects in C-Mod. This work will be done by the RF SciDAC Center (Drs. P. T. Bonoli, D. Batchelor, and L. Berry) in collaboration with Dr. S. J. Wukitch, Professor R. Parker, and graduate student A. Bader. A novel approach for studying finite ion orbit effects is also being pursued with Drs. R. Harvey and Y. Petrov at CompX. In this work, the DC (Lorentz force orbit Diffusion Coefficient calculator) is being applied to the minority ICRF heating scheme in C-Mod in order to verify the ICRF quasilinear diffusion model used in AORSA for prediction of ICRF-generated nonthermal tails. Recent results using the full spectrum of toroidal modes launched by the finite-length antenna indicate that reasonable accuracy is obtained with the computationally less demanding random phase - approximation quasilinear model.

Detailed validation studies of simulation capability of mode converted ICRF wave fields are also planned. The ultimate goal here is to have a predictive model for mode converted ICRF waves that can be used in mode conversion current drive and flow drive calculations. As described in the Recent Progress Section, both AORSA and TORIC have been used to simulate mode converted wave fields with very similar results. However, the simulated PCI signals from the AORSA and TORIC fields are much larger by at least a factor ten when compared to experiment [10.10]. During FY10 and FY11 we plan to investigate if there is missing physics in the full-wave codes such as parasitic losses or nonlinear effects. It may also be possible that proper accounting of the toroidal localization of the wave fields (a 3-D effect) is necessary. The sensitivity of the measured PCI signal to some of these effects can be tested quite easily using the synthetic diagnostic for the PCI.

During FY10-FY11, an aggressive validation activity is planned for our simulation capability in the area of LH wave physics. The combined full-wave / Fokker Planck code TORIC-LH and CQL3D will be validated by comparing simulated hard x-ray spectra and current density measurements with experiment. One goal here will be to understand the relative importance of diffraction, scattering, and toroidicity in the spectral broadening observed in the simulations and inferred from experiment. During this time period it is expected that a fully implicit 3-D ($v_\perp$, $v_{//}$, r) solver will have been implemented in CQL3D by Dr. R. Harvey at CompX and that he will provide support for implementing and using this new capability at C-Mod. In particular, this new solver should greatly simplify application of CQL3D to the study of radial transport of LHRF-generated fast electrons and the analysis of simulated hard x-ray spectra, since the fully implicit 3-D solver performs about 50 times faster. The combined full-wave / Fokker Planck LH simulations will also be carried out in 3-D by linearly superposing field reconstructions from individual toroidal mode components that comprise the LH launcher spectrum. This capability has been made possible by an improved parallel solver that was implemented in the TORIC code by graduate student J. P. Lee and that has resulted in at least a factor of five reduction in the matrix inversion time in TORIC [10.13]. This solver has already been used by to perform the first 3-D field reconstructions ever for an Alcator C size plasma that reproduces resonance cone behavior of the LH wave.

A substantial concern regarding use of LH current profile control for ITER is parasitic damping on fast alpha particles [10.14]. Fast ion tails produced in the C-Mod ICRF experiment can be used to simulate this effect. However, the present quasilinear model in CQL3D is not suited to ion damping at the very high cyclotron harmonics of LHRF experiments. A new high-harmonic ion quasilinear diffusion will be added to the code based on the "unmagnetized" diffusion in the perpendicular-to-B direction. The present model will find the simultaneous solution of the 2-D ($v_\perp$, $v_{//}$) bounce-averaged Fokker-Planck equations versus radius for both electrons and ions. The ion QL diffusion will treat both ICRF and LH diffusion and self-consistency with damping will be obtained by iteration. This will provide a new level of realism in the model compared to past 1-D ($v_{//}$) calculations [10.14]. Work for this model is to be completed in the 2010 time-frame by R. Harvey at CompX and will provide direction for future C-Mod ICRF/LH experiments.

In past work [10.15], a reduction in magnitude of the LH magnetized QL diffusion relative to the unmagnetized ion diffusion when the RF electric field exceeds a stochastic threshold was examined. It was found that even if the LH wave cannot reach the region of $(\omega / k_\perp) < 4\ v_{ti}$ that suprathermal ions, such as from the ICRF, can still be heated. Nevertheless, there remains substantial theoretical uncertainty in the appropriate LH ion diffusion coefficient for this interaction. During the period from FY11-12, Dr. R. Harvey at CompX will use the Lorentz force DC code to study LH ion diffusion, integrating ion orbits using LH fields from the TORIC-LH solver and the EFIT magnetic equilibrium reconstruction. This is a straightforward extension of present work (described above) in which ICRF diffusion coefficients are computed using wave fields from the AORSA solver and EFIT magnetic equilibrium reconstructions.

A significant effort is planned for FY10 – FY12 to address the problem of edge to core coupling and absorption in both the ICRF and LHRF regimes. This problem is sufficiently complicated and important that a number of simultaneous paths are planned. Dr. S. Shiraiwa of the Alcator Group working with Professor R. Parker and graduate student O. Meneghini plan to investigate coupling of an ICRF spectral solver such as the TORIC code with the commercially available finite element method (FEM) package COMSOL. This would allow effects of the actual ICRF launcher and vacuum vessel geometries to be coupled with the core ICRF wave propagation. This work would complement research being carried out by graduate student M. Garrett who is working with Professor Parker and Dr. Wukitch on a detailed representation of the new four strap ICRF antenna for C-Mod, also using the COMSOL package. That work is aimed less at the edge to core integration and more at studying the near field of the ICRF antenna, ICRF sheath generation and the role of slow wave propagation in the low density SOL. They anticipate this work to be started in FY10 and continue through FY12.

During the FY11-FY12 time frame we plan to couple the core spectral solver TORIC-LH with an FEM representation of the plasma edge and LH launching structure. This would be done in collaboration with Drs. P. Bonoli, J. C. Wright, and the RF SciDAC Center. Also, Dr. David Smithe at Tech-X, is developing models for ICRF and LHRF interactions in the plasma edge region by incorporating CAD file representations of the antennas in the particle-in-cell code VORPAL to simulate the wave fields. By integrating particle orbits, Dr. R. Harvey at CompX will use these fields to examine heating in the RF fields immediately in front of the ICRF and LH antennas. This research is planned for FY11-12.

The SciDAC Center for Wave Plasma Interactions also plans to investigate edge to core ICRF coupling through several approaches. First they plan to continue work in FY10 that is now partially completed where the TOPICA antenna code has been coupled to the TORIC code. TOPICA employs a boundary element method (BEM) in the form of Green's functions to represent the antenna structure and the coupling to plasma is done through an admittance matrix, with a realistic plasma load description such as TORIC. The work will be completed by Dr. J. Wright in collaboration with Dr. R. Maggiora from Torino and should provide an improved numerical description of the linear ICRF antenna coupling. In addition, a new FEM solver that accounts for misalignment between the metal wall and magnetic field has been developed by graduate student H. Kohno working in collaboration with Drs. J. Myra, D. D'Ippolito, J. C. Wright and Professor J. Freidberg. The code employs a metal wall boundary condition with a sheath boundary condition that is a nonlinear function of the electric field at the wall. This numerical model includes both the fast and slow waves and a parallel version of the code is under development to facilitate resolution of the slow wave and should be completed in FY10. By the end of FY11 it is expected that realistic launcher and edge geometries will be included in this code and that it will be coupled to the core spectral solver TORIC, possibly through an admittance matching technique. Finally we will study the possibility of doing a complete edge to core ICRF integration by developing the conductivity operator in the ICRF wave equation in an appropriate FEM basis. This approach is considered longer term and it is likely to not b completed until the end of FY12.

