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SUMMARY REPORT ON THE NATIONAL TOKAMAK PLANNING WORKSHOP MIT SEPTEMBER 17-19, 2007

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GENERAL ATOMICS

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INTRODUCTION

This report summarizes the National Tokamak Planning Workshop hosted by MIT in September 2007. The purpose of the workshop was to bring together representatives from the research teams of the Alcator C-mod, DIII-D, and NSTX facilities to present and discuss coordinated research over the period FY09-13. These three major U.S. facilities for magnetic fusion energy research are preparing five-year (FY09-13) program plans for submission to the DOE Office of Fusion Energy Sciences in FY08. Over 60 U.S. scientists attended the meeting at MIT and two traveled from Europe to participate; others participated by videoconference (start and end times each day were shifted to accommodate participation by scientists across the country). The agenda and list of speakers is attached.

Coordinated Research

The close interaction between the three major toroidal research programs in the U.S. was carefully described in a 2004 report to the Fusion Facilities Coordinating Committee coauthored by Synakowski, Taylor, and Wolfe. We copy very generously from the introduction of that report in the introduction here. Close interactions are a key feature of Fusion Energy Science research supported by the DOE. Each facility provides a unique set of plasma parameters, configurations, heat, current and particle sources and control methods, diagnostics, etc. Taken together, C-Mod, DIII-D and NSTX provide a strong portfolio of complementary capabilities and resources, which are being effectively utilized to address the major issues in fusion science research.

The 2004 report pointed out that cooperation, collaboration, and coordination take place on several levels within the U.S. fusion program, ranging from high-level long-range planning to individual experiments and one-on-one scientific interaction between scientists on specific topics. It pointed out the great value of collaborative and cooperative research, namely:

- Performing experiments in different devices makes it possible to address critical physics questions because the important parameters over which theoretical models apply are often not available in a single device, and it is only through cross machine comparisons that the models can be tested. Model validation is critical for developing predictive capability of future plasma characteristics, dynamics, and fusion performance.

- Performing experiments in different devices using different technical approaches and experimental techniques provides valuable confirmation of the reproducibility of important new results, a foundational principle of scientific research. New experimental results require duplication and validation before they are accepted and extrapolated to next step experiments such as ITER.
- Collaboration yields more efficient operation of the overall US fusion program by allowing individual programs to address issues of common interest using the tools, techniques, and parameters most readily available to it, while in parallel, other aspects of the problem are addressed more appropriately or efficiently elsewhere.
- Collaboration enables more cost effective research, such as when diagnostic techniques and/or actual hardware can be shared among the experimental groups, not only eliminating costly duplication of effort, but also providing a basis for comparison of results obtained under different conditions using similar or identical instrumentation.

Closely coordinated joint experiments involve participation by researchers from all the institutions in planning, execution and analysis of the work, either by direct onsite presence or via remote access capability. Various program planning activities and forums are in place to support such activities. For example, experimental ideas are often identified, discussed, and developed in ongoing scientific forums such as the Transport Task Force, the US Burning Plasma Organization (USBPO), the ITPA, and the annual Research Opportunities Forums organized by each research facility. The National Tokamak Planning Workshop provides another such opportunity. The three programs have extended their reach by soliciting researcher participation from within each others' programs, from other university and national laboratory programs throughout the country, and from around the world. Ultimately, the continuing high level of individual initiative and creativity in identifying, developing, and executing ideas for cooperative research lies at the heart of identifying and completing joint experiments successfully.

Coordinated research between multiple facilities was divided into three categories, in the Wolfe, Synakowski, and Taylor report: Joint Coordinated Research Experiments, Complementary and Cooperative Research Activities, and Single Device Research.

1. Joint Coordinated Research Experiments — These are experiments which have been allocated run time on two or more devices. They represent areas for which precisely prescribed conditions are required to make sensible comparisons between devices. These experiments generally benefit from on-site participation from the other programs involved. These experiments are particularly useful when device, plasma, or diagnostic differences (e.g., plasma collisionality, normalized gyroradius, and plasma geometry/configuration) can be used to test specific physics models.

2. *Complementary and Cooperative Research Activities* — These generally are experiments and campaigns which do not require precise coordination and control of plasma conditions on different devices, but for which comparative studies and frequent comparison of results and sharing of tools has high scientific leverage. Typically, broad scientific issues are studied in this manner. Examples include the challenge of developing high-bootstrap fraction sustained operation, which is needed both for ITER and for the ST configuration. A more specific example includes the development of complementary approaches to ITER steady-state scenarios on DIII-D and C-Mod. Resistive wall mode research on NSTX and DIII-D has both complementary and cooperative components as well as specific, coordinated elements.

3. *Single Device Research Taking Advantage of Particular Strengths* — Unique plasma characteristics, device capabilities, or diagnostic capabilities can provide important research opportunities that are best explored on one device. Results from such studies can yield cost-effective benefits to the broader U.S. fusion program.

Overview of the National Tokamak Planning Workshop

The National Tokamak Planning Workshop was organized around five topical science areas that were both inclusive of the breadth of research activities and useful for developing coordination plans. These five areas are congruent with the six major research topics outlined in the 2005 FESAC Priorities Subpanel Report on *"Scientific Challenges, Opportunities, and Priorities for the U.S. Fusion Energy Sciences Program,"*

(http://www.ofes.fusion.doe.gov/more_html/FESAC/PP_Rpt_Apr05R.pdf).

The five areas covered by our report on the planning workshop are:

1. Integrated Scenario Research (Steady-state, current profile control, high performance)
2. Multi-scale Transport Physics (Turbulence, rotation, confinement)
3. Macroscopic Plasma Physics (MHD physics and feedback control)
4. Research on Waves and Energetic Particles (RF, fast ion transport, Alfven modes)
5. Plasma Boundary Interfaces (Divertor, SOL, Plasma-wall interactions)

Topical science areas 2-5 map directly to corresponding campaigns in the FESAC report, while the first area integrates elements from many of the FESAC campaigns. Each facility mapped their five-year research plans into this common structure for the purposes of the workshop. Within each area, groups of three scientists (one from each facility) were tasked with constructing a draft coordinated research plan, and two scientists from the community at large were invited to coordinate and facilitate the workshop discussions and to provide 30 minute summary talks during the concluding session.

The Workshop opened with three overview presentations highlighting the main elements of the facility research plans. This was then followed by sequential 2-3 hour blocks focused on each of the five topical science areas. The final day of the workshop concluded with summary discussions of the facility plans led by the two facilitators from each topical science area. The agenda and presentations are available on the web at: http://www.psfc.mit.edu/tpw07/. The remainder of this report contains the five reports on plans for coordinated research for each topical science area. The agenda is included in Appendix A and the list of presenters, facilitators, and research coordination teams are contained in Appendix B at the end of this report.

1. COORDINATED INTEGRATED SCENARIO RESEARCH

S. Wolfe, D. Gates, and J. Ferron

1.1 Introduction

Integrated scenario research seeks to:

1. Develop operating scenarios appropriate to fusion-relevant plasmas by integrating elements of plasma regimes at relevant parameters with appropriate control tools.
2. Establish the physics basis to extrapolate these scenarios from present-day experiments to future devices.

More specifically, this work is targeted toward developing methods to produce the type of high-performance, long pulse plasmas anticipated in future machines such as ITER, FDF, NHTX, ST-CTF, and DEMO. These machines have a variety of requirements depending on whether the operating point includes a burning plasma, the pulse length envisioned, and the expectation for the noninductive current fraction. Thus, the integrated scenario research area is expected to study a number of possible operating scenarios.

Integrated scenario research involves combining solutions to the various challenges of producing a tokamak reactor plasma that are being developed as part of more narrowly targeted work in the areas of transport, stability, boundary physics, wave particle physics, materials and plasma control. The primary research issues, then, involve the compatibility of the various solutions to be combined. Some examples are:

- Compatibility of core current and pressure profiles with the various current drive and heating sources.
- Compatibility of core and boundary solutions, an issue which extends beyond the last closed flux surface to the open field lines.
- Interaction with plasma-facing materials, including heat flux and particle control.
- Control of the operating point, discharge startup, approach to the high-performance phase and the shutdown phase.

In the remainder of Sec. 1, we summarize the anticipated opportunities for coordinated research on C-Mod, DIII-D, and NSTX in the area of Integrated Scenario Research.

1.2 Resources for coordinated integrated scenario research

This area of research will take advantage of the resources already specified in the plans for the three facilities. Among others, these include:

– NBI and RF/microwave heating sources
– Non-inductive current drive (NICD)

– Internal profile diagnostics including means for current-profile reconstruction
– MHD feedback control techniques
– Particle control techniques
– Diagnostics for transport, confinement, stability, and boundary-plasma studies

The five-year facility plans include significant upgrades to significantly enhance capability for integrated scenario research.

The coordinated research plan will take advantage of both the similarities and differences in diagnostics, tokamak parameters, and control actuators in order to determine the best approach for development of operating scenarios for future machines and in order to establish the necessary physics basis for extrapolation. Remote participation and collaboration tools would be used as necessary. Development, benchmarking (code to code), and validation (code to experiment) of shared modeling and analysis tools are also goals for integrated scenario research.

1.3 Topical areas with potential for coordinated research

There are many areas where there could be coordinated research on integrated scenarios. Since DIII-D, NSTX and C-MOD are quite different in their designs and hardware capabilities, much of the coordinated research would involve comparisons of different approaches to the same problem. This brings new challenges to planning coordinated research. For example, different approaches to NICD put current at different locations within the plasma, either more centralized or further out toward the edge. This can, in turn, affect transport and stability. So, in some cases only two of the experiments would be able to contribute significantly, whereas in other topical areas, complementary research at all three facilities is optimal. Here then follows a list of 13 areas where there is potential for coordinated research, with comments on the appropriateness for each of the facilities.

