Report of the 2005 FESAC Facilities Panel, Vol. 1:

Characteristics and Contributions of the Three Major United States Toroidal Magnetic Fusion Facilities

Report of the Fusion Energy Sciences Advisory Committee
August, 2005

Facilities Panel Members

JILL DAHLBURG, Naval Research Laboratory (Chair)
STEVEN L. ALLEN, Lawrence Livermore National Laboratory (Vice-Chair)
RICCARDO BETTI, University of Rochester
STEPHEN KNOWLTON, Auburn University
RAJESH MAINGI, Oak Ridge National Laboratory
GERALD A. NAVRATIL, Columbia University
STEVEN A. SABBAGH, Columbia University
JOHN SHEFFIELD, University of Tennessee
JAMES W. VAN DAM, University of Texas at Austin
DENNIS WHYTE, University of Wisconsin at Madison

Executive Summary

The three major toroidal fusion research facilities in the U.S. have diverse and complementary characteristics, which were developed on the basis of evolving U.S. innovation in fusion energy sciences. Taken together these three facilities provide the U.S. with a very effective presence in the world program of fusion research. Their success has enabled the U.S. to have substantial impact on the direction and progress of the field, including world leadership in understanding fundamental transport processes in magnetically confined plasmas, and continuing optimization of the magnetic configuration for confinement of high pressure plasmas.

Unique and complementary characteristics

The characteristics that make each of these toroidal facilities unique in research capability stem from their initial research motivations:

> C-Mod to understand plasma behavior at very high magnetic field with the plasma pressure and field appropriate for sustaining a burning plasma for energy production, but at a smaller size than a self-heated plasma.
>
> DIII-D to understand and improve plasma confinement and stability as a function of plasma shape and magnetic field distribution in a collisionless plasma.
>
> NSTX (the most recently commissioned, in 1999) to apply advances in understanding of magnetic field configuration to optimize both plasma stability and confinement in a proof-of-principle, low aspect ratio toroidal experiment (the spherical torus).

One standard for judging progress in fusion is the “triple product” of plasma density, temperature, and confinement time. This may be rephrased as the product of plasma β, the square of the magnetic field, and confinement time. (The dimensionless quantity β is the ratio of the plasma pressure to the magnetic field pressure.) In this rephrased form, the triple product delineates the three approaches to fusion foci that are embodied in the three major facilities of the U.S. Specifically, Alcator C-Mod is the highest magnetic field, diverted tokamak in the world; DIII-D pursues the direction of high confinement through the development of advanced tokamak operational regimes with long confinement times; and NSTX pursues the direction of very high β, resulting from its extreme toroidicity (i.e., low aspect ratio). These are three distinct dimensions that span the operational parameter space for toroidally confined plasmas. Moreover, while these three facilities are distinctly different in some regards, they have a valuable degree of commonality that permits them to compare plasma behavior under similar dimensionless — but different dimensional — parameters (e.g., magnetic field strengths from 0.3 T up to 8 T, all at high performance).

Today, each facility – with its unique capabilities and program of planned research – is a leading element of the world program in magnetic fusion research:

C-Mod (with FY05 annual budget of $22,038,000 U.S.) is one of two tokamaks tied with world's highest tokamak magnetic field and is the only facility capable of studying plasma/ wall interactions and radio frequency heating and current drive in ITER-like geometry with a divertor and magnetic field and plasma pressure characteristic of a burning plasma.

DIII-D (with FY05 annual budget of $55,751,000 U.S.), with its unparalleled plasma transport diagnostics and world-leading capability for plasma shaping and control of major instabilities limiting high pressure plasmas, has established itself as a center for developing long-pulse, high performance advanced tokamak operation.

NSTX (with FY05 annual budget of $34,584,000 U.S.) is the world's most capable spherical torus, exploring high-β plasma stability and confinement at extreme toroidicity (low aspect ratio) and is the major U.S. experiment for concept innovation in magnetic confinement now in operation.

While these three facilities are clearly distinct, they also have a degree of commonality that makes them highly effective as a group. The *commonality* is useful to allow comparisons, while the *distinctive features* of these facilities allow extrapolations. This combination provides the U.S. with a world leadership role in advancing the most important frontier areas in fusion energy sciences research, recently described in detail in the April 2005 FESAC Program Priorities Report.