In the next two years Dr. A. Ram in the PSFC Theory Group will lead an effort to investigate the scattering of RF waves by density blobs. He has already developed a Fokker-Planck model for the scattering of RF waves by a random distribution of blobs in collaboration with K. Hizanidis [10.16]. They find that real space diffusion is important for electron cyclotron (EC) waves which could impact the use of EC waves in ITER for NTM control. They find that for LH waves, diffusion in wave vector space is important, which could lead to a spreading of the parallel wave vector. This effect will be assessed for LH wave propagation in Alcator C-Mod.

10.3.6. Integrated Scenario Modeling

The overarching goal of integrated scenario modeling for FY10-FY12 will be to carry out simulations in support of plasma control and advanced tokamak experiments on Alcator C-Mod with the most advanced physics components available. The approach taken in order to accomplish this will be to use the Integrated Plasma Simulator or "IPS" which is a parallel framework developed through the SWIM Prototype Fusion Simulation Project. During the past year Drs. P. Bonoli and J. Wright have developed expertise in using this framework which currently allows time dependent plasma simulations using TSC for plasma discharge evolution, GLF23 for transport evolution, and the AORSA and TORIC solvers for ICRF heating. By the end of FY10 it is expected that the GENRAY ray tracing code and the CQL3D Fokker Planck codes will be running as components of the IPS. This will make it possible in FY11-FY12 to simulate LH current profile control experiments in C-Mod as well as experiments where LHCD and ICRF heating are applied simultaneously. Lower hybrid current drive will be computed first using GENRAY alone with an adjoint calculation for the LHCD efficiency and then GENRAY will be iterated with the electron Fokker Planck code CQL3D to self-consistently evaluate LHCD including two-dimensional velocity space effects. In addition, the non-thermal minority tail produced during ICRF heating will be computed self-consistently by iterating AORSA (and then later TORIC) with the ion Fokker Planck version of CQL3D in a time dependent loop. These calculations will be possible with short (less than one day) turn around time because the IPS framework allows concurrent execution of parallel simulation codes. These types of simulations will make it possible to scope parameter space for the most promising scenarios for LH current profile control of sawteeth and internal transport barriers. In addition, it will be possible to validate the simulated LHRF-generated and ICRF-generated non-thermal tails against experiment through comparison with synthetic diagnostic codes for hard x-ray spectra and neutral particle analyzer counts.

Finally as part of an integrated effort to understand the interaction of RF waves with MHD phenomena, Dr. J. Ramos of the PSFC Theory Group is working on the development of kinetic closure relations for coupling RF and MHD codes, as part of the SWIM Project. The essential problem here is that spatial regions exist near rational flux surfaces within which the RF waves, the kinetic evolution of the velocity distribution, and extended MHD phenomena are all tightly coupled. A closure scheme that would be applicable to sawtooth stabilization via energetic particles (ICRF) in C-Mod may be formulated by the FY12 time frame, but not yet solved numerically.

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[10.10] N. Tsujii, M. Porkolab, E. M. Edlund, Y. Lin, S. J. Wukitch, P. T. Bonoli, J. C. Wright, and E. F. Jaeger, "Experimental studies of ICRF mode conversion with phase contrast imaging and comparison with full-wave simulations", Bulletin of the American Physical Society **54**, 232 (2009).

[10.11] J. C. Wright, P. T. Bonoli, A. E. Schmidt, C. K. Phillips, E. Valeo, R. W. Harvey, and M. Brambilla, "An assessment of full wave effects on the propagation and absorption of lower hybrid waves," Physics of Plasmas **16,** 072502 (2009).

[10.12] G. W. Wallace, "Behavior of Lower Hybrid Waves in the Scrape Off Layer and Pedestal of a Diverted Tokamak", Doctoral Thesis, Department of Nuclear Science and Engineering (2009); also submitted for publication to the Physics of Plasmas (2010).

[10.13] J. P. Lee and J. C. Wright, "A versatile parallel block tri-diagonal solver for spectral codes", to be submitted to ACM Transactions on Mathematical Software (2010).

[10.14] E. Barbato and A. Saveliev, “Absorption of lower hybrid wave power by α-particles in ITER-FEAT scenarios”, Plasma Physics and Controlled Fusion **46**, 1283 (2004).

[10.15] C. F. F.Karney and N. J. Fisch, “Stochastic ion heating by a lower hybrid wave”, Physics of Fluids **21**, 1584 (1978).

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11. MDSplus

11.1. Recent Highlights

The major software development activity in recent years continues to be the support and enhancement of the MDSplus Data System which now runs on a wide variety of computing platforms. MDSplus is by far the most widely used data system in the international fusion program. It is used in its entirety for the data acquisition and analysis systems for TCV (EPFL - Switzerland), RFX (IGI - Padua), NSTX (PPPL), Heliac (ANU - Australia),MST (U. Wisconsin), HIT, TIP, TCS and ZAP (U. Washington), PISCES (UCSD), CHS (NIFS - Japan), LDX (MIT), HBT-IP and CTX (Columbia U.) KSTAR (NFRI - Korea) and of course C-Mod. It is used to store processed data for DIII-D, for the collaborative data archives assembled by the ITPA, and for the inputs and outputs of several widely used codes, including EFIT, TRANSP and GS2. JET and ASDEX-Upgrade are using MDSplus as a remote interface to existing data stores and KSTAR has adopted it as a data acquisition engine for data stored in other formats. The result is a *de facto* standard which greatly facilitates data sharing and collaborations across institutions. At the same time, the breadth and variety of uses for MDSplus has increased the burden of supporting and documenting the system. A web site, http://www.mdsplus.org has been built to support the widespread user base and documentation is currently undergoing a major upgrade. The web site has been converted completely to a WIKI based system which makes it much easier to manage and update. MDSplus users worldwide are encouraged to participate in keeping the MDSplus online documentation accurate and up to date. Installation kits for all the supported platforms have been built and can be accessed this web site.

MDSplus software maintenance, bug fixes and ongoing support for off-site installations continues to be a major activity for the MDSplus development group. New support for data acquisition hardware was added, with emphasis on CPCI devices which have become increasingly popular. Support for new IEEE-1394 camera models has been added. Site specific work has been done for experimental groups at Columbia University, PPPL, University of Wisconsin, LDX (MIT/Columbia University), DIII-D, UCLA, University of Washington. A list of active fusion sites is appended at the end of this document.

The MDSplus system has also been enhanced with capabilities for handling storage of continuous data streams. Prior to these enhancements, MDSplus was suitable for storing only pulse based data but now it is possible to append data from continuous data sources. For example, much of the trending data such as the monitoring of the vacuum and cooling systems of the C-Mod experiment are now recorded using MDSplus. These enhancements will make it possible to use MDSplus on experiments with long pulse or continuous operation where it is impractical to wait for the pulse to complete before acquiring and analyzing the data. This feature is currently in use at the C-Mod experiment at MIT for storing continuous trend data and for storing data from high speed cameras.