A. Steady-state scenarios with 100% non-inductive current drive.

Development of fully non-inductive discharge scenarios is key for the steady-state mission of ITER and for DEMO. The three facilities can target development of such discharges, utilizing a wide range of values for the bootstrap current fraction. C-MOD should be able to produce fully non-inductive discharges with nearly 100% external current drive or with large bootstrap fraction and less external current drive. DIII-D and NSTX would typically seek to achieve fully non-inductive discharges with large bootstrap current fractions and a relatively small amount of external current drive. Research goals for this topic over the next five years could include the following:

i. Demonstrate operation with 100% non-inductive current drive in high-performance scenarios which scale to ITER (i.e., with H-mode confinement at relevant density and plasma shape).

ii. Compare techniques for maximizing the bootstrap current fraction, including the dependence on aspect ratio, pressure profile shape (broad profile versus ITB), q profile, and plasma shape. Experiments comparing NSTX to C-Mod and DIII-D can identify the effects of aspect ratio, while comparisons between C-Mod and NSTX and DIII-D can illuminate the role the pressure profile since C-Mod places more emphasis on high-performance discharges with an ITB whereas DIII-D and NSTX more typically produce discharges with broad pressure profiles (though ITB scenarios can be produce). Also, NSTX and DIII-D operate at higher β_N than C-Mod.

iii. Execute a coordinated experiment in which all three machines run a 100% noninductive discharge with high bootstrap current fraction with discharge parameters as near equal as possible. This would probably be executed at relatively low plasma current and high poloidal beta.

iv. Compare various combinations of current drive and heating schemes for producing 100% noninductive discharges. These comparisons would take advantage of the different techniques and power levels for current drive and heating available for the three tokamaks. (Lower Hybrid CD, Mode Conversion CD, Electron Cyclotron CD, Neutral Beam CD, Fast Wave CD, Electron Bernstein CD).

v. Compare techniques for maximizing the fusion performance. Discharges with 100% noninductive current can be achieved at high values of q_{95}, but these cases do not necessarily have high fusion performance, which is proportional to the product of beta and energy confinement time. The three facilities will have different limitations on the ability to combine high noninductive current fraction with high values of $\beta\tau_E$.

vi. Use the inter-machine comparisons to establish the scaling to future devices and the optimum heating and current drive requirements.

B. Demonstrate current profile control

Current profile control is needed to maintain MHD stability at high β. The current profile also affects transport and confinement, hence overall fusion performance. Each discharge scenario with a significant fraction of noninductive current drive has unique issues for current profile control. Coordinated experiments using the current drive, heating and q measurement tools available at each facility will improve the physics basis needed for future devices and speed development of effective control methodologies. Coordinated research is proposed for three areas:

i. Develop model based control algorithms, including work by the broader research community. Control model development includes testing/validation on experiment.

ii. Compare effectiveness of various actuators (LHCD, NBCD, ECCD, and FWCD), including the study of combinations and synergies among the techniques and the effects of the differing current drive localization.

iii. Compare control capability in the strong external actuator versus weak actuator (large bootstrap fraction) regimes (LHCD at C-Mod gives strong current drive capability, thus strong actuator. ECCD is a relatively weak actuator at DIII-D where high bootstrap current fraction is anticipated. At NSTX, neutral beams are the primary external current drive source providing up to 25% of the current, and are also the primary heat source. So the capability to externally modify the current profile may be limited.)

C. Hybrid scenario sustainment and control

This work includes the lower q_{95} case sometimes called "advanced inductive." Key topics are the role of RF (C-MOD) and neutral beam current drive (NSTX and DIII-D) and MHD (DIII-D, NSTX C-MOD) in regulating central q. The role that neoclassical tearing modes play in flattening the current profile remains poorly understood and it is expected that comparative experiments will provide important new information.

D. Demonstration of the ITER ELMing H-mode baseline scenario

This scenario has sawteeth and $q(0) < 1$, a regime where NSTX does not operate. Demonstration of the ITER baseline scenario provides a platform for other ITER-relevant physics studies and a point of comparison for the hybrid and steady-state scenario discharges. Experiments can be conducted on DIII-D and C-Mod.

E. Core-Edge Integration Experiments

Comparative experiments demonstrating methods for integration of edge solutions with high performance core plasmas is needed to prepare for DEMO. Potential edge solutions include the radiative divertor, flux expansion, liquid metal targets (in NSTX), and rotating 3D edge perturbations. Solutions also must be compatible with ELM control techniques (e.g., RMP, for which capability will be available in DIII-D, NSTX, C-Mod). The effect of different wall materials (solid metal in C-MOD, and carbon in NSTX and DIII-D) and divertor pumping must also be examined.

F. Feedback stabilization of MHD modes

For operation at high β_N, active feedback stabilization will be required to control instabilities. RWM feedback stabilization is relevant to high bootstrap current fraction discharges in DIII-D and NSTX where β_N is expected to be above the no-wall stability limit. NTM stabilization using current drive sources is relevant in many scenarios, particularly the ITER ELMy H-mode baseline scenario as studied in DIII-D and C-Mod.

G. Burn simulations

In all three facilities it appears possible to simulate feedback control of burning plasmas by developing feedback models that compute the expected alpha heating rate and adjust auxiliary heating to match it. RF/microwave heating offers the greatest flexibility for such experiments due to the ability to adjust the deposition profile.

H. Comparisons of integrated scenario operation with reactor-relevant parameters

Discharges with $T_e = T_i$ are feasible in C-Mod, DIII-D, and NSTX, and discharges with low rotation have been produced in C-Mod and DIII-D. The methods used to produce these discharge conditions will vary between the facilities. There may be cases of overlap, such as primarily RF or EC heated discharges.

I. Discharge Evolution studies

Comparisons of the evolution to steady-state as the ratio of pulse length to resistive time is increased are planned. All three facilities can run high performance discharges for times equal to or greater than one resistive profile relaxation time.

J. Plasma control development

Research on plasma control benefits greatly from coordinated research because of the common physics issues across all the US facilities. Each device brings to bear different control actuators and controller feedback algorithms. Feedback control systems have grown significantly more sophisticated over the past five years and similar changes are needed and expected over the next five years as understanding of plasma behavior improves. Possibilities exist for coordinated research on:

i. Profile control.
ii. Simulation and modeling.
iii. Control algorithm development .
iv. Control system hardware and software (NSTX, DIII-D. C-Mod).
v. Implementation of elements of the ITER CODAC.
vi. Testing of ITER control approaches.
vii. Operation of tokamaks to mimic ITER dynamic control characteristics.
viii. Real-time stability boundary estimation (NSTX, DIII-D, C-Mod).

K Implementation of machine protection algorithms

Potential areas of work are instability proximity detection, safe shutdown from off-normal events and disruption mitigation. As part of the integrated scenario work, these techniques would be incorporated into routine operation, offering robust protection while minimizing false alarms, etc.

L Access to $\beta_N >> \beta_N$ limit without a conducting wall

Steady-state scenarios with significant bootstrap current fraction favor high β_N operation, even above the no-wall stability limit; however, β_N can be at a moderate value in discharges with the bootstrap current fraction near 50%. In order to minimize the externally driven current, such as in DEMO, the bootstrap current fraction should approach 100%, requiring values of β_N beyond what is presently obtained for long pulses in both NSTX and DIII-D. Target values would be, for instance, $\beta_N = 5$ in DIII-D or $\beta_N = 8$ in NSTX. In both experiments, scenarios are envisioned with a broad current profile in order to maximize wall stabilization. This is a possible area for coordinated joint experiments. Comparisons of alternative scenarios, such as high ℓ_i, with the broad current profile case would be possible. This is also an area where the effects of aspect ratio could be studied.

M. Modeling of integrated scenario discharges

There is presently a large set of modeling codes in use for developing integrated scenario discharges for present and future experiments. These codes should be benchmarked using experimental data from the three experiments to validate the models. These tools can then be used to design integrated scenario discharges for both the present set of experiments and for future machines.

2. COORDINATED MULTI-SCALE TRANSPORT PHYSICS RESEARCH

E. Doyle, D. Mikkelsen, and S. Scott

2.1 Introduction

Multi-scale transport physics research is a rich and varied field, encompassing experimental confinement and turbulence studies, transport modeling and simulation, as well as fundamental theory. In the following we focus on possibilities for experimental collaborations between the three main US facilities under the headings of model validation, ion and electron thermal transport, momentum transport and plasma rotation, impurity and ion particle transport, density peaking and particle pinches, as well as operation under reactor relevant condition. In addition, there are also sections containing comments on diagnostic implications as well as possible methods to improve collaboration mechanisms. The relative priority of the issues listed has not been established, though it is expected that model validation as well as improved understanding of electron and momentum transport will form important focus areas for all three devices.

2.2 Validating models of turbulent transport

Transport model validation has been repeatedly identified as the most important and critical issue facing the transport community for the immediate future. Validated transport models are essential for ITER and beyond. Appropriate theory-based models exist and are available for testing, but none have yet been validated. Due to time scale constraints and ITER diagnostic capability limitations, this validation effort can only be performed on present devices, where the US has world-leading capabilities.

Validation of theoretical models of plasma transport inherently involves coordination across tokamaks; general validity of a model cannot be demonstrated by tests involving a single tokamak. In addition, validation is truly convincing when transport models are tested on multiple scales. First, are macroscopic tests based on profile measurements. Transport models are typically “stiff,” with large changes in transport at critical gradients as turbulence modes are destabilized. Consequently, a key experimental test of these models is to verify that ITG, TEM and ETG modes are stabilized/destabilized at the predicted gradient levels. Experiments to explore turbulence mode boundaries and scaling have already been performed and should be expanded. Second, “microscopic” turbulence measurements can be compared to gyrokinetic code predictions, e.g. fluctuation amplitudes, spectrum, correlation lengths, etc. These should be measured for as many fluctuating fields as possible, not just density fluctuations, since simultaneous measurements of multiple fields provide a more stringent constraint on the codes. Other critical tests of fundamental theory relate to the turbulence self-regulation provided by zonal flows and GAMs, as well as 'external' turbulence suppression mechanisms such as $E \times B$ shear and magnetic shear. These critical model validation tests can be performed to varying levels on all three devices and are a clear focus area for the US program over the next five years.

In addition to these technical validation issues, a separate urgent need in this area on all facilities is for an increased number of "analysts" – technical specialists who can close the loop between code predictions and experimental measurements.

2.3 Ion and Electron Thermal Transport

The challenge of improving transport understanding in this area includes dealing with a wide variety of issues, from seeking further detailed confirmation of the present picture of drift wave dominated ion transport, to the urgent need to determine the fundamental source of anomalous electron transport (what type, scale?). Validation and benchmarking of transport models and theory based understanding can be substantially aided by coordination between the three main US facilities. Examples of possible areas of coordination are as follows:

What turbulence scale(s) govern(s) electron transport? For electron transport studies, both DIII-D and NSTX have recently invested substantially in new high-k (small scale) turbulence diagnostics, while CMOD has long had a PCI system. With these systems, turbulence can now be monitored over a broad wavenumber range, covering long to short wavelengths. With these new diagnostics it is now possible to address the existence and scaling of turbulence in ITG, TEM and ETG wavelength ranges with electron transport. Here coordination could take two forms: First, identification of common experiments, such as variation of plasma gradients so as to stabilize/destabilize ETG or TEM modes and correlate with changes in electron transport. In addition, together the machines can clearly test a broader range of conditions than individually, e.g. CMOD routinely operates with equilibrated electron and ion temperatures, DIII-D can test electron transport in multiple improved core confinement regimes, while NSTX has electron dominated transport, where long wavelength turbulence is often calculated to be stabilized. Note that recent diagnostic improvements on CMOD, specifically the addition of a 30+ chord x-ray crystal spectrometer will allow separate ion and electron transport rates to be ascertained for the first time.