Some of the most important examples of forefront research led by one or more of these facilities are:

<u>Multiscale Plasma Transport:</u> The U.S. is one of the leaders of the international research effort to determine the underlying instabilities responsible for plasma transport. The U.S. transport program is made strong by DIII-D, which has an integrated program with measurements over the important wavelength ranges of the turbulent instabilities responsible for the observed transport, as well as the most comprehensive gyrokinetic transport model in the world; and by C-Mod and NSTX, which also contribute unique elements to this line of research.

<u>Plasma-Boundary Interfaces:</u> C-Mod and DIII-D have led this research effort in the U.S. The U.S. facilities, taken together with computational modeling, provide a world-leading capability in plasma-boundary science. While it was a U.S. physics insight that identified the controlling role of the plasma parameters at the top of the H-mode pedestal, there yet exists no first-principles model to predict the pedestal parameters for ITER. The U.S. is leading an effort to understand pedestal physics by means of comprehensive and complementary diagnostics and models applied across all three facilities.

<u>Optimization of Magnetic Confinement:</u> NSTX was designed and built to lead the world effort to apply our improved knowledge of magnetic field topology to a new toroidal configuration at very low aspect ratio (the spherical torus). Enhanced shaping has been predicted to increase global plasma stability at high plasma β, and NSTX quickly

demonstrated record levels of plasma β. It provides a unique set of parameters complementary to the higher aspect ratios in C-Mod and DIII-D, valuable for extending parametric studies in transport, MHD stability, and boundary physics.

U.S. toroidal research as a whole

Scientific understanding of magnetic fusion plasmas comes from a combination of theoretical and experimental investigation. Crucial experimental verification of theory requires facilities that can vary plasma conditions over an extended range. No single facility can adequately span all desired variations. The U.S. fusion program has a complementary set of three major experimental facilities that are able to access a very wide range of variations. The healthy synergy of joint experiments conducted between two or more of these facilities, worldwide, contributes to exploiting their full potential to test theory and techniques. With their unique machine capabilities and outstanding diagnostics, all three major U.S. facilities are carrying out world-leading magnetic fusion plasma research.

The world program in fusion research includes major toroidal facilities in Europe (JET, ASDEX-U, TCV, FTU, TEXTOR, Tore Supra, and the spherical torus device MAST) and Japan (JT-60U and the superconducting stellarator LHD). The superconducting tokamaks – K-STAR (Korea), and China's EAST that should have first plasma in early 2006 – will make important contributions to issues of long-pulse operation. The JET and JT-60U tokamaks are each larger than any of the three major U.S. facilities and so can perform studies closer to ITER and fusion power plant parameters; taken together, however, the three U.S. facilities make influential contributions to the larger world effort in key areas such as plasma transport, MHD stability, plasma-boundary science, radio frequency heating and current drive, and energetic particle physics. Among spherical tori, NSTX has exceptional performance and diagnostic capabilities. DIII-D and C-Mod use their unique and complementary capabilities to influence and expand the world research agenda in the areas of transport, advanced tokamak physics, and divertor development. Backed up by the substantial U.S. effort in plasma computational modeling, the research programs planned for the next five years on the three major U.S. facilities emphasize these areas of strength and promise a continued major role as the worldwide fusion effort moves towards ITER experimental operation. Equally important, the three experiments keep the U.S. at the forefront of fusion energy sciences – transport, magnetohydrodynamics, etc – which is of significance beyond the immediate needs of tokamaks.

U.S. toroidal research coordination internationally, impact on ITER design

The U.S. was a founding member (with Europe, Japan, and Russia) of the International Tokamak Physics Activity (ITPA), established in 2000 to coordinate world activity in tokamak research aimed primarily at burning plasma physics. The U.S. is active in this important coordinating body, which is currently headed by a U.S. researcher. Historically, the U.S. facilities DIII-D and C-Mod have been important contributors to and leaders of the international databases on tokamak confinement, plasma disruptions, and divertor physics. In the past few years, coordinated joint experiments on dimensional and dimensionless scaling across devices of different size, magnetic field, and aspect ratio (under the auspices of the IEA agreements) have engaged substantial experimental run time on all three U.S. facilities, and plans have already been made for significantly more of these coordinated international experiments involving the exchange of both data and personnel. These joint experiments are an important precursor to establishing the coordinated experimental program necessary to operate ITER, as well as to guide a number of ITER systems design issues not yet resolved.