A major development project initiated last year added a common objected-oriented API to the MDSplus system. The MDSplus system was originally designed with many object oriented features but the primary programming interfaces did not expose these features to the application developers. A set of object classes have been designed which represent the MDSplus pulse files and data items and which greatly increases the ease of developing applications when using object-oriented languages such as Python, Java, and C++. The implementation of this objected-oriented programming interface is near completion in the Python and C++ languages. Initial work has begun on Java and Matlab implementations. UML based design documents have been developed and web based documentation for the Python interface has been prepared. Some of the MDSplus internal code has been enhanced to enable the use of object-oriented languages for developing plug ins such as Python based data acquisition hardware support.

MDSplus installation kits are upgraded frequently and made available on the MDSplus web site for a number of computing platforms. To date there have been over 8000 downloads of MDSplus installation kits and over 1300 in just the past year.

The MDSplus developers have been included in design discussions for the ITER CODAC system to ensure that the system used on ITER includes many of the valuable capabilities provided by MDSplus.

11.2. Plans

Support for remote MDSplus sites will be increasing as the number of sites and the number of users increases. An ongoing effort to improve online documentation and to train local support staff at each of the major sites where the code is used will be made. The hope is to continue to hold an MDSplus users meetings on a biennial basis. The next meeting is scheduled in conjunction with the IAEA technical meeting in San Francisco in 2011. MDSplus software maintenance will continue to be a principle activity

We anticipate adding enhancements to the long pulse extensions as we gain more experience with their use. Some of the visualization tools will be enhanced to better handle continuous data streams. The KSTAR and EAST tokamaks have expressed particular interest in these developments. The MDSplus developers at MIT and Padova, Italy will be meeting in Padova this May to discuss the design of additional MDSplus capabilities to meet the data handing requirements of very long pulse Fusion devices.

Work will continue on the MDSplus object-oriented language interfaces. A common suite of classes and methods is planned to support programming languages such as Python, Java, C++, Matlab and perhaps FORTRAN9x. It is anticipated that we will continue to enhance the MDSplus internal code to take advantage of the power of these languages in developing enhancements to the base MDSplus system.

We are investigating schemes to aggregate small I/Os. This has the potential to improve remote I/O performance, especially over high latency networks.

Support for additional data acquisition devices – particularly CPCI will be provided as useful modules are identified. MDSplus capabilities have been extended to permit the development of device specific support using the new python MDSplus objects.

11.3. Partial List of Known MDSplus Sites.

US:

- PSFC - MIT
- PPPL
- GA
- U. Wisconsin
- U. Texas
- UCLA
- Columbia
- U. Washington
- Auburn University
- Los Alamos
- University of Maryland
- University of Utah
- U.C. Irvine
- SAIC, San Diego
- UCSD
- LBL
- NASA, Huntsville
- Ad Astra Rocket, Houston,TX

International:

- IGI- Padua, Italy (RFX)
- EPFL – Lausanne, Switzerland (TCV)
- EFDA-JET – Culham, UK (JET)
- UKAEA – Culham, UK
- IPP-Garching, Germany
- CEA – Cadarache, France (TORE-SUPRA)
- Kurchatov Institute of Nuclear Fusion – Moscow, Russia
- IPP – Hefei, China (HT-7, EAST)
- National Fusion Research Institute, Taejon, S. Korea (KSTAR)
- NIFS – Toki, Japan
- Australia National University, Canberra (HELIAC)
- ENEA - Frascatti, Italy
- University of Quebec
- Institute for Plasma Research, Gandhinagar,India
- Ad Astra Rocket Company, Liberia,Costa Rica

Appendix A

Alcator C-Mod Publications –2009 to present

Papers Published in Refereed Journals:

Binus, A., Wukitch, S.J., Lin., Y., Murray, R., Pfeiffer, A., Gwinn, D., “An intermediate power amplifier upgrade for 40 to 80 Mhz for the Alcator C-Mod ICRF transmitters”, *Proceedings – Symposium on Fusion Engineering*, 2009.

Binus, A., Lin, Y., Wukitch, S.J., Pfeiffer, A., et al., “Engineering considerations in the implementation of high power fast ferrite tuners on Alcator C-Mod”, *Fusion Science and Technology*, **56**, 2009.

Edlund, E.M., Porkolab, M., Kramer, G., et al., “Phase contrast imaging measurements of reversed shear Alfvén Eigenmodes during sawteeth in Alcator C-Mod”, *Physics of Plasmas*, **16**, 2009.

Edlund, E.M., Porkolab, M., Kramer, G.J., et al., “Observation of reversed shear Alfvén Eigenmodes between sawtooth crashes in the Alcator C-Mod tokamak”, *Physical Review Letters*, **102**, 2009.

Ferrara, M., Hutchinson, I.H., Wolfe, S.M., “State reconstruction and noise reduction by Kalman filter in the vertical position control on Alcator C-Mod”, *Fusion Science and Technology*, **56**, 2009.

Greenwald, M., “Tutorial: Verification and validation for magnetic fusion”, *Phys. Plasmas*, **17**, 058101, 2010.

Hillairet, J., Voyer, D., Frincu, B., Meneghini, O, “Modeling of lower hybrid antennas using the ALOHA code and comparisons with Tore Supra experiments”, *Fusion Engineering and Design*, **84**, 2009.

Humphreys, D.A., Casper, T.A. Eidietis, N., Ferrara, M., … Hutchinson, I.E., et al., “Experimental vertical stability studies for ITER performance and design guidance”, *Nuclear Fusion*, **49,** 2009.

Ince-Cushman, A., Rice, J.E., Reinke, M., Greenwald, M., … et al., “Observation of self-generated flows in tokamak plasmas with lower-hybrid-driven current”, *Physical Review Letters*, **102**, Jan. 2009.

Ko, J., Scott, S., Shiraiwa, S., Greenwald, M., Parker, R., Wallace, G., “Intra-shot MSE calibration technique for LHCD experiments”, *Review of Scientific Instruments*, **81**, 2010.

Koert, P., Wukitch, S.J, Beck, W.K., Lin, Y., Doody, J., Mucic, N., “New Alcator C-Mod rotated 10 4-strap ICRF antenna”, *Proceedings – Symposium on Fusion Engineering*, 2009.

Kolmogorov, V.V., “A high-voltage power supply of the diagnostic neutral beam injector of the Alcator C-Mod tokamak”, *Instruments and Experimental Techniques*, **52**, 2009.

Lin, L., Porkolab, M., Edlund, E. M., et al., “Studies of turbulence and transport in Alcator C-Mod H-mode plasmas with phase contrast imaging and comparisons with GYRO”, *Physics of Plasmas*, **16**, 2009.

Lin, L., Porkolab, M., Edlund, E.M., et al., “Studies of turbulence and transport in Alcator C-Mod ohmic plasmas with phase contrast imaging and comparisons with gyrokinetic simulations”, *Plasma Physics and Controlled Fusion*, **51**, 2009.

Lin, Y., Binus, A., and Wukitch, S.J., “Real-time fast ferrite ICRF tuning system on the Alcator C-Mod tokamak”, *Fusion Engineering and Design,* **84**, 2009.