Beta scaling of transport. The beta scaling of transport is critical to the US vision of creating an attractive AT based tokamak or ST. Here, what had been an emerging experimental consensus (that the beta scaling is weak or non-existent), has been thrown into confusion by more recent experiments showing such a dependence, in line with previous global database studies. This question requires input from multiple devices, and the US machines provide an excellent platform for such studies, with a wide beta range available. Obvious collaboration possibilities in this area include similar beta scans, shaping scans, reducing variations in ELM activity as beta varies, and increased emphasis on determining local turbulence changes as beta is varied, etc.

Fundamental tests of turbulence characteristics. The model validation possibilities via comparisons to measured turbulence characteristics discussed in the previous section are directly applicable to ion and electron transport studies and are possible on all three devices. Other specific possibilities for coordination in this area include the tests of electron-scale transport discussed above, as well as zonal flow/GAM studies. It has been established theoretically that turbulence generates zonal flows that, in turn, limit the turbulence amplitude through $E \times B$ shearing and/or energy exchange. Here, the term zonal flows includes both the zero-mean-frequency zonal flows as well as the coherent higher-frequency Geodesic Acoustic Mode (GAM), which has been clearly identified experimentally in several ways (BES, HIBP, probes, Doppler Reflectometry). Greater experimental characterization of zonal flow behavior is required to understand this complex process and validate nonlinear simulations. Zonal flow dynamics are critically dependent on several parameters, including collisionality, safety factor, and spatial location. Zonal flows have been measured on DIII-D using BES and reflectometry, and can in principle be measured with Doppler Reflectometry on any machine. The feasibility of BES and HIBP on NSTX will be investigated and implementation would greatly facilitate these studies. A coordinated series of experiment to identify zonal flows on all machines, and vary collisionality and safety factor in a systematic fashion would enhance our understanding of the driving, damping, and turbulence-zonal flow interaction processes.

2.4 Momentum Transport and Plasma Rotation

Studies of momentum transport and intrinsic rotation are planned by each group, with some coordination of intrinsic rotation experiments already well established (C-Mod/DIII-D). Intrinsic rotation can be thought of as the result of momentum transport in the absence of external torques, and there is much evidence for similar non-diffusive components of momentum transport in the presence of torques applied by neutral beam injection. C-Mod is expected to contribute very heavily to characterization of intrinsic rotation and testing of models because rotation measurements are available in virtually every discharge and there are never any significant externally applied torques. Extensive intrinsic rotation studies in TCV exhibit very complex dependences on density/collisionality, q, L- vs H-mode; corresponding results from C-Mod would be very helpful in developing an understanding of both intrinsic rotation and in interpreting more complex momentum transport experiments that include external torques in DIII-D and NSTX. The C-Mod intrinsic rotation studies will continue to be coordinated with DIII-D, which has already provided ‘dimensionlessly similar’ discharges that serve to establish the identity of relevant dimensionless parameters. This activity may be extendable to NSTX (although it will probably not be feasible to match the aspect ratio*) if* a suitably non-perturbative diagnostic can be used. The existing CHERS system on NSTX can probably be re-configured in software (join spatial channels to make up for lost light, do joint line fits simultaneously) to make effective use of shorter NBI ‘blips’, but this has not yet been tested. In addition, late in the 5-year plan a curved crystal X-ray spectrometer (of the type used this year on C-Mod for momentum transport studies) is to be installed on NSTX; this would enable more ubiquitous rotation measurements. NBI-based rotation diagnostics in DIII-D are already less perturbing than in NSTX, and further reductions will result from planned upgrades. RF-heated ‘ITER-similarity’ experiments varying ρ^* within each device and across the set of TCV, C-Mod, DIII-D could provide the basis for an extrapolation to ITER. The very wide range of conditions available (including NSTX as well) will enable a comprehensive test of theoretical models of intrinsic rotation.

Momentum transport coefficients are inferred from the plasma response to external torques supplied by NBI in DIII-D and NSTX. These torques can be modulated, and the direction can even alternate between co- and ctr- during a shot in DIII-D. Magnetic braking of rotation is a proven tool that can be applied on all three devices. The analysis of the momentum transport experiments should include the viscosity caused by non-axisymmetric error fields (or the error fields should be effectively nulled out with correction coils). Sophisticated torque calculations are in hand, but there are few tractable theoretical models for momentum transport available for testing. Momentum transport is included in gyrokinetic turbulence simulations, and a broad array of fluctuation diagnostics are installed and planned for each tokamak. Measurements of rotation fluctuations have been proposed and would be valuable, but such measurements are not presently available. Much of the hardware and software needed for a comprehensive study of intrinsic rotation and momentum transport is available (distributed across the three tokamaks), but this subject receives less attention than heat transport and the greatest need is more researchers. The very wide range of conditions available will enable a comprehensive test of theoretical models of momentum transport.

2.5 Density peaking and electron particle pinch

The ubiquitous inward electron particle pinch generates moderately peaked electron density profiles even in H-mode plasmas with no core particle fueling from beams. This behavior affects ITER performance projections because for a given τ_E, peaking the density profile increases the generated fusion power and affects the bootstrap current profile. There is an ITPA activity dedicated to improved understanding of this phenomena to which all three U.S. facilities contribute (CDB-9, "density profiles at low collisionality"). The emphasis on collisionality scaling in CDB-9 arises mostly from observed correlations of the density profile peakedness with collisionality in H-Mode plasmas on several tokamaks, which indicate that density peakedness increases as the collisionality decreases – a favorable trend for ITER.

Studying electron particle transport including the inward pinch provides an additional – and currently underutilized -- opportunity to benchmark gyrokinetic models of turbulence. Generally, the neoclassical Ware pinch is small compared to turbulence-driven pinches (an exception being V_p observed inside internal transport barriers in C-Mod, which is consistent with the Ware pinch alone) and if desired it can be eliminated entirely thru current drive that yields $V_{loop} = 0$.

Both analytic theory and various numerical simulations of ITG/TEM turbulence predict inward particle fluxes. These are qualitatively grouped under two categories, a 'thermo diffusion' pinch driven by $\tilde{\nabla} T/T$ and a 'turbulent equipartition' pinch driven by $\tilde{\nabla} q$. The thermo diffusion pinch is computed to weaken with collisionality (Angioni (2005) and Estrada-Mila(2005)) but the collisionality range with density peaking is predicted to be about 30x smaller than what is seen in experiments (Greenwald, TTF 2007). Benchmarking the turbulence simulations against experimental observations is still in its infancy. Even basic questions such as whether the pinch is dominantly controlled by $\tilde{\nabla} T/T$ or by $\tilde{\nabla} q$ in various regimes has not been rigorously answered.

Going forward, experimental studies of density peaking and the particle pinch can be roughly divided into three categories:

1. Purely empirical (parametric) scaling studies of the density peaking factor in various regimes (L- and H-mode). Here, the emphasis will be on extending the collisionality range on each facility as well as examining possible dependencies on aspect ratio and beta (NSTX). Information may also be derived from database mining relating to the relative importance of $\tilde{\nabla} T/T$ versus $\tilde{\nabla} q$ on the pinch velocity. However, it should be noted that both NSTX and DIII-D have reported that simple global correlation studies of density peakedness with collisionality are of limited utility.
2. Controlled experiments to carefully separate effects of $\tilde{\nabla} T/T$ versus $\tilde{\nabla} q$ on the pinch velocity. DIII-D (Baker, 2005) used current ramps and ECH heating to vary $\tilde{\nabla} q$ at constant L_{Te} in L-mode plasmas (as well as varying L_{Te} at constant $\tilde{\nabla} q$) to demonstrate that $\tilde{\nabla} q$ affects the pinch velocity much more strongly than L_{Te}. This result is in qualitative agreement with NSTX/MAST comparisons which find a strong correlation between density peakedness and $j*$ (which is q-related) in both L- and H-mode plasma. The DIII-D experiment could be duplicated on

C-Mod using LHCD to modify the current profile and ICRF to control L_{Te} and perhaps on NSTX as well, using current-ramps to vary $\tilde{\nabla} q$.

3. Detailed comparisons of electron particle transport with comprehensive gyrokinetic turbulence simulations. It is well known that comparing several computed transport channels *simultaneously* against experimental data is highly valuable for challenging the turbulence simulations. Ideally, these experiments will simultaneously measure power and momentum flows, the steady-state electron density profile as well as perturbative measurements of D_e, V_p (with e.g. puffing or beam-blip) D_{imp}, V_{imp}, and fluctuation spectra.

Initially, these simulation/experiment comparisons can be simply performed in ‘vanilla’ L- and H-Mode plasmas in each facility, with little regard to matching relevant plasma parameters such as ν^*. These should be supplemented by experiments in individual facilities in their respective enhanced confinement regimes to evaluate the behavior of the pinch velocity over a wider range of turbulence drive mechanisms (ITG, TEM, ETG, neoclassical). Subsequent multi-machine controlled experiments to assess the influence of a particular parameter will be guided by the results of the initial experiments.

TABLE 2-1. Facility capabilities relating to electron particle transport studies.

	DIII-D	C-Mod	NSTX
q(r) control	Ip ramp	LHCD	Ip ramp
LTe control	ECH	ICRH	HHFW, EBW?
regimes	L, H, VH, NCS, hybrid	L, H, ITB	L, H, RS
ν*	Low-moderate	Moderate-high	Low-moderate?
special	Controllable beam torque. Good control system.	High B_T. No core particle source or external torque. Ti=Te.	high-β. Spans e-s to e-m dominated. Electron dominated heat transport. Large ExB shear.