The U.S. facilities DIII-D and C-Mod had a substantial impact on the ITER design, which was finalized between 1998 and 2001. Most notable of these impacts are the inclusion of greater shaping capability in the plasma cross-section for advanced tokamak performance (pioneered by research on DIII-D), the reduction of the divertor heat flux to tolerable levels using the concept of a 'detached' plasma (developed and diagnosed by C-Mod and DIII-D), the 'vertical target' design for the ITER divertor in order to facilitate detached plasma operation (initiated by C-Mod), the ITER design criteria for plasma disruption forces (led by C-Mod measurements of current asymmetries during disruptions), and plans for possible active control of MHD instabilities on ITER (based on the discovery in DIII-D of the resistive wall mode as a pressure-limiting mode and its world leadership in demonstrating control of these modes). NSTX was under construction at the time of the ITER design completion; it is now contributing to design improvements and operational implementations for ITER through the ITPA.

Contributions to fusion science and vitality of U.S. fusion program

The recently published FESAC Program Priorities Report describes ten-year plans for fusion science research that show the three U.S. major facilities as key contributors or leaders for most of the critical issues in the Scientific Campaign areas of macroscopic plasma physics, multi-scale transport, plasma-boundary interfaces, waves and energetic particles, and fusion engineering science. What may not be so clear from that report is that the three U.S. facilities are also national centers for education and integration of national competency across the broad range of fusion science. All are operated as "national research centers" with substantial numbers of university collaborators. In the case of DIII-D, more than half of the scientific staff members are collaborators, with participants from ten major universities. NSTX has about a half of its scientific program carried out by collaborators, worldwide, and about one third from other U.S. institutions. C-Mod is located at a high-profile U.S. research university, where it is the centerpiece of fusion research. All three facilities are heavily involved in supporting doctoral and post-doctoral training. For many scientists at universities, participation in these three world-class facilities provides expanded impact and drive and direction for on-campus activity in both experiment and theory, while at the same time university expertise and insights provide expanded capabilities, depth of expertise, and contact with broader scientific community to the three facilities. Another important aspect of fusion research is that there is essentially no separation between "facility" and "user", as can be the case for other disciplines (e.g. beam lines at a synchrotron light facility). Fusion research requires the simultaneous contributions of facility hardware, operation know-how and diagnostic capabilities, all focused on the common goal of producing magnetic confinement fusion. In other words, the evolving facility **is** part of the research. The U.S. national fusion program is stronger overall for having these three national research facilities, with improved capability to innovate and advance the understanding of frontier fusion science.

What research opportunities would be lost

The three major U.S. facilities—C-Mod, DIII-D, and NSTX—represent a massive investment of talent, intellect, and finances in tackling the key issues of toroidal confinement. Each of these facilities has made seminal contributions to the development of toroidal plasma confinement and the fundamental science and technology that undergird it. Their *combined* resources allow the U.S. to continue with a major role in answering critical scientific questions in magnetic fusion research. The U.S. is well positioned by having these three facilities, whose complementary capabilities enable unique research contributions. The premature loss of any of the three facilities would significantly undermine this U.S. strength.

Some specific examples of the losses that would be suffered by the U.S. fusion program with premature closure for each facility, respectively, are:

C-Mod: This is the only U.S. facility capable of developing lower hybrid current drive for control of the plasma current near the plasma edge. The U.S. would lose a key opportunity to study this important system now being considered for steady-state advanced tokamak operation in ITER, and it is implementing a major effort to do so. The leadership role of the U.S. in the divertor and plasma-boundary area would be seriously harmed. C-Mod features the combination of an all-(heavy)metal wall and reactor-relevant power densities, thus providing critically needed integrated data on material choices for ITER. As the largest fusion experiment at a major U.S. research university, there would also be significant damage to student recruitment and training as well as the loss of a major point of contact for fusion science with the larger scientific community.