Lin, Y. Rice, J.E., Wukitch, S.J., Greenwald, M.J., et al., “Observation of ion cyclotron range of frequencies mode conversion plasma flow drive on Alcator C-Mod”, *Physics of Plasma*, **16**, 2009.

Lipschultz, B., Whyte, D.G., Irby, J., LaBombard, B., Wright, G.M., “Hydrogenic retention with high-Z plasma facing surfaces in Alcator C-Mod”, *Nuclear Fusion,* **49**, 2009.

Maddison, G.P., Hubbard, A.E., Hughes, J.W., Snipes, J.A., LaBombard, B., et al., “Dimensionless pedestal identity plasmas on Alcator C-Mod and JET”, *Nuclear Fusion*, **49**, 2009.

Marmar, E., Bader, A., Bakhtiari, M., Barnard, H., et al., “Overview of the Alcator C-Mod research program”, *Nuclear Fusion*, **49**, 2009.

McDermott, R.M., Lipschultz, B. et al., “Edge radial electric field structure and its connections to H-mode confinement in Alcator C-Mod plasmas”, *Physics of Plasmas*, **16**, 2009

Meneghini, O., Shiraiwa, S., Parker, R., “Full wave simulation of lower hybrid waves in Maxwellian plasma based on the finite element method”, *Physics of Plasmas*, **16**, 2009.

Mikkelsen, D.R., Dorland, W., “The dimits shift in realistic gyrokinetic plasma turbulence simulations”, *Phys. Rev. Letters* **101**, 2009.

Milanesio, D., Menehini, O., Lancellotti, V., et al., “A multi-cavity approach for enhanced efficiency in TOPICA RF antenna code”, *Nuclear Fusion*, **49**, 2009.

Murray, R.A., Kanojia, A., Burke, W., Terry, D., Binus, A., Wukitch, S.J., Lin, Y., Parkin, B., “Upgrade of the ICRF fault and control systems on Alcator C-Mod”, *Proceedings – Symposium on Fusion Engineering*, 2009.

Rice, J.E., Ince-Cushman, A.C., Bonoli, P.T., Greenwald, M.J., Hughes, J.W., et al., “Observations of counter-current toroidal rotation in Alcator C-Mod LHCD plasmas”, *Nuclear Fusion*, **49**, 2009.

Reinke, M.L., Ince-Cushman, A., Podpaly, Y., Rice, J.E., et al., “Analyzing the Radiation Properties of high-Z impurities in high-temperature plasmas”, *AIP Conference Proceedings*, 2009.

Sattin, F., Agostini, M., Cavazzana, R., Scarin, P., Terry, J.L., “Fluctuations and power spectra in edge plasmas”, *Plasma Physics and Controlled Fusion*, **51**, 2009.

Sattin, F., Agostini, M.., Scarin, P., Vianello, N., Cavazzana, R., Marrelli, L., Serianni, G., Zweben, S.J., Maqueda, R.J., Yagi, Y., Sakakita, H., Koguchi, H., Kiyama, S., Hirano, Y., Terry, J.L., “On the statistics of edge fluctuations: comparative study between various fusion devices”, *Plasma Physics and Controlled Fusion***,** **51**, May 2009.

Simakov, A., Catto, P., “Response to comment on Magnetic topology effects on Alcator C-Mod scrape-off layer flow”, *Plasma Physics and Controlled Fusion*, 51, 2009.

Sips, A.C.C., Casper, T.A., Doyle, E.I., …, Hutchinson, I.H., et al., “Experimental studies of ITER demonstration discharges”, *Nuclear Fusion*, **49**, 2009.

Smick, N. and LaBombard, B., “Wall scanning probe for high-field side plasma measurements on Alcator C-Mod”, *Rev. Sci. Instrum*. **80**, 2009.

Smirnov, R.D., Rosenberg, M., Pigarov, A. Yu., Yu, J.H., Roquemore, A.L., Terry, J.L., et al., “Dust dynamics and radiation in fusion plasmas”, *Proceedings of 2009 IEEE 36th International Conference on Plasma Science (ICOPS),*2009.

Smirnov, R.D., Krasheninnikov, S.IK., Yu, J.H., Pigarov, A.Yu., Rosenberg, M., Terry, J.L., “On visibility of carbon dust particles in fusion plasmas with fast framing cameras”, *Plasma Physics and Controlled Fusion*, **51**, 2009.

Terry, D., Casey, J.A., MacGibbon, P.A., Burke, W.M., et al., “Transmitter protection system upgrade design for lower hybrid current drive system on Alcator C-Mod”, *Fusion Science and Technology*, **56**, 2009.

Terry, J.L., Zweben, S.J., Umansky, M.V., Cziegler, I., Grulke, O., LaBombard, B., Stotler, D.P., “Spatial structure of scrape-off-layer filaments near the midplane and X-point regions of Alcator C-Mod”, *Journal of Nuclear Materials*, **390-391**, 2009.

Wilson, J. R. Parker, R., Bitter, M., et al, “Lower hybrid heating and current drive on the Alcator C-Mod Tokamak”, *Nuclear Fusion*, **49**, 2009.

Wright, J.C., Bonoli, P.T., Schmidt, A.E., Phillips, C.K., et al., “An assessment of full wave effects on the propagation and absorption of lower hybrid waves”, *Physics of Plasmas*, **16**, 2009.

Wukitch, S.J., LaBombard, B., Lin, Y., Lipschultz, B., Marmar, E., Reinke, M.L., Whyte, D.G., “ICRF specific impurity sources and plasma sheaths in Alcator C-Mod”, *Journal of Nuclear Materials*, **390-391**, 2009.

Zweben, S.J., Scott, B.D., Terry, J.L., LaBombard, B., Hughes, J.W., Stotler, D.P., “Comparison of scrape-off layer turbulence in Alcator C-Mod with three dimensional gyrofluid computations”, *Physics of Plasmas*, **16**, 2009.

Submitted for Publication:

Cziegler, I., Terry, J.L, Hughes, J.W., LaBombard, B., “Experimental studies of edge turbulence and confinement in Alcator C-Mod”, submitted to *Physics of Plasmas*, 2009.

Fiore, C.L., Rice, J.E., Podpaly, Y., Bospanyatnov, I.O., Rowan, W.L., Hughes, J.W., Reinke, M., “Rotation and Transport in Alcator C-Mod ITB Plasmas”, submitted to *Nuclear Fusion*, 2009.

Hughes, J.W., Hubbard, A.E, Wallace, G., et al., “Modification of H-Mode pedestal structure with lower hybrid waves on Alcator C-Mod”, submitted to *Nuclear Fusion*, 2009.

Lee, J. P. and Wright, J. C., "A versatile parallel block tri-diagonal solver for spectral codes", submitted to *ACM Transactions on Mathematical Software*, 2010.

Marr, K.D., Lipschultz, B. et al., “Comparison of neoclassical predictions with measured flows and evaluation of a poloidal impurity density asymmetry”, submitted to *Nuclear Fusion*, 2009.

Shiraiwa, S., Meneghini, O., Parker, R., Bonoli, P., Garrett, M., Kaufman, M.C., Wright, J.C., Wukitch, S., “Plasma wave simulation based on a versatile FEM solver”, submitted to *Plasma of Physics*, 2009.