Required diagnostics and hardware capability: All three facilities already have adequate diagnostics to carry out the experiments proposed here, including multi-point, calibrated Thomson scattering and a variety of fluctuation diagnostics. We will need a follow-up analysis to quantify the capability on C-Mod and NSTX to modify the L_{Te} profile with various heating sources (ICRF, LH, HHFW). One additional issue that needs to be discussed is the role of impurities in fueling the plasma edge that may confuse particle transport studies. For example, the carbon impurity in NSTX sometimes remain in the vicinity of the plasma edge (low transport coefficient) thereby generating density ‘ears’.

2.6 Impurity and ion particle transport

Impurity and ion transport affects performance projections for ITER and DEMO primarily through ash accumulation and radiative losses. There are no high-priority ITPA tasks directly targeted at impurity transport, presumably reflecting a consensus that possible performance deterioration associated with impurities is less serious than other issues.

A robust experimental observation on DIII-D is that in regimes of anomalously high transport, the measured particle diffusivity is close to the measured thermal diffusivity. The net radial transport can be small because it is counteracted by an inward pinch, nevertheless there is little tendency for central impurity peaking. By contrast, in the presence of enhanced energy confinement the impurity diffusivity drops to neoclassical values while the pinch term (including the neoclassical Ware pinch) remains, thereby yielding a net inward flow of impurities and a centrally-peaked impurity density profile. Perturbative gas puff experiments on NSTX have also observed low impurity diffusivities in regimes with large $E \times B$ shear.

Based on an empirical model of impurity transport that matches observed behavior in multiple regimes (L-, H, VH, and NCS with an ITB) on DIII-D, Wade et al. (JNM 290-3 (2001) 773-777) argue that impurity accumulation could become an issue for burning plasmas that operate in enhanced-confinement regimes.

There is robust experimental evidence from multiple tokamaks that ‘dilution’ of the plasma by impurities can significantly improve global energy confinement. This behavior is qualitatively consistent with gyrokinetic simulations (flux-tube limit, $\nu^* = 0$) which show that dilution by helium can significantly improve particle confinement, and modestly improve energy confinement. The GYRO gyrokinetic simulation code can include impurities and can compute the expected impurity particle flows as a function of both impurity density and its gradient, which presents yet another opportunity to benchmark the simulation. All three facilities have impressive CER capabilities to measure the radial profile of low-Z impurities (Table 2).

The favorable confinement effect of dilution due to low-Z impurities also represents a hazard when comparing measured power flows in plasmas against gyrokinetic simulations. The dilution is not negligible in many plasmas of experimental interest because the plasma-facing components are themselves low-Z (carbon in NSTX and DIII-D) or are coated with a low-Z (boron in C-Mod, lithium in NSTX), so the density of low-Z impurities is important as part of the standard suite of profile diagnostics used to characterize plasmas when comparing their performance to gyrokinetic simulations (these are measured on DIII-D and NSTX).

TABLE 2-2. Low-Z impurity density profile diagnostics on C-Mod, NSTX, and DIII-D.

		Impurity	Nchan	Dt (msec)
C-Mod	Edge-pol	B	20	5-6
	Edge-tor	B	20	5-6
	Core-ver	B	19	
	Core-tor	B	10	
NSTX		C (Li,He,Ne)	51	10
DIII-D	ver	C, He, Ne	24	>0.2
	hor	C, He, Ne	31	>0.2

2.7 Transport under reactor relevant conditions and interdependence of transport channels

In gyrokinetic simulations the transport depends sensitively on T_e/T_i, and most experiments are far from the conditions expected in ITER and beyond. It is critical that we understand the impact on transport and confinement of transitioning present operating regimes to operation with $T_e/T_i \sim 1$, dominant electron heating, low momentum input and high density operation. This list also brings forward the further point that in fusion plasmas the transport in different channels (electron, ion, particle and momentum) cannot be considered in isolation, as they are all strongly coupled and influence each other. Recent and proposed facility upgrades enable a renewed focus on this issue, e.g. new co/counter-NBI capability and additional ECH heating on DIII-D, planned EBW system on NSTX, while CMOD routinely operates with such reactor relevant conditions. Consequently, coordinated experiments can be envisioned to study the coupled impact on transport from independently varying the T_e/T_i ratio, momentum input and operating density. In addition, the collective capability of the three machines is a major advantage in validating gyrokinetic calculations in this area, and should be utilized.

2.8 Diagnostic Implications

As has already been mentioned in passing, the ability to perform coordinated transport experiments is critically dependent on having the capability to obtain complete sets of detailed profile and turbulence measurements on each device. Examples of key profile measurements are q and T_i profiles. Comparison of turbulence measurements would be aided by having similar capabilities on all or multiple machines, e.g. BES may be possible on NSTX in addition to DIII-D, and a soft X-ray system similar to BES may be possible on C-Mod, while Doppler/correlation reflectometer measurements are certainly possible on all three devices.

2.9 Improved collaboration mechanisms

In addition to topics suitable for multi-machine collaboration, there is also the issue of how to improve and enable the collaboration process itself. Individual cross-participation and time are key to these collaborations, such that increased collaboration has obvious programmatic implications in terms of resources. In this context, existing US structures in the transport area such as the TTF and BPO seem underutilized. One possible innovation would be to create/expand standing "three-party" working groups chartered by the TTF and/or BPO in areas such as:

Validation of codes versus turbulence and transport measurements
Electron transport studies
Intrinsic rotation and momentum transport
Density peaking and impurity transport, etc.

Each such group should have at least one member from each tokamak, and members should participate in the relevant experimental planning, mini-proposal generation, etc., on all three devices.

3. MACROSCOPIC TRANSPORT

A. Garofalo, S. Gerhardt, and I. Hutchinson

3.1 Introduction

This report aims at establishing a set of collaborative ideas in the area of macroscopic MHD, as part of the three-tokamak planning meeting help at MIT in September 2007. Topics related to fast-ion MHD are included in the Waves and Energetic Particles report, and the stability of the edge pedestal is discussed in the Boundary Physics report. It is hoped that these proposals for ideas will stimulate discussion and future collaborations. Neither the ideas listed nor the references are intended to be exhaustive, and the ordering of the topics and subtopics is not intended to indicate priority.

In the following pages, we lay out the importance of studying this topic and present plans for collaborative research on the three facilities in the accompanying tables.

3.2. RWM/RFA physics/control

The Resistive Wall Mode (RWM) is an MHD kink-ballooning instability presenting a major challenge to any tokamak operating at β_N near the no-wall limit, including ITER in its advanced steady-state scenario.[1] Control techniques include rotational stabilization with a nearby wall (1.1 below) and active feedback using magnetic coils (1.2 below).

Experiments on DIII-D and JT60-U have demonstrated that the RWM can remain stable in high B_N scenarios even with the low rotation resulting from nearly balanced neutral beam injection. Progress in theoretical modeling of RWM stabilization at slow plasma rotation has been considerable, leading toward a more detailed kinetic modeling including effect of resonances with diamagnetic and precession drifts. However, quantitative agreement with experimental observation is still elusive. Therefore, it is not possible to predict with confidence the stability characteristics of the RWM in ITER.

A stable RWM in ITER would manifest itself by amplifying the resonant error field, a phenomenon known as resonant field amplification (RFA)[2] which has a strong impact on the threshold of tolerable error fields (1.3 below). The amplification by the stable RWM of resonant error fields has been measured to increase by a up to a factor of ten with β increasing from below to above the no-wall limit. For ITER, this β- dependence of the RFA means a reduction of at least a factor of five in the tolerable error field should be conservatively expected for operation at β exceeding the no-wall limit by as little as

[1] T.C. Hender et al, Nuclear Fusion **47**, S128 (2007)
[2] A.M. Garofalo, et al, Phys. Plasmas **9**, 1997 (2002)

20%. A reduction of one order of magnitude in the tolerable error field may be required for higher β_N ITER explorations.

Note that if the plasma rotation in ITER is not sufficient to provide passive RWM stabilization, direct feedback control of unstable RWMs will be needed to operate at β_N above the no-wall limit.

TABLE 3-1. Suggested Collaboration Activities related to resistive wall mode and resonant field amplification.

Topic	Importance	Collaboration Scheme	Additional Comments
3.2.1 Passive Stabilization of the RWM	Improved understanding of passive stabilization physics would allow better stability predictions for future devices.	• Apply similar computational tools (MARS) to test dissipation models across devices. • Comparison of rotation thresholds in DIII-D and NSTX using injection balance and non-resonant magnetic braking. • Study dependence of threshold on residual error fields.	• Future burning plasmas will not have net large torque input, as in many present tokamaks. • The similar sound speeds, but different Alfven speeds, allow important dissipation physics to be tested. • Rotation profile control possible using co/counter injection balance (DIII-D) and magnetic breaking (DIII-D, NSTX). • Collaboration already in existence.[3]
3.2.2 Feedback Physics And Algorithm Development	Important to develop optimal algorithms for RWM stabilization, including effects such as mode rigidity and the effects nearby passive conductors.	Test combinations of the feedback sensors and stabilizing coils. Study mode rigidity and its impact on feedback efficiency.	Existing midplane coils and proposed off-midplane coils in NSTX, and combination of midplane (C-coils) and off-midplane coils (I-coils) in DIII-D allow for extended feedback-control collaboration.
3.2.3 Resonant Field Amplification	Above the no-wall limit, the rotationally stabilized RWM can couple to and amplify an error field. Growing mode can damp plasma rotation, ultimately leading to the growth of an RWM.	• Apply active MHD spectroscopy[4] in NSTX and DIII-D, in order to study the properties of the stable RWM. • Study RFA dependence on β_N, the rotation profile, and the q-profile, • Study n>1 RFA, and n=1 poloidal harmonic variation on RFA growth. • Determine how error field threshold changes with β_N.	Beyond RFA physics, active MHD spectroscopy is a potential means to test for proximity to the no-wall/with-wall limits in a real-time plasma control scheme.

[3] for example XP-739 on NSTX in CY07
[4] H. Remeirdes, et al, Phys. Rev. Lett. **98**, 055001 (2007)

3.3. Neoclassical Tearing Modes

The Neoclassical Tearing Mode (NTM) is an important beta-limiting instability in any hot tokomak plasma, and is likely to be the main beta limiting instability in ITER ELMy H-Mode. These types of plasmas have significant non-inductive current driven by the radial pressure gradient, known as "bootstrap" current. When a magnetic island is formed, possibly through coupling to a sawtooth crash or classical tearing mode triggering, the pressure flattening across the island causes a hole to develop in the bootstrap current. This hole in the bootstrap current then causes the island to grow, leading the further loss of bootstrap current and island growth. In their saturated state, these neoclassical islands can cause a significant degradation in confinement for the $m/n = 3/2$ NTM, and severe confinement degradation and possible disruption for the $m/n = 2/1$.[5]

The physics understanding of these modes is however, incomplete. The mode evolution is strongly affected by the physics of small magnetic islands, where further research remains (3.3.1 and 3.3.3 below). The onset and saturated state is also affected by the equilibrium velocity shear (3.3.2 below) and possibly the presence of error fields (3.3.5 below), although these physics are not fully understood. Techniques are being developed to avoid or mitigate the effects of this instability. For instance, the initial seed island that spawns the instability can be controlled by manipulating the sawtooth period.[6] Once the instability is formed, current drive techniques using LHCD and ECCD have been developed to replace the missing O-point current (3.3.4 below).