DIII-D: The U.S. would lose its centerpiece capability in plasma transport, control of MHD instabilities, divertor physics, and development of the scientific basis for advanced tokamak operation on ITER. In addition, as our primary national facility for maintaining and developing core competencies for the study of burning plasmas, closure of DIII-D would significantly damage the ability of the U.S. to participate effectively in the ITER scientific program.

NSTX: As the newest U.S. facility, aiming at concept innovation and anchoring our leadership in the world spherical torus program, the loss of NSTX would seriously undermine the U.S. program in concept innovation and eliminate a promising candidate for a Component Test Facility, identified as an important element in U.S. Department of Energy fusion energy development long range plan. Activity in this area would then be turned over to Europe through MAST, although this device lacks significant capabilities that NSTX has. Of equal importance, the loss of the parametric breadth provided by

NSTX would weaken many coordinated studies in MHD stability, transport, divertor physics, and energetic particles, as well as significantly narrow the scientific scope of fusion research in the U.S.

Recommendation

In a February 2005 report titled *'The Knowledge Economy: is the United States losing its competitive edge?'*, the Task Force on the Future of American Innovation "developed a set of benchmarks to assess the international standing of the U.S. in science and technology. These benchmarks in education, the science and engineering workforce, scientific knowledge, innovation, investment and high-tech economic output reveal troubling trends across the research and development spectrum. The U.S. still leads the world in research and discovery, but our advantage is rapidly eroding, and our global competitors may soon overtake us."

In the particular area that is the subject of this report, that of the scientifically interesting and broadly important topic of magnetic fusion energy research, the Facility Panel finds that the U.S. at present holds a position of international strength and leadership in many areas. The next major magnetic fusion research facility will be the offshore burning plasma experiment, ITER: the recent decision on its site has fostered growing interest by the international community. The loss of any of the three of the major U.S. toroidal magnetic fusion facilities would fundamentally jeopardize the ability of U.S. researchers to perform relevant fusion research, and thus would undermine the current U.S. position of international excellence.

The Panel's recommendation is that the three major United States toroidal magnetic fusion facilities continue operation to conduct important unique and complementary research in support of fusion energy sciences and ITER.

Table of Contents

Volume 1:

Characteristics and Contributions of the Three Major U.S. Toroidal Magnetic Fusion Facilities

Volume 2:

Requested Information Provided by the Three Major U.S. Troidal Magnetic Fusion Facilities

1. Introduction and Context:
Toroidal Magnetic Fusion Research in the U.S.

Over the past thirty-five years, toroidal magnetic fusion research in the U.S. has developed from a collection of independent institutions and research activities into a cohesive scientific research program that has as its basis a complementary set of collaborative national experimental research facilities. A consequence of this melding of intellectual vigor and innovation from a spectrum of diverse organizations, many with active international partners, is that fusion researchers in the U.S. now agree that the 'knowledge base is now in hand to produce a burning plasma – a plasma whose high temperature is sustained predominantly by energy from alpha particles produced by fusion reactions.'[1] This confidence results from a number of remarkable and important toroidal confinement U.S. achievements in the last decade, including demonstrated control of tokamak major disruptions, the enhancement of ion confinement through sheared flows and transport barriers, the first-ever studies of burning plasma behavior, and the sustained ability to operate high performance plasmas in the MHD-stable advanced tokamak mode, with high self-driven current, good particle and energy confinement, and a plasma edge that enables particle and power handling. The now-sited international experimental toroidal magnetic fusion reactor, ITER, which is the highest-priority new facility of the U.S. Department of Energy (DOE) Office of Science, will benefit significantly from these central U.S. accomplishments. The vitality and vibrancy of the present U.S. program leads to the expectation that the contributions to the world fusion program from U.S. toroidal fusion energy sciences research will continue to be seminal, substantive, and of high impact.