Wallace, G.M., Parker, R.R., Bonoli, P.T., Harvey, R.W., Hubbard, A.E., Hughes, J.W., LaBombard, B., Meneghini, O., Schmidt, A.E., Shiraiwa, S., Smimov, A.P., Whyte, D.G., Wilson, J.R., Wright, J.C., Wukitch, S.J., “Absorption of lower hybrid waves in the scrape off layer of a diverted tokamak”, submitted to *Physics of Plasmas*, 2010.

Wright, John C., Lee, Jungpyo, Bonoli, Paul, Valeo, Ernie, Phillips, Cynthia K., and Harvey, Robert H., “Challenges in self-consistent full wave simulations of lower hybrid waves”, submitted to *Journal of IEEE Transactions on Plasma Science on Numerical Simulation of Plasma*, 2010.

Conferences:

International Sherwood Fusion Theory Conference, Denver, CO, USA May 2009

Talks

Bonoli, P.T., Wright, J.C., Richardson, A.S., Schmidt, A.E., Phillips, C.K., Valeo, E., and the RF Sci Dac Team, “Parametric Studies of Lower Hybrid Wave Propagation in Tokamak Plasmas Using an Electromagnetic Field Solver”.

Richardson, A.S., Bonoli, P, Wright, J., “A Matched Asymptotic Treatment of the Reflection of LH Waves from a Cutoff”.

Wright, J., Bonoli, P., Phillips, C., Valeo, E., Harvey, R., and the RF-SciDAC Team, “Self-Consistent Full Wave Simulations of Lower Hybrid Waves”.

22nd US Transport Task Force Workshop, San Diego, CA, USA, April 2009

Talks

Edlund, E., “Observations of Alfvén Eigenmodes During Sawteeth and the Current Ramp”.

Hughes, J.W., “Effects of Lower Hybrid Waves on H-Mode Transport on Alcator C-Mod”.

Lin, L., “Studies of Turbulence and Transport with Phase Contrast Imaging in Alcator C-Mod Ohmic Plasmas and Comparisons with Gyrokinetic Simulations”.

Posters

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

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

36th European Physical Society Conference on Plasma Physics, Sofia, Bulgaria, June 2009

Poster

Lin, Y., “ICRF Mode Conversion Flow Drive on Alcator C-Mod”.

51st Annual Meeting of the APS Division of Plasma Physics, Atlanta, GA, USA, Nov. 2009

Invited Talks

Cziegler, I.., “Structures and Velocities of the Edge Turbulence in Alcator C-Mod.”
Greenwald, M., “Verification and Validation for Magnetic Fusion: Moving Toward Predictive Capability”.

Hughes, J., “Edge Pedestal and Confinement Regulation on Alcator C-Mod”.

Lin, L., “Comparison of Experimental Measurements and Gyrokinetic Turbulent Electron Transport Models in Alcator C-Mod Plasmas”.

Porkolab, M., “Taming Magnetically Confined Plasmas with RF Waves: A Historical Perspective

Shiraiwa, S., “Plasma waves Simulation Based on a Versatile FEM Solver on Alcator C-Mod”.

Contributed Orals

Bose, B., “Studies of Filament Formation During Lithium Pellet Injection in Alcator C-Mod”.

Dominguez, A., “Initial Results from the Swept Frequency O-Mode Correlation Reflectormeter System in Alcator C-Mod”.

Edlund, E., Measurement of the Adiabatic Index through the Temperature Scaling of Reversed Shear Alfvén Eigenmodes”.

Fiore, C., “Analysis of Rotation and Transport Data in C-Mod ITB Plasmas”.

Hubbard, A., “Overview of Recent Research on the Alcator C-Mod Tokamak”.

Kessel, C., “Simulations of ITER and ITER-like Discharges in Alcator C-Mod”.

Lin, Y., “ICRF Mode Conversion Flow Drive on Alcator C-Mod”.

Lin, L., “Comparison of Experitmental Measurements and Gyrokinetic Turbulent Electron Transport Models in Alcator C-Mod”.

Lipschultz, B., “Fuel Recovered Following Un-Mitigated Disruptions in Alcator C-Mod with High-Z PFCs”.

Marmar, E., “Studies of Enhanced Energy Confinement Discharges with L-Mode-like Edge Particle Transport in Alcator C-Mod”.

Reinke, M., “Implications of C-Mod Disruption Mitigation Studies for ITER”.

Smick, N., “Closing the Poloidal Flow Loop in the High-Field Side Scrape-Off Layer of Alcator C-Mod”.

Terry, J., “Poloidal and Radial Characteristics of Edge Turbulence at the High-Field-State Midplane of Alcator C-Mod”.

Wallace, G., “Observations of Lower Hybrid Wave Interactions in the Scrape-Off-Layer of a Diverted Tokamak”.

Wukitch, S., “Evaluation of ICRF Heated Discharges with Boron Coated Molybdenum Tiles”.

Zweben, S., “Relationship of SOL Turbulence to SOL Width in Alcator C-Mod”.

Posters

Arai, K., “Detection of Lower Hybrid Waves on Alcator C-Mod with Phase Contrast Imaging Using Electro-Optic Modulators”.

Bader, A., “Measurements of Fast Ion Distribution in ICRF Heated Plasmas”.

Baek, S., “Nonthermal ECE as a Diagnostic of LH Driven Fast Electrons on Alcator C-Mod”.

Barnard, H., “Student of Tungsten Migration in the Alcator C-Mod Divertor Using Particle Induced X-Ray Emission Analysis”.

Besparnyatnov, I., CXRS Impurity Density Measurement Techniques in C-Mod”.

Bonoli, P., “Full-Wave Analysis of Lower Hybrid Wave Propagation in the Edge Plasma of a Tokamak”.

Brunner, D., “Plasma Sheath Heat Flux Transmission in the Alcator C-Mod Divertor”.

Garrett, D., “ICRF Antenna Design Studies for Alcator C-Mod”.

Hartwig, Z., “Development of a Novel In-Situ Accelerator-Based Surface Diagnostic for Alcator C-Mod”.

Howard, N., “Impurity Transport Studies Using the Multi-Phase Laser Blow-Off System on Alcator C-Mod”.

Ko, J., “Thermal issues and Relevant Upgrades of the MSE Diagnostic on Alcator C-Mod”.

LaBombard, B., “Initial Results from Divertor Heat-Flux Instrumentation on Alcator C-Mod”.

Lau, C., “SOL Reflectometer”.

Liao, K., “The Emission Spectrum of Fast Neutrals in Alcator C-Mod and its Use as a Fast Ion Diagnostic”.

Lee, J., “Investigation of Velocity Diffusion in the Presence of a Broad-Band Lower Hybrid Wave Spectrum”.

Ma, Y., “Electron Thermal Transport Studies with Improved n_eT_e Profiles Measured by Core Thomson Scattering Diagnostic on Alcator C-Mod”.

Mikkelsen, D., “Multi-Species Particle Transport in GYRO Simulations of Low-Collisionality, Peaked-Density H-Mode Plasmas in C-Mod”.

Meneghini, O., “Integration of Fokker Planck Calculation FEM Simulation of LH Waves”.