[5] T.C. Hender et al, Nuclear Fusion **47**, S128 (2007)
[6] O. Sauter, et al., PRL **88**, 105001 (2002)

TABLE 3-2. Suggested NTM Collaboration Activities.

Topic	Importance	Collaboration Scheme	Additional Comments
3.3.1 Threshold β_N Scalings[7]	Needed for predicting NTM onset thresholds for future devices.	For similar plasma shapes, conduct controlled β_N ramps in order to establish, if possible, the onset threshold as a function of dimensionless variables.	Results well established in DIII-D. Extend studies to low-A (NSTX) and high field (C-Mod).
3.3.2 Rotation effects on NTM stability	Future burning plasmas will have small direct momentum input, yet there is evidence that rotation modifies the onset beta threshold.[8]	At similar plasma shape, vary plasma rotation magnetic breaking (NSTX, DIII-D) or co/counter balance (DIII-D). Look for systematic variation in threshold with rotation.	Possible reasons for trend include differences in shear effecting seeding, ion polarization effects, classical tearing seeding. Continuation of already established NSTX/DIII-D collaboration.[9]
3.3.3 Marginal beta-limit by β_N rampdowns.[10]	Allows critical small island physics to be studied separately from trigger physics.	Generate the 2/1 and/or 3/2 NTM, then ramp-down β_N by decreasing the heating power.	Continuation of already established NSTX/DIII-D collaboration.
3.3.4 NTM Stabilization By External Current Drive	If NTMs cannot be avoided in ITER via control of seed islands, then active stabilization techniques will be required.	Compare direct mode stabilization via ECCD (DIII-D)[11,12] with stabilization via modification of the equilibrium current profile (Δ')[13] (LHCD in C-Mod and possibly EBWCD/NBICD in NSTX[14]).	Wide variety of current drive techniques allows important comparisons.
3.3.5 Error field impact on NTM threshold.	Error fields have been observed to lower NTM threshold in JET[15] and DIII-D.[16]	Study impact of small n=1 fields on NTM threshold in similar discharges in C-MOD, DIII-D and NSTX.	

[7] R.J. La Haye, et al, Phys. Plasmas **7**, 3349 (2000)
[8] R. J. Buttery, et al, *Rotation and Shape dependence of Neoclassical Tearing Mode thresholds on JET*, 28th EPS Conference on Contr. Fusion and Plasma Phys.
[9] Collaboration resulted in experiments XP-739 and XP-740 in the FY2007 NSTX run campaign.
[10] O. Sauter, et al, Plasma Phys. Control Fusion **44**, 1999 (2002)
[11] R. J. LaHaye et al, Nuclear Fusion **46**, 451 (2006)
[12] R. Prater et al, Nucl. Fusion **47**, 371 (2007)
[13] C.D. Warrick et al, Phys. Rev. Lett **85**, 574 (2000).
[14] Direct stabilization (O-point current replacement) of NTM by EBWCD in NSTX not anticipated during the FY09-FY13 research period.
[15] R. J. Buttery et. al., Nuclear Fusion **40**, 807 (2000)
[16] R.J. Buttery et al, *Cross-machine NTM physics studies and implications for ITER*, Paper EX/7-1, 28th IAEA conference, Vilamoura, Portugal, 2004

3.4 Disruptions

Disruptions, the sudden release of the plasma thermal energy and rapid quench of the plasma current, have important operational consequences for burning plasma experiments. This importance manifests itself in the discussion of disruption understanding/prediction/mitigation filling ~ 50% of the MHD chapter (Chapter 3) in the Revised ITER Physics Basis.

A typical tokamak disruption has at least two phases: a fast thermal quench, where the thermal energy of the plasma is rapidly lost to the wall, and a current quench, where the plasma current rapidly collapses. The former can lead to extreme impulsive heat-loading on plasma facing components, while the latter can lead to large forces on the in-vessel components due to eddy-currents. Furthermore, if the equilibrium vertical position control fails, the plasma can move rapidly up or down in the vessel. “Halo currents” then occur when currents flow between the plasma and the vessel components that the plasma comes in contact with; these halo currents can also lead to major forces on in-vessel components. The studies of these areas (the thermal quench, the current quench, and halo currents) are internationally coordinated though the ITPA Disruption Database activity[17], to which all three tokamaks contribute.

Given the damage a large disruption can cause, it is clear that prediction and mitigation techniques are necessary. The mitigation area in particular has seen substantial collaboration in the implementation and modeling of massive gas jet (MGI) mitigation techniques (3.4.2 below). Disruption predictor methods using neural networks have shown signs of progress, but much work remains in order to develop acceptable prediction methods (3.4.3 below).

[17] J.C. Wesley et al, Disruption Characterization and Database activities for ITER, paper IT/PI-21, 2006 IAEA FEC, Chengdu, China

TABLE 3-3. Suggested Disruption Collaborations.

Topic	Importance	Collaboration Scheme	Additional Comments
3.4.1 Divertor heat loading/ wall erosion during disruptions.	Necessary for predicting divertor melting/ablation in burning plasma devices.	• Use fast IR cameras to measure the spatial broadening and timescale of the divertor heat pulse during disruptions. • Comparison of high-Z/low-Z divertor sputtering/erosion during disruption (C-Mod/DIII-D)	Present ITER projections assume that all heat is conducted to divertor, with broadening factor of 1-10,[18] though some experimental results indicate much larger SOL broadening.[19,20] Parameters are included in ITPA disruption database.
3.4.2 Disruption Prediction	In order to initiate a fast shutdown/mitigation procedure, it is necessary to predict disruption onset with sufficient look-ahead time to take action.	Develop disruption predictor algorithms based on common set of diagnostics,[21] and test algorithms across different devices. Study if disruption predictor is specific to each machine, or to each type of discharge/disruption.	Test of common neural-net algorithm on JET and ASDEX was encouraging, but other approaches may be needed for ITER.[22] Range of three devices allows extended test of prediction algorithms.
3.4.3 Disruption Mitigation via MGI[23]	Limited unmitigated disruptions allowed for ITER, essentially none for DEMO.	• Continue simulations (DIII-D, NSTX, and C-Mod)[24] and experiments (DIII-D and C-Mod) related to MHD activity[25] during MGI and other delivery techniques. • Experimentally study the scaling of impurity mixing rate with machine size.	DIII-D & C-Mod already engaged in collaboration on this topic.[26]

18 M. Sugihara, et al., Nuclear Fusion **47** 337 (2007)
19 G. Pautasso, et al., 31st European Physical Society Conference on Plasma Physics, London, 2004, Paper P-4.132
20 V. Ricardo, S. Walker, and P. Noll, Nuclear Fusion **42**, 29 (2000)
21 Disruption predictor could include neural networks, formula based determination of the β_N limit, active MHD spectroscopy (see Phys. Rev. Lett. **93**, 135002 (2004)) to actively control β_N near stability limits, or a combination of these and other techniques.
22 C.G. Windsor, et al, Nuclear Fusion **45**, 337 (2005)
23 D.G. Whyte et al, Phys. Rev. Lett. **89**, 055001 (2002)
24 V.A. Izzo, Nuclear Fusion **46**, 541 (2006)
25 E.M. Hollman et al, Phys. Plasmas **14**, 012502 (2007)
26 R. Granetz et al., Nuclear Fusion **47** 1086 (2007)

3.5. Error Fields & Error Field Locked Modes

Error fields result from non-axisymmetries in the design of tokamak systems and small imperfections in component manufacture and assembly. Uncorrected, these fields can results in locked modes which tend to limit the performance or cause a disruption, and which may present problems during current ramp and early flat-top phase of ITER discharges[27]. Error field locked modes are understood to arise from the braking torque applied to the plasma from the static resonant error field, which can bring the rotating plasma to rest and allow islands to form. In the 1999 IPB[28] an empirical scaling was reported for the $n=1$ error field threshold B_{pen} above which locked modes are induced: $B_{pen}/B_t \propto n^{\alpha_n} B_t^{\alpha_B} q_{95}^{\alpha_q} R^{\alpha_R}$. However, uncertainty remains about the exponents on B_T and R in this expression (3.5.1 below), as well as the role different m-numbers in determining the $n=1$ error field threshold (3.5.2 below). Also, this scaling was derived for low β plasmas, and does not include any dependence on β or plasma rotation.

Nonresonant error fields also apply torques on a plasma. For instance, nonresonant error fields may counterintuitively cause acceleration of the plasma rotation[29], or may decelerate the plasma and assist in the process of mode locking. The physics of nonresonant braking is under intense collaborative study (3.5.3 and 3.5.4 below) and our understanding is making rapid progress[30].

Even in the absence of mode locking, the braking torque from error fields (both resonant and nonresonant) can reduce the plasma rotation with deleterious effects on the stability of NTMs, RWMs, and the energy confinement. Hence, the research topics suggested below are of crosscutting importance to all areas of tokamak performance.

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[27] T.C. Hender et al, Nuclear Fusion **47**, S128 (2007)
[28] ITER Physics Basis, *Nucl. Fusion* **39** 2137 (1999)]
[29] A. Cole, C. Hegna, and J. Callen, PRL **99**, 065001 (2007)].
[30] W. Zhu, et al, Phys. Rev. Letters **96,** 225002 (2006)

TABLE 3-4. Suggested Error Field Collaboration Activities.

Topic	Importance	Collaboration Scheme	Additional Comments
3.5.1 Verification of B & R scaling of locked-mode threshold.	Critical in order to predict required error-correction for future devices.	Dimensionless identity experiments between DIII-D, C-Mod, and JET to resolve locked mode scaling. NSTX data add to the scaling to help resolve the roll of aspect ratio.	Collaboration between C-Mod, DIII-D, and JET[31] already helping to resolve questions about scaling based on previous data.[32]
3.5.2 Determination of Importance of Harmonic Content on Locked Mode threshold	Viscous and toroidal coupling impact the rotation breaking at the q=2 surface, resulting in a need for multi-mode error-field correction.	• Utilize non-axisymmetric coil capabilities to improve common prediction of sideband terms[33] in threshold prediction. • Study variation in locking threshold when low-order rational surfaces (q=1, q=2) are excluded.	• All three devices will have the capacity for extensive n=1, m≥1 error field control, but different aspect ratios and absolute field strengths.
3.5.3 Error-Field Threshold for n>1	Though the threshold for n>1 error fields is higher, they are also more likely to be intentionally introduced.	• Conduct experiments similar to 4.1, but with n>1 error fields. • Use results to further asses the roles of resonant vs. nonresonant rotation damping.	• Higher-n error fields may be intentionally introduced for purposes of ELM mitigation.[34] • Large flexibility for n>1 fields enables these experiments. • Preliminary results in NSTX and DIII-D indicate improved plasma performance with n=3 correction.
3.5.4 Plasma Rotation Braking Physics	Important to improve understanding of plasma rotation damping, due to both external error fields and plasma modes.	Study the effect of externally applied fields on similar plasmas, and compare to theories or resonant and non-resonant rotating breaking (for instance, the Fitzpatrick model[35] or NTV[36])	

[31] S.M. Wolfe et al, Phys. Plasmas **12** 056110 (2005)
[32] R.J. Buttery et al, Nuclear Fusion **39**, 1827 (1999)
[33] J.T. Scoville & R.J. La Haye, Nuclear Fusion **43**, 250 (2003)
[34] T. E. Evans, et al, Nuclear Fusion **45**, 595 (2005)
[35] R. Fitzpatrick, Nuclear Fusion **33**, 1049 (1993)
[36] K.C. Shaing et al, Phys. Fluids **29** 521 (1986)

3.6 Sawtooth physics/control

The behavior of the core plasma under the influence of the $m = 1$ kink is key to several important overall performance questions. For example, the attractive "hybrid" scenario is one where sawteeth are avoided or partially stabilized by programming of the magnetic shear profiles. The sawteeth themselves constitute an important constraint on the plasma inductance. It has been observed that energetic particles can play an important role in stabilizing or destabilizing the sawtooth precursors or changing their characteristics. The theory of the sawtooth mechanisms has been developed to a high level of subtlety, but definitive experimental validation of all the details has not yet been established. This is the main objective of topic (3.6.1 below).

TABLE 3-5. Suggested Sawtooth Collaboration Activities.

Topic	Importance	Collaboration Scheme	Additional Comments
3.6.1 Validation of sawtooth trigger models.	Sawtooth instability may be useful in expelling impurities in a burning plasma, but may also trigger dangerous NTMs.[37]	Study the effects of q-shear and fast ions on sawtooth precursors in NSTX and DIII-D (and perhaps C-Mod).	Extensive MSE & fast-ion diagnostics on DIII-D and NSTX facilitate these studies.

3.7. Axisymmetric Control

The current ITER design is close to vertical stability limits for dynamic axisymmetric control under some conditions, yet ITER must avoid this loss of control as a high priority, because of the potential damage to the machine of Vertical Displacement Events. It has been proposed to use vertical position feedback control algorithms more elaborate than those typically in use at existing tokamaks, in order to enhance the vertical stability. Such algorithms can be used in current experiments, and must be explored to demonstrate both that they are robust to disturbances and noise, and to find out the extent to which they do offer substantial stability enhancement (3.7.1 below).

Another important area of axisymmetric control in ITER arises because for cost-efficiency it must operate close to the current (and voltage)capabilities of its coils. If a scenario drives the coils to their current limits, then a linear controller of the type usually used in present tokamaks will saturate the current, with the result that the coil current becomes effectively uncontrolled for small perturbations. This will likely compromise plasma shape control and possibly even vertical stability. Therefore more "intelligent" algorithms must be developed which avoid coil current saturation by adaptive programming that takes the current limits into account. The experimental study of such non-linear adaptive algorithms is in its infancy, but all three facilities possess control systems capable of implementing them. The outcome is anticipated to be an experimental

[37] T.C. Hender et al, Nuclear Fusion **47**, S128 (2007)

demonstration and the validated theoretic understanding of improved control algorithms that will allow future tokamaks to take full advantage of their coil capabilities without fear of operational failure through saturation (3.7.2 below).

TABLE 3-6. Suggested Axisymmetric Control Collaboration Activities.

Topic	Importance	Collaboration Scheme	Additional Comments
3.7.1 Vertical Stability Control Robustness Studies	ITER relies on feedback of its main PF coils, outside a double-walled vacuum vessel for vertical control. High-order position-controllers are proposed to improve its stability. Current tokamaks have not yet demonstrated definitively the benefit of such controllers.	The benefits of various proposed schemes will be evaluated for similar plasmas in DIII-D and C-Mod, which exemplify higher and lower coil flexibility respectively.	Although a near term ITER issue, this will probably remain an important topic for the future.
3.7.2 Axisymmetric Tokamak Control Near Machine Limits	To utilize efficiently the full capability of the magnets of a facility like ITER requires operation close to coil current (and possibly voltage) limits. However, current saturation is inherently non-linear and can lead to loss of shape control and even to vertical instability. Future machines will need smart control systems that avoid the problems of saturation and non-linearity while operating close to limits.	Compare schemes developed or favorably evaluated by one machine on the other, to verify the portability of the solutions to tokamaks with different characteristics.	All three facilities have sophisticated axisymmetric control systems capable of applying complex control algorithms in real time.
3.7.3 Axisymmetric plasma response physics	Further validation of plasma response models for designing controllers in low aspect ratio	Triggered VDE experiments combined with equilibrium perturbation experiments.	Continuation of an existing PCS development collaboration between DIII-D and NSTX.

4. COORDINATED RESEARCH ON WAVES & ENERGETIC PARTICLES

P. Bonoli, E. Fredrickson, and R. Prater

4.1 Introduction

The study of energetic particle driven instabilities on present machines is motivated by the role of non-thermal particle distributions in many plasma heating schemes, including plasma heating by fusion α's, as well as some forms of plasma heating by waves. For example, ICRF waves may accelerate ions to high energies, which may then, in turn, excite energetic particle modes resulting in redistribution of the fast particles. However, this is not a problem for many forms of plasma heating with waves, particularly those schemes which heat electrons (ECH/ECCD/EBW). So mostly, Waves and Energetic Particles represent two distinct areas of research and are treated separately below.

4.2 Waves

In the area of Waves, some aspects of wave research are naturally coordinated just by the physics requirements. The application of ECRF on DIII-D, LHRF on C-Mod, EBW on NSTX and ICRF on DIII-D, C-Mod, and NSTX follows what is naturally dictated by physics and technology considerations. ECH is well suited to DIII-D, but it isn't very suitable for C-Mod because the highest available source frequency is too low for most plasma conditions, and for NSTX the cutoff density is too low for ECH to be of interest. EBW is under development on NSTX because of the low cutoff density of ECH, but it doesn't provide new opportunities for DIII-D or C-Mod. DIII-D does have a high power ECH system and can run over dense plasmas at f_{ce}, so some limited coordinated research is possible, for example in the high density H-mode regime, however, with existing equipment, EBWs can be studied on DIII-D only with great difficulty and little flexibility. LHCD is perfect for C-Mod, but accessibility is too limiting for LHCD to be desirable on lower field devices like DIII-D and NSTX. Coordination between devices isn't relevant in those regimes. ICRF is of substantial interest for all three devices:

<u>HHFW/ICRF Research</u>

All three facilities, C-Mod, NSTX and DIII-D have ICRF systems, creating significant opportunities for coordinated ICRF physics research that would directly benefit ITER and other future machines that plan to use ICRF. C-Mod has RF transmitters that allow fundamental and second harmonic heating at typical C-Mod magnetic fields. DIII-D has RF transmitters that can operate at 60-120 MHz, corresponding to the $4\Omega_D$-$8\Omega_D$ at $B_T(0) = 2\mathrm{T}$. Areas of coordinated research could include RF coupling (e.g. surface wave propagation and edge parametric decay) and wave propagation, heating and current drive. The experimental research should be closely integrated with ongoing RF modeling development, in particular through SciDAC and the USBPO. The wide operating regime (minority to high harmonic) represented by ICRF experiments on C-Mod, DIII-D, and NSTX offers an exhaustive testbed for assessing the predictive capability of the ICRF wave propagation, absorption, and coupling codes being developed through initiatives such as SciDAC. The absorption of waves by the different components of the plasma, including particularly minority species or the energetic ions generated by neutral beam injection or by fusion reactions in a reactor, is a topic of great importance to ITER that can be well addressed through this combination of coordinated research of these devices and continuation of the SciDAC modeling activities.

A principal goal of the research on all three machines is to study the compatibility of ICRF with high performance plasmas. This requires addressing many issues, including reduction of impurity production and penetration, understanding the role of RF sheaths on impurity generation and associated convective cells on impurity penetration, and finally to develop designs for better antenna configurations using materials which reduce impurity generation. Development of antennas with reduced tendency to arc at high voltage, robust arc detection, and resilience to ELMs is required. Antennas which can couple high power through increased gaps between the plasma and the wall is needed due to the energetic particle content of many high performance plasmas. Technology developments of this nature are needed for ITER as well as present devices.

Efficient heating of high performance plasmas requires good coupling, the ability to deliver maximum power and voltage over wide range of plasma conditions. On NSTX and C-Mod there are plans to develop methods which minimize the impact of ELMs on RF sources due to changes in reflected power as the antenna matching changes on the ELM timescale. This will require the development of real time matching control and understanding and modeling the influence of the edge density profile, edge neutrals and magnetic field on antenna power and voltage limits. Alternatively, work on DIII-D has shown that if the RF frequency is fixed, the RF sources can be robustly protected against power reflected by ELMs through passive means.

For all devices, the ability to use internal measurements of the RF wave field will be especially valuable for the validation of RF codes such as the AORSA and TORIC electromagnetic field solvers, the 3D electromagnetic antenna code TOPICA, and the orbit code ORBIT-RF. These codes are needed to model minority and second harmonic heating in H-mode plasmas. The development and validation of a coupling model, from the antenna to the plasma core is needed, which includes RF sheath effects and the excitation of surface waves and coaxial modes.