Mumgaard, R., “Upgraded Thermal Management of the MSE Diagnostic on Alcator C-Mod”.

Mumgaard, R., “Alcator C-Mod In Situ Inspection System”.

Ochoukov, R., Interpretation and Implementation of an Ion Sensitive Probe (ISP) as a Plasma Potential Diagnostic”.

Parker, R., “Lower Hybrid Wave Induced Rotation on Alcator C-Mod”.

Payne, J., “Design and Analysis of Divertor Calorimetry in Alcator C-Mod”.

Podpaly, Y., “Calculating the Source Terms in the Momentum Diffusion Equation”.

Rowan, W., “Simulation of Turbulent Impurity Transport in Alcator C-Mod”.

Sears, J., “Excitation of Unstable TAEs and Stable n=0 Modes in Alcator C-Mod”.

Shiraiwa,.S., “Plasma wave simulation based on a versatile FEM solver on Alcator C-MOD”.

Tsujii, N., “Experimental Studies of ICRF Mode Conversion with Phase Contrast Imaging and Comparison with Full-Wave Simulations”.

Wilson, R., “Analysis and Scenario Modeling of LHCD on Alcator C-Mod”.

Wright, J., “Full Wave/Fokker-Planck Analysis of Driven Current and Hard X-Ray Emission Profiles During Lower Hybrid Experiments on Alcator C-Mod”.

Xu, P., “The Alcator C-Mod FIR Polarimeter”.

Zhang, J., “Thomson Scattering Diagnostic for the Scrape Off Layer of Alcator C-Mod”.

Fusion Power Associates Annual Meeting, Washington, DC, December 2009

Talks

Hubbard, A.E., “U.S. Burning Plasma Organization: Supporting US Scientific Contributions to ITER”.

Marmar, E., “Alcator C-Mod Research Highlights and Plans”.

Porkolab, M., “Plasma Physics and Fusion Energy”.

18th Topical Conference on Radio Frequency Power in Plasmas, Gent, Belgium, June 2009

Invited Talks

Lin, Y, “ICRF Mode Conversion Flow Drive on Alcator C-Mod and Projection to Other Tokamaks”.

Parker, R., “Modification of Current Profile, Toroidal Rotation and Pedestal by Lower Hybrid Waves in Alcator C-Mod”.

Wright, J., “Self-Consistent Full Wave Simulations of Lower Hybrid Waves”.

Wukitch, S., “Results from Alcator C-Mod ICRF Experiments”.

Posters

Bader, A., “Measurements of Fast Ion Distribution in ICRF Heated Plasmas”.

Bonoli, P., “Full Wave Studies of Lower Hybrid Wave Propagation in Tokamak Plasmas”.

Meneghini, O., “Integrated Numerical Design of an Innovative Lower Hybrid Launcher for Alcator C-Mod”.

Shiraia, S., “Plasma Wave Simulation Based on a Versatile FEM Solver on Alcator C-Mod”.

Schmidt, A., “Measurement of Fast Electron Transport by Lower Hybrid Modulation Experiments in Alcator C-Mod”.

Tsujii, N., “Measurements of Mode Converted Ion Cyclotron Wave with Phase Contrast Imaging in Alcator C-Mod and Comparisons with TORIC”.

Wallace, G., “Observations of Lower Hybrid Wave Absorption in the Scrape Off Layer of a Diverted Tokamak”.

Wukitch, S., “Results from Alcator C-Mod ICRF Experiments”.

16th International Conference on Atomic Processes in Plasmas Conference, Monterey, CA, USA March 2009

Talks

Reinke, M., “Verification of Radiating Properties of High-Z Impurities in High Temperature Plasmas”.

Posters

Bespamyatnov, I., “A New Integrated CXRS/BES system for accurate measurement of the impurity density profiles of the Alcator C-Mod Tokamak”.

Podpaly, Y., “High Resolution of 2x1/2 – 2p3/2 Transitions in W71+ through W65+”.

Rice, J.E., “The Arl7+ Lya2/Lya1 Ratio in Alcator C-Mod Plasmas”.

Rowan, W., “The Emission Spectrum of Fast Ions in Alcator C-Mod and its Application to Measurement of the Fast Ion Distribution”.

19th International Toki Conference, Toki, Japan, Dec. 2009

Poster

Shiraiwa, S., “FEM based simulation of IC and LH antenna-plasma coupling”.

12th International Workshop on H-Mode Physics and Transport Barriers, Princeton, NJ, Sept. 30-Oct. 2, 2009

Talks

Hughes, J.W., “H-Mode Pedestal Regulation with Lower Hybrid Waves on Alcator C-Mod”.

Rice, J.E., “ITB Formation in Alcator C-Mod MCFD Plasmas”.

Posters

Hubbard, A.,E., McDermott, R.M., Dominguez, A., et al., “Improved L-Mode Plasmas with Decoupled Energy and Particle Edge Barriers in Alcator C-Mod”.

Other Invited Talks

Hubbard, A.E., “Physics of Edge Transport Barriers and Importance for Fusion Experiments”, presented at the International Centre for Theoretical Physics, Summer College on Plasma Physics, Trieste, Italy, Aug. 2009.

Porkolab, M., “Experimental Studies of RSAEs on Alcator C-Mod”, presented at the Energetic Particles Workshop, General Atomics, Aug. 2009.

Rice, J.E., “A Spatially Imaging High Resolution X-ray Spectrometer System for the Alcator C-Mod Tokamak”, presented at the 1st International Conference on Frontiers in Diagnostic Technologies, Frascati, Italy, Nov. 2009.

Rice, J.E., “Spontaneous Rotation and Momentum Transport in Alcator C-Mod Plasmas”, presented at the 4th TM on Theory of Plasma Instabilities, Kyoto, May 2009.

Rice, J.E., “Spontaneous Rotation and Momentum Transport in Alcator C-Mod Tokamak Plasmas”, presented at the kickoff meeting of the Center for Momentum Transport and Flow Organization, Univ. California, San Diego, Jan. 2009.

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

Marmar, E., “Status and Plans for C-Mod Research including Joint ITPA Experiments”, IEA/ITPA Joint Planning Workshop, Daejon, Korea, Dec. 2009.

Marmar, E., “Alcator C-Mod Research Highlights and Plans”, presented at Columbia University Plasma Physics Seminar, New York, NY, March 2009.

Marmar, E.S., “Overview of C-Mod Research Highlights and Plans”, NSTX Research Opportunities Forum, PPPL, (presented remotely by video conference), Dec. 2009.

Rice, J.E., “Intrinsic and RF Driven Rotation in Alcator C-Mod Plasmas”, presented at the Physics Colloquium, Columbia University, Feb. 2009.

Shiraiwa, S, “Plasma wave simulation based on a versatile FEM solver on Alcator C-Mod”, presented at the Physics Seminar, GA, San-Diego, Sep. 2009.

Shiraiwa, S., “Recent progress and future plan of Alcator C-MOD lower hybrid experiment”, presented at US-Korea-Japan RF plasma physics workshop, Toki, Japan, March 16-19, 2009.

MIT IAP Talks

Whyte, D.G., “Designing a 24/7 Fusion Device Towards Solving Plasma-materials Issues”, presented at IAP, MIT Pasma Science and Fusion Center, Jan. 2010.

Plasma Science and Fusion Center Research Reports

Edlund, E.M., “A Study of Reversed Shear Alfvén Eigenmode in Alcator C-Mod with Phase Contrast Imaging”, PSFC/RR-09-8, Sept. 2009.

Ko, J-S., “Current Profile Measurements Using MSE On Alcator C-Mod”, PSFC/RR-09-7, June 2009.

Lin, L., “Turbulence and Transport Studies with Phase Contrast Imaging in the Alcator C-Mod Tokamak and Comparisons with Gyrokinetic Simulations”, PSFC/RR-09-9, June 2009.

McDermott, R.M., “Edge Radial Electric Field Studies Via Charge Exchange Recombination Spectroscopy”, PSFC/RR-09-5, May 2009.

Patacchini, L., “Collisionless Ion Collection by Non-Emitting Spherical Bodies in E x B Fields”, PSFC/RR-09-13, Oct. 2009.

Smick, N., “Plasma Flows in the Alcator C-Mod Scrape-Off Layer”, PSFC/RR-09-15, Dec. 2009.

Wallace, G.M., “Behavior of Lower Hybrid Waves in the Scrape Off Layer of a Diverted Tokamak”, PSFC/RR-09-14, Nov. 2009.

Awards and Prizes

Meneghini, O., Shiraiwa, S., “Electromagnetic Wave Simulation in Fusion Plasmas”, Best Poster at COMSOL Conference, Boston, March 2009.

Porkolab, M., 2009 James Clark Maxwell Prize for Plasma Physics.

Appendix B: OFES Facility Budget Planning Meeting Data***

Dollars in Thousands

				FY12 Proposal				
Alcator C-Mod	FY09 Obligated (accomplished)	FY10 Appropriation (current plan)	FY11 Request	No Operations (ready standby)	9 weeks	Base weeks (15)	19 weeks	Full Utilization (25 weeks)
	not including ARRA*							
Facilities Research - Total	**9,002**	**9,035**	**9,744**	**9,077**	**9,637**	**10,255**	**10,930**	**12,125**
Research team (home institution)	6,532	6,621	6,893	6,719	7,073	7,467	7,830	8,605
Research team (collaborators)	2,240	2,237	2,476	2,358	2,564	2,788	2,890	3,245
Diagnostic upgrades (list) ops funding [when applicable]	230	177	375	-	-	-	210	275
FTEs	58.3	56.6	56.8	47.0	51.6	57.4	60.1	64.4
# post-docs included in above	2.4	4.1	5.1	2.0	3.8	4.8	5.4	7.5
Facility Operations - Total	**15,724**	**17,424**	**17,758**	**10,623**	**15,115**	**17,797**	**19,322**	**21,740**
Base operations	10,000	11,124	11,463	8,460	10,450	11,734	12,388	13,176
Maintenance and refurbishment	2,678	2,555	2,615	1,763	2,350	2,615	3,101	2,917
Power and other consumables	1,756	1,790	1,840	350	1,590	1,873	1,968	2,546
Major facility upgrades ops funding [when applicable]	890	1,355	1,540	50	725	1,575	1,865	3,101
Major facility upgrades CE funding [when applicable]	-	400	-	-	-	-	-	-
Diagnostic upgrades CE funding [when applicable]	-	-	-	-	-	-	-	-
Other CE funding [when applicable]	400	200	300	-	-	-	-	-
FTEs	69.3	73.8	77.1	66.0	71.4	77.8	78.9	83.7
# post-docs included in above	0.6	1.1	1.2	0.5	1.2	1.2	1.6	2.8
Run Weeks	9	13+5**	15	-	9	15	19	25
Users	186	190	192	105	174	192	196	212
Facility Total	**24,726**	**26,459**	**27,502**	**19,699**	**24,752**	**28,052**	**30,252**	**33,865**

*ARRA Increments: 935k (operations) + 4960k (upgrades)

**FY10 operations enhanced with FY09 ARRA increment

***Does not include International Collaboration (~80k/year)

Appendix C: Alcator C-Mod Program Detail in Bullet Form

FY10: 18 weeks total research operations (1 week = 4 days, 8 hrs/day)

Areas of Research Emphasis

- Inductive H-mode scenarios (ITER baseline)
  - Scenario demonstrations: ramp-up, ramp-down; half-field scenarios
  - H-mode access: power requirements for high confinement
  - Pre-nuclear phase: comparisons of L-H thresholds in He and D; ELM behavior in He H-mode
  - Impurity seeding in ITER-like discharges (ITPA IOS-1.2)
  - ELM pacing experiments
- Hybrid and steady state operation (ITER-AT)
  - Commission and test advanced Lower Hybrid Launcher
  - Assess LHRF at high density ($>1x10^{20}/m^3$)
- Core transport studies
  - Impurity transport and database
  - Experimental documentation of electron transport/gyrokinetic discrepancies
  - Begin qualifying TGLF for C-Mod regimes
  - Proof-of –principle tests for X-ray Doppler fluctuation measurements (ion temperature and flow velocity)
- Pedestal studies
  - Flux-gradient relationships in edge transport barriers across topologies and regimes
  - Resolve boundaries for edge relaxation mechanisms
  - Expand type-I ELM data sets with full diagnostic coverage; comparisons with modeling
  - Trigger conditions for L-H transitions, including all edge profile measurements
- Plasma Boundary Physics and Technology
  - Power flow in SOL (joint facility target)
  - Blob generation and dynamics with improved diagnostics
  - Impurity effects: toroidal symmetry, detachment, effects on ICRF coupling/antenna performance
  - RF-sheath measurements with new diagnostics
  - Fuel retention, material migration
  - DEMO-like divertor upgrade design
- Wave-Plasma Interactions
  - Ion Cyclotron RF
    - Mode conversion flow drive
    - ICRF impurity effects: identify source location(s); characterize sheaths
    - Sawtooth stabilization/destabilization

    - Lower Hybrid RF
        - Characterize advanced launcher performance: coupling; power handling; reflections
        - Density limit effects: new relfectometer to measure SOL wave fields; explore SOL modifications
        - Pedestal modification, including ELMy regimes
    - ICRF-LH interactions: compatibility, coupling
- Macroscopic Stability
    - Disruptions and mitigation: scaling of quench timing; toroidal asymmetry of radiation; NIIMROD modeling; runaway electron dynamics; ITPA database
    - Alfven modes: ICRF drive; AE mode structure and damping

Plain English Goals
- Testing models of fuel retention in the first-wall
- Runaway electron dynamics during mitigated disruptions
- Accessibility conditions for small ELMs

Awards
- Miklos Porkolab, 2009 James Clark Maxwell Prize for Plasma Physics
- Orso Meneghini, Syun'ichi Shiraiwa, "Electromagnetic Wave Simulation in Fusion Plasmas", Best Poster at COMSOL Conference, Boston, March 2009

FY11D *10% below FY11A Guidance (*10 weeks research operations)

Plans
- Highest priority investigations will be carried out, particularly for ITER/ITPA needs and thesis research, including: Pedestal characterization (joint facility target); LHRF physics (density limits, current drive efficiency); advanced ICRF antenna (sheath reduction, impurity effects with high Z PFC's); disruption mitigation