The ICRF systems can be used for electron heating and current drive as well as for ion heating. Current drive is important for current profile tailoring in AT regimes, central seed current generation in high bootstrap fraction regimes and sawtooth instability control in high performance H-modes. High harmonic electron heating and CD are suitable for NSTX, but an area requiring additional research is the potential problem of damping on fast NBI ions in DIII-D and NSTX. Ion cyclotron current drive (ICCD) and mode conversion current drive (MCCD) for sawtooth control for high performance H-modes will be investigated on C-Mod. Ion cyclotron heating could be used on DIII-D, but there is high power NBI already available, so the main interest is in central electron heating and FWCD, where ion damping is a loss mechanism.

An area as important as the physics of RF coupling, heating and current drive is to improve the technology of RF power production. It is important to develop techniques which can reduce RF antenna conditioning time and extend the transmitter tube lifetime. Experiments will be dedicated to characterization of antenna voltage and power conditioning. Likewise, the influence of boronization (C-Mod) on antenna conditioning and HHFW heating compatibility with the liquid lithium divertor (NSTX) will be investigated.

4.3 Energetic Particle Research

Great strides have been made in understanding the linear stability of a wide variety of energetic particle driven modes. However, the interest has always been in the role that these modes play in the redistribution of fast ions, and possibly in the channeling of fast ion power flow (alpha-channeling). Further progress requires a more aggressive effort to study the nonlinear dynamics of mode saturation. The five year programs on all three devices focus to a large extent on the non-linear dynamics, interaction of energetic particle driven modes and redistribution of energetic particles. NSTX has a tremendous capability here because it will have excellent diagnostics for both the fluctuations and the fast ions and it has such fascinating variety of nonlinear behavior (chirping, avalanches). DIII-D will revisit comparative studies between NSTX and DIII-D that exploit the mutual excellent diagnostic capabilities, but different nonlinear dynamics.

Multimode drive fast ion redistribution measurements with both NPA and FIDA diagnostics should be performed on NSTX and DIII-D, as well as CNPA measurements of fast ion redistributions on C-Mod. The effect of the ion redistribution on NBI current drive will also be studied. The direct measurement of energetic particle redistribution will be complemented by detailed internal measurements of the mode structure and amplitudes with reflectometer or PCI diagnostics, allowing careful validation of models for energetic particle redistribution.

A proposed diagnostic upgrade to the DIII-D Mirnov coil system will complement the NSTX measurements of mode polarization. Proposed extensions of the Mirnov coil array on NSTX will allow studies of the effect of aspect ratio on the poloidal localization of CAE modes in cross-machine experiments on DIII-D and NSTX. Also of interest is a cross machine study of the BAAE excitation with DIII-D.

With the planned antenna excitation of Alfven Eigenmodes (AE) on NSTX we can perform AE damping rate measurements that can be coordinated with C-Mod, where such an antenna is already installed. These measurements will be valuable for the benchmarking of Nova-K calculations of mode damping rates, as well as help to identify natural eigenmodes of the plasma, as opposed to eigenmodes created by the presence of a significant fast ion population (EPMs). The measurements will identify stable intermediate *n* Alfven Eigenmodes and measure their damping rate in the presence of fast ions to understand the effects of fast ions on mode stability. Of particular interest is the possible study of the radiative damping by AE modes.

TABLE 4-1. HHFW/ICRF Coordinated Research Activities.

Research Topic	**Coordinated Joint Experiments**	**Complementary Research Activities**	**Additional Comments**
Validation of RF codes	Joint experiments with DIII-D, NSTX, and C-Mod. Uses wide range of plasma parameters available over three machines to test RF codes. International element: Couples to ICRF research on JET and provides tools for design of ITER RF system.	Each program contributes to this research, coordinated with ITPA C-Mod: ? NSTX: Studies of HHFW coupling propagation and damping physics. DIII-D: ?	High, immediate ITER relevance. - ICRF heating Physics complementarity of devices: - Wide range of operating fields, coupling geometry?
Mitigation of ELMs impact on RF coupling	Modifications to transmission lines, coupling systems to minimize reflected power effects on transmitters.	NSTX: modifications planned to coupling network. C-Mod: ? DIII-D: Some mitigation capability already present?	High, immediate ITER relevance - Requirement for ITER operations in H-mode regime?
Coupling across wider plasma-antenna gaps		NSTX: antenna modifications planned to improve coupling. C-Mod: ? DIII-D:	High, immediate ITER relevance

TABLE 4-2. Coordinated Research on Energetic Particles.

Research Topic	**Coordinated Joint Experiments**	**Complementary Research Activities**	**Additional Comments**
Multi-mode transport studies	Joint experiments with DIII-D, NSTX	Each program contributes to this research, coordinated with ITPA **C-Mod:** RF driven fast ions, PCI measurements of mode structure, CNPA and FILD diagnostics **NSTX**: Studies of HHFW coupling propagation and damping physics. **DIII-D:** FW, NBI, and ECH systems coupled with detailed measurements of mode structure, FIDA measurements	**High, immediate ITER relevance**. - RF heating of tails **Physics complementarity of devices**: - Wide range of operating fields, mode coupling, heating schemes, and geometry?
Chirping and impact of fast ion transport	Joint experiments with DIII-D, NSTX	Study importance of aspect ratio on tendency for chirping	
Measurements of mode linear growth rates through external excitation	Joint experiments with NSTX and C-Mod	Study TAE damping rates over wide range of parameters.	

5. PLASMA BOUNDARY INTERFACES

V. Soukhanovshii, P. West, and D. Whyte

5.1 Introduction

The Plasma Boundary Interfaces is a multi-scale, multi-discipline fusion science area that includes plasma, atomic, and material physics as well as a number of engineering disciplines. The key scientific challenge is to develop and demonstrate plasma regimes and surface material interfaces compatible with a burning plasma core and plasma facing surface heat loads and erosion, as noted in the FESAC Priorities Subpanel Report on "Scientific Challenges, Opportunities, and Priorities for the U.S. Fusion Energy Sciences Program" (April 2005). The Report defined four major research thrusts, following a natural division of the boundary-plasma interface into four unique regions governed by their own physical processes: the pedestal region, the scrape-off layer region, the first wall and divertor region, and the composition of divertor and first wall. These research directions have been used in the discussions as a basis of the proposed program for future research coordination and collaborative activities for the three US fusion facilities. The selection criteria for the research program themes also included 1) relevance to burning plasma research 2) U.S. experimental and modeling expertise 3) opportunity for contribution from all three facilities.

5.2 Resources and facilities for coordinated research

The Alcator C-Mod, DIII-D, and NSTX Five Year Boundary Physics research programs focus on the experimental areas where each machine can exploit its strength in diagnostic and facility capabilities. Alcator C-Mod is a compact, high-field, high-Z (Mo/W) wall tokamak with unique properties such as the high parallel power density (500 MW/m^2), divertor plasma densities spanning that of ITER, and a short mean free paths in SOL and divertor ideal for accessing ITER regimes. The DIII-D tokamak, with all-graphite wall and well developed density control tools, has several ITER-relevant plasma and machine capabilities: core performance; pedestal collisionalities; shape flexibility; pulse length approaching first order wall time constants, and unique magnetic coils for edge localized mode (ELM) control or suppression. The NSTX device, being a low-aspect ratio spherical torus with low magnetic field, has unique capabilities for plasma shaping spanning a wide range of triangularities and elongations, and novel lithium particle handling techniques. Unique plasma capabilities in NSTX include ITER-relevant divertor heat fluxes, Type V ELMs, and the enhanced pedestal H-mode regime.

A number of on-going inter-machine collaborations and collaborations through the ITPA Edge and Pedestal (PEP) and Divertor and Scrape-Off Layer (DSOL) Groups benefit the current Alcator C-Mod, DIII-D, and NSTX research programs. Examples of the existing joint research efforts are the SOL radial transport, disruption mitigation with massive gas injection, deposition studies, and pedestal scaling studies between Alcator C-Mod and DIII-D; blob transport studies and comparison with turbulence models, studies of small ELM characteristics, and RF edge effect studies between Alcator C-Mod and NSTX; and ELM control with resonant magnetic perturbation (RMP), as well as edge stability and turbulence studies between DIII-D and NSTX.

The proposed research themes for future coordination and collaborative activities include 1) Particle balance and inventory studies, 2) ELM control, with emphasis on ELM suppression using RMP, 3) Scrape-off layer heat flux distribution physics, 4) Plasma-material interaction (PMI) diagnostic development. While these themes address the challenges and needs of future fusion devices, it was noted that some research topics are beyond the present capability of Alcator C-Mod, DIII-D and NSTX. For example, the long-pulse issues due to thermal and particle wall equilibrium, and ITER-specific mixed material issues cannot be presently studied.

5.2.1 Particle balance and inventory studies

Control of D/T fuel and D/T fuel wall retention is a critical issue for both present and future fusion devices. The three facilities are well positioned for both coordinated and complementary research in this area owning to their plasma facing component (PFC) materials (Li, C, Mo, W), developed diagnostic systems, and available pumping and fueling tools. The Alcator C-Mod program includes on-going studies of D/T retention, erosion/redeposition, recovery, and wall-conditioning techniques with molybdenum and tungsten PFCs, as well as a new upper divertor cryo-pump and advanced surface diagnostics. In DIII-D, the poloidal deposition and fueling profiles, the role of poloidal drifts in particle exhaust, and the role of surface temperature on particle retention are studied in all-carbon PFC environment, contributing to the ITER-relevant issue of tritium control with carbon walls. DIII-D also has a comprehensive set of edge diagnostics for particle balance studies, the DiMES probe for PMI studies, and upper and lower divertor cryopumps. NSTX, also a carbon PFC machine, is experimenting with evaporated lithium coatings for pumping and a high-pressure supersonic gas jet for plasma fueling. Near-term plans on NSTX include installation of a liquid lithium divertor module to study its effect on density, plasma performance, recycling and divertor heat flux.

The proposed coordinated and collaborative activities in the particle balance and inventory area are aimed at the development of common approach to experiments and analysis. In particular, the impact of wall conditioning techniques (boronization, glow discharge cleaning), particle (gas) balance models, and interpretation of diagnostics (e.g. "window frame technique") will be compared, with a focus on key differences in high-Z metallic and low-Z plasma facing materials. The inter-machine FY09 Joule milestone summarizes the proposed activities and the expected outcome: "Conduct experiments on major fusion facilities to develop understanding of particle control and hydrogenic fuel retention in tokamaks. In FY09, FES will identify the fundamental processes governing particle balance by systematically investigating a combination of divertor geometries, particle exhaust capabilities, and wall materials. Alcator C-Mod operates with high-Z metal walls, NSTX is pursuing the use of lithium surfaces in the divertor, and DIII-D continues operating with all graphite walls. Edge diagnostics measuring the heat and particle flux to walls and divertor surfaces, coupled with plasma profile data and material surface analysis, will provide input for validating simulation codes. The results achieved will be used to improve extrapolations to planned ITER operation."