Implications
- Personnel reductions: 2.5 scientists, 1 post-doc, 3 engineers, 2 technicians; do not replace 2 graduate students
- Reduced research runtime
    - Even fewer of high priority runs can be completed (to less than ¼)
- Delay DEMO-like divertor (~6 months)
- Delay second advanced LH launcher (1 year)
- Defer Hybrid scenario target to FY2012
- Defer data acquisition and computing infrastructure upgrades
    - Hardware becomes obsolete on time scale of about 3 years

FY11A Guidance Budgets: 15 weeks research operations

Prioritized increments:
- Add 5 weeks research operation (to 15 total), and restore personnel cuts

    - Increased productivity across all topical areas
        - Particularly important to take advantage of new tools (LH, ICRF, diagnostics) developed in FY10
- Student training maintained
- Second ICRF antenna on schedule (reduced impurity generation at full power, FY12)
- Second LHRF launcher on schedule (4 MW source total in FY12)
- Advanced DEMO-like divertor scheduled maintained (for FY12 installation)
- Data acquisition upgrade pace maintained
    - Improving reliability and productivity

Areas of Research Emphasis

- Inductive H-mode scenarios (ITER baseline)
    - Extend ITER demo studies to higher and lower ν^*
    - Current rampup scenarios with LHCD
    - Impurity seeding/feedback control of divertor radiation
    - Routine application of disruption mitigation
- Hybrid and steady state operation (ITER-AT)
    - Hybrid scenario feasibility with LHCD
    - Continue LHRF high density assessment
    - LHCD into I-mode plasmas
    - Understand and exploit LHRF pedestal modification
    - Pulse-length extension and integration of advanced scenarios with impurity seeding
- Core transport studies
    - Compare data to simulations of particle and impurity transport
    - Effects of LHCD shear manipulation on turbulence and transport
    - Employ RF flow drive as perturbation for momentum transport
- Pedestal studies
    - Compare pedestal structure with code predictions
    - Role of edge magnetic shear on pedestal widths
    - Explore suppression of pedestal density and possible ELM modification with LHRF
- Plasma Boundary Physics and Technology
    - Critical gradients and edge plasma *phase space*
    - Feedback control of detachment
    - BOUT modeling of C-Mod flow, fluctuation data
    - Finalize DEMO-like divertor design
    - ready RFQ accelerator for installation
- Wave-Plasma Interactions
    - Ion Cyclotron RF
        - Commission and test advanced rotated 4-strap antenna
        - Characterize impact of ICRF power on SOL density profiles
        - Validation of TORIC with mode conversion, utilizing PCI data
        - Evaluate fast wave heating
    - Lower Hybrid RF

        - Continue studies on density limit and rotation: add PCI diagnostic for core wave detection
        - Combine LHCD with ICRF for rotation profile control
        - Complete design of second advanced launcher, begin fabrication
- Macroscopic Stability
    - Disruption mitigation
        - Effects of 2nd gas jet on Prad asymmetry
        - Error field effects
        - Real-time prediction
    - Alfven modes
        - Benchmark AE mode structure & damping with NOVA-K
        - Characterize direct effects on fast particle loss (with new diagnostics for confined and lost fast particles)

Plain English Goals

- ICRF sheath physics with advanced field-aligned antenna
- Characterize H-mode pedestal
- Hybrid scenarios

FY11B Incremental budget: 20 weeks research operation

Prioritized increments:

- Add 5 weeks of research operation
- Earlier implementation of ICRF power supply upgrades
    - Increased reliability, increase of total available source power
- Diagnostic upgrades (core fluctuations)

Research Highlights

- Substantial increased progress across all topical science areas and integrated thrusts, with particular emphasis on high priority ITER R&D and ITPA joint research, both in FY11 and subsequent years

FY12D: 10% Decrement from FY11A (9 weeks research operation)

Plans

- Continue highest priority studies, but at significantly reduced pace

Implications:

- Personnel reductions: 3 scientists, 3 engineers, 1 Post-Doc, 3 technicians; do not replace 2 graduate students
- Reduced runtime (6 weeks)
    - Even fewer of priority runs can be completed (to less than ¼)
- Delay installation of second rotated ICRF antenna (to FY13)
- Delay installation of DEMO-like divertor (to FY13)
- Delay installation of second advanced LH launcher – full power not available for experiments until FY13

- Delay/defer implementation of key diagnostic upgrades (Accelerator-based surface analysis system, fast ion loss diagnostic, Doppler reflectometry)

FY12A Guidance Budgets (2% above FY11A): 15 weeks research operation

Prioritized increments:

- Add 6 weeks research operation and restore personnel cuts
- 2nd advanced LH launcher on-schedule (install FY12)
- DEMO-like high temperature divertor on-schedule (install FY12)
- Complete and install 2nd advanced 4-strap ICRF antenna
- Key diagnostics on-schedule

Areas of Research Emphasis

- Inductive H-mode scenarios (ITER baseline)
  - Simulated burn control experiments
  - Continue impurity seeding and power handling studies with high heat flux symmetric divertor
  - Development and testing of advanced plasma control/fault sensing algorithms
- Hybrid and steady state operation (ITER-AT)
  - Increase non-inductive fraction toward steady-state scenarios
  - Increase combined LH+IC power, aiming for higher beta and bootstrap fraction
  - Investigation of non-inductive I-mode scenarios
  - Upgrade LH models to incorporate high density edge effects, compare with experiments
- Core transport studies
  - Momentum transport and self-generated rotation: compare data to simulations
  - Manipulation of magnetic shear and electron/ion temperature ratios for creation of ITBs
- Pedestal studies
  - Role of the pedestal in spontaneous flow generation
  - Relation of particle and thermal transport to fluctuations
  - ExB shear suppression in ELMy and EDA H-modes, I-modes
- Plasma Boundary Physics and Technology
  - Elimination of boron coatings in high heat-flux tungsten divertor
  - Connection of turbulence measurements to modeling (including BOUT)
  - Local fuel retention and surface changes (RFQ accelerator diagnostics, DEMO-like divertor)
- Wave-Plasma Interactions
  - Ion Cyclotron RF
    - TOPICA validation with loading, antenna impedance and SOL density profile measurements
    - Fast ion absorption of LH waves

        - Antenna voltage and power handling, including application of advanced materials
    - Lower Hybrid RF
        - Installation of second advanced launcher with full system capability (4 MW source, >2.5 MW forward power from launchers) for Advanced Scenario studies
- Macroscopic Stability
    - Disruption mitigation
        - D_2 opacity studies
        - Runaway electron position control and deceleration
        - NIMROD modeling

Plain English Goals

- Effects of plasma density on Lower Hybrid wave accessibility and current drive efficiency
- Low collisionality edge energy transport barriers (I-mode)

FY12B Incremental budget (10% above FY11A): 19 weeks research operation (assuming guidance budget in FY11)

Prioritized increments:

- 4 weeks additional research operation, to 19 weeks
- Upgrade to fast vertical control power systems for improved stability at high elongation
- Upgrade to correction coil power supplies for increased capabilities to study effects of applied non-axisymmetric fields
- additional LH klystrons and ICRF FPA to accommodate increased operations
- CO_2 scattering diagnostic, for core fluctuation studies, including LH waves

Research Highlights (see 5 year proposal, dated March 2008, for details)

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