5.2.2 ELM control, with emphasis on ELM suppression using RMP

The control of ELM particle and energy losses with minimal degradation of core and pedestal is an important task for ITER and future fusion devices. Large (e.g. Type I) ELMs can lead to PFC erosion, thermal stress, and limit the divertor and first wall PFC lifetime. The ELM control techniques that have shown promise include RMP, pellet injection, and development of small ELM H-mode regimes. In the ELM and pedestal area all three facilities have detailed research plans that take advantage of the unique facility capabilities. The proposed coordinated activities will focus on the implementation of common modeling and analysis tools across devices. Experiments on ELM control with RMP have been on-going on DIII-D and NSTX, and Alcator C-Mod is presently considering installing non-axisymmetric perturbation coils. The DIII-D has been leading the effort by demonstrating Type I ELM suppression using $n = 3$ RMP for a range of shapes and collisionalities, and studied impurity transport, plasma rotation, and divertor heat flux during $n = 3 - 6$ and mixed n RMP using its I-coils. On NSTX, six midplane coils can be used for $n = 1 - 6$ RMP, including a unique $n = 6$ low field side capability. Also on NSTX, a small (Type V) ELM regimes at lower collisionality and stronger shaping have been developed as a prototype high-performance target. A number of analysis tools, including ELITE pedestal stability analysis, and numerical codes for coil and mode configuration optimization are used to analyze DIII-D and NSTX experiments. Future plans encompass the installation of new high and low field side RMP coils on DIII-D, and further RMP experiments with optimized RMP spectra on DIII-D and NSTX. Alcator C-Mod is planning to continue studies of the pedestal relaxation physics in ELM-free and small ELM regimes with high-triangularity shapes.

5.2.3 Scrape-off layer heat flux distribution physics

The ability to understand and predict transport and turbulence in the SOL region is essential for divertor and wall heat flux management in ITER and future devices. Whereas progress has been made in experiment-based understanding of SOL and divertor heat and particle transport in recent years, the ability to reliably predict heat and particle fluxes for future devices is still lacking. Each facility has an active research program in this area supported by comprehensive and often unique profile and fluctuation diagnostics, and on-going modeling efforts with state-of-the-art fluid and turbulence codes. Common diagnostics include high-resolution SOL and divertor measurements with infrared and visible cameras, reciprocating and tile Langmuir probes, thermocouples, bolometers, gas-puff imaging diagnostics for turbulence visualization, Thomson scattering and interferometry. The proposed coordinated and collaborative activities will focus on comparison of various experimental techniques for q_{peak}, λ_q, λ_T, λ_n measurements (e.g. IR cameras, Langmuir probes, thermocouples), and comparison between obtained empirical scalings of q_{peak}, λ_q, λ_T, λ_n as functions of key plasma parameters (e.g. n_e, ν^*, P_{SOL}, I_p, B_t). Complementary efforts will also be pursued. Alcator C-Mod plans to improve diagnosis of heat flux pattern in divertor, and study $e-i$ equipartition and role of electron / ion conduction. On DIII-D, it is planned to install 3 IR/Visible TV periscopes for full toroidal/poloidal coverage, thermocouple calorimeters, and a divertor bolometer array toward high-resolution SOL/pedestal profiles and fluctuation measurements. A gas puff imaging system for transport and turbulence studies is also being considered. On NSTX experiments and modeling will be performed toward the FY08 milestone on variation and control of heat flux in the SOL, and experiments will be attempted to control SOL transport by induced poloidal electric field.

5.2.4 Plasma-material interaction (PMI) diagnostic development

Improvements in PMI diagnostics are needed for better understanding of erosion, retention, and deposition processes in present and future devices. The proposed activities in this area will be mostly complementary: Alcator C-Mod, DIII-D and NSTX would provide test environment for in-situ PMI diagnostics development. On Alcator C-Mod, extensive boronization studies are underway. New advanced PMI diagnostics are also under development: a surface-science station was installed in 2007, and accelerator (RFQ) in-situ ion beam analysis and radio-isotope in-situ surface analysis (ARRIBA) are coming into operation. On DIII-D, in addition to the on-going successful DiMES and MiMES probe research program, quartz microbalance installation is underway, and in-situ plasma facing surface deposition measurements are being considered. Dust studies using the DiMES probe, video imaging, and Mie scattering techniques will be performed. On NSTX, on-going dust transport studies with unique dust diagnostics - stereoscopic imaging and electrostatic dust detectors, and deposition studies with quartz microbalances will continue. A DiMES-like system for lithium deposition studies is being considered.

Appendix A: Agenda for the National Tokamak Planning Workshop

NSTX, C-Mod, and DIII-D National Tokamak Planning Workshop
MIT September 17-19, 2007
Agenda

Monday

10 am OFES Perspective

Overview of Tokamak Facility Plans (30min presentation, 5min questions)
10:15 am C-Mod 5- year Plan Overview:E. Marmar
10:50 am DIII-D 5 year Plan Overview: M. Wade
11:25 am NSTX 5-year Plan Overview: M. Ono
12:00 pm *Lunch*

Topical Area Coordination Discussions:

Topical Area contents as outlined in FESAC Priorities Subpanel Report on *"Scientific Challenges, Opportunities, and Priorities for the U.S. Fusion Energy Sciences Program,"* except where noted.
http://www.ofes.fusion.doe.gov/more_html/FESAC/PP_Rpt_Apr05R.pdf

Speakers and Discussion Facilitators are noted.

Integrated Scenario Research
1:15 pm Facility plans (DIII-D: Luce, C-Mod: Hubbard, NSTX: Menard)
(15 min presentations, 5 min for clarifying questions)
2:15 pm Summary of preliminary coordination activities (15 min presentation)
(J. Ferron, S. Wolfe, D. Gates)
2:30 pm Discussion: Integrated Scenario Research Plans
(C. Kessel, G. Sips)
4:30 pm *Break*

Multi-scale Transport Physics Research
(*includes pedestal scaling, L-H transition, and internal barriers)*
4:45 pm Facility plans (C-Mod: Greenwald, NSTX: Kaye, DIII-D: Burrell)
(15 min presentations, 5 min for clarifying questions)
5:45 pm Summary of preliminary coordination activities (15 min presentation)
(S. Scott, D. Mikkelson, E. Doyle)
6:00 pm Discussion: Multi-scale Transport Physics Research Plans
(P. Terry, W. Dorland)
7:45 pm *Adjourn*

Tuesday

<u>Macroscopic Plasma Physics Research</u>
(*includes disruption avoidance and mitigation)*

9:30 am	Facility plans (NSTX: Sabbagh, DIII-D: Strait, C-Mod: Granetz) (15 min presentations, 5 min for clarifying questions)
10:30 am	Summary of preliminary coordination activities (15 min presentation) (S. Gerhardt, A. Garofalo, I. Hutchinson)
10:45 am	<u>Discussion:</u> Macroscopic Plasma Physics Research Plans (C. Hegna, R. Buttery) (*break included)*
12:15 pm	*Lunch*

<u>Research on Waves and Energetic Particles</u>

1:15 pm	Facility plans (NSTX: Taylor, C-Mod: Wukitch, DIII-D: Nazikian) (15 min presentations, 5 min for clarifying questions)
2:15 pm	Summary of preliminary coordination activities (15 min presentation) (E. Fredrickson, P. Bonoli, R. Prater)
2:30 pm	<u>Discussion:</u> Research Plans for Waves and Energetic Particles (W. Heidbrink, C. Philips)
4:15 pm	*Break*

<u>Research on Plasma Boundary Interfaces</u>
(*includes ELM control such as pellet pacing and RMP)*

4:30 pm	Facility plans (C-Mod: Lipschultz, DIII-D: Allen, NSTX: Maingi) (15 min presentations, 5 min for clarifying questions)
5:30 pm	Summary of preliminary coordination activities (15 min presentation) (D. Whyte, P. West, V. Soukhanovskii)
5:45 pm	<u>Discussion:</u> Plasma Boundary Interfaces Research Plans (T. Rognlien, R. Nygren)
7:30 pm	*Adjourn*

Wednesday

<u>Summary Reports of Discussions</u>

	(15 min presentations, 5 min for questions)
9:30 am	Macroscopic Plasma Physics Research (C. Hegna, R. Buttery)
9:50 am	Waves and Energetic Particles (W. Heidbrink, C. Philips)
10:10 am	Integrated Scenario Research (C. Kessel, H. Zohm)
10:30 am	*Break*
10:45 am	Plasma Boundary Interfaces (T. Rognlien, R. Nygren)
11:05 am	Multi-scale Transport Physics Research (P. Terry, W. Dorland)
11:25 am	*Adjourn*

Appendix B: Presenters and Workshop Responsibility

Tokamak Planning Workshop Presenters

Science Area	Facility Plan Presentations	Preliminary Coordination	Topical External Facilitators
Program Overview	Marmar (C-Mod) Wade (DIII-D) Ono (NSTX)		
Integrated Scenario Research	Hubbard (C-Mod) Luce (DIII-D) Menard (NSTX)	Wolfe (C-Mod) Ferron (DIII-D) Gates (NSTX)	C. Kessel, G. Sips
Waves and Energetic Particles	Wukitch(C-Mod) Nazikian (DIII-D) G. Taylor (NSTX)	Bonoli (C-Mod) Prater (DIII-D) Fredrickson (NSTX)	W. Heidbrink, C. Phillips
Macroscopic Research (MHD)	Granetz (C-Mod) Strait (DIII-D) Sabbagh (NSTX)	Hutchinson (C-Mod) Garofalo (DIII-D) S. Gerhardt (NSTX)	C. Hegna, R. Buttery
Multi-scale Transport Research	Greenwald (C-Mod) Burrell (DIII-D) Kaye (NSTX)	S. Scott (C-Mod) Doyle (DIII-D) Mikkelson (NSTX)	P. Terry, W. Dorland,
Plasma Boundary Interfaces	Lipschultz (C-Mod) Allen (DIII-D) Maingi (NSTX)	Whyte (C-Mod) West (DIII-D) Soukhanovskii (NSTX)	T. Rognlien, R. Nygren (R. Doerner)