The U.S. magnetic fusion energy sciences research program evolved to its present situation, with three major toroidal facilities, following advice derived from a number of studies undertaken by various DOE federal advisory committees (Magnetic Fusion Advisory Committee, Fusion Energy Advisory Committee, Fusion Energy Sciences Advisory Committee), panels of the National Academy of Sciences, national committees (Toroidal Steering Committee, Ignition Physics Study Group, Transport Task Force), and fusion community input (Coolfont Retreat, Snowmass Summer Studies, Burning Plasma Physics Meetings). The three facilities were deliberately planned for their respective capabilities, and they plan their research programs jointly, e.g., through the Fusion Facilities Coordinating Committee.

Each of the three major toroidal facilities in the current U.S. portfolio developed from a particular thrust pioneered in the U.S. fusion energy sciences program. In chronological order:

- **DIII-D**: The concept that non-circularity might improve the performance of a toroidal plasma (tokamak) was developed in a number of countries, notably, Russia, France, and the U.S. In Europe, this thrust led to JET (U.K.) and later to TCV (Switzerland). In the U.S., non-circular plasmas have been developed mainly in the General Atomics program on D-II, D-III, and DIII-D. Following demonstration of the advantages of non-circularity, the ASDEX (Germany) and JT-60 (Japan) facilities were upgraded to non-circular operation.

- **Alcator C-Mod:** The use of high magnetic field to improve plasma performance and permit compact ignition was proposed at the Massachusetts Institute of Technology—taking advantage of their well-known Magnet Laboratory—and has been pursued through MIT's program on Alcator, Alcator-C, and Alcator C-Mod. This work has provided the underpinnings for all the proposed high-field burning plasma tokamaks (Ignitor, CIT, BPX, and FIRE).
- **National Spherical Torus Experiment (NSTX):** The main advantage of a tokamak with very low aspect ratio is to provide access to operational regimes with extremely high relative plasma pressure (i.e., high β, where the dimensionless quantity β is the ratio of the plasma pressure to the magnetic field pressure). Successful, practical demonstration of this innovative confinement concept—also known as a spherical torus—began in the relatively small-size START device (U.K.) and has been followed up by the present MAST device (U.K.) and the present NSTX device, both of which are proof-of-principle class devices with plasma current on the order of 1 mega-ampere. The spherical torus configuration complements the other facilities by confirming important physics theories in higher aspect ratio tokamaks.

Machine and plasma parameters of these facilities are defined in Appendix 1, Tables A and B respectively. In Appendix 1 Table C, machine and plasma parameters of these U.S. major facilities are compared with those of international facilities: ASDEX-U, FTU, JET, JT-60U, MAST, TCV, and ITER. Two superconducting tokamaks, EAST and K-STAR, will start operation soon. The tables succinctly illustrate that the U.S. toroidal magnetic facilities collectively provide researchers with access to the broad range of parameters necessary for unique and important, forward-directed magnetic fusion research.

1.1. U.S. Position in Toroidal Magnetic Fusion Research

The three diverse and complementary U.S. major toroidal facilities, derived from U.S. innovations, keep the nation at the forefront of magnetic fusion energy sciences research worldwide. To cite a few unique features of these three facilities (in alphabetical order):

- **Alcator C-Mod (C-Mod),** with FY05 annual budget of $22,038,000 U.S., has the highest magnetic field and highest value of field strength to major radius (B/R) among the world's tokamaks.
- **DIII-D,** with FY05 annual budget of $55,751,000 U.S., has demonstrated significant flexibility for the creation of advanced operational scenarios.
- **NSTX,** with FY05 annual budget of $34,584,000 U.S.**,** is the most capable of the world's low-aspect-ratio tokamaks, with sufficient heating power and stabilization systems for accessing very high β and normalized β values.

Two major shared strengths of these facilities are exceptional diagnostics and integration of experiments with theory and numerical simulation. From the 1980's, the strongly coordinated community-wide activity to understand anomalous plasma transport, under the auspices of the U.S. Transport Task Force, has driven the development of new diagnostics, benchmarking with theory and simulations, and targeted experiments. The Office of Science major programmatic initiative, Scientific Discovery through Advanced Computing (SciDAC), has more recently enabled substantial development of U.S. capabilities for numerical simulation of plasma behavior in specific targeted areas. This latter effort has led to the enunciation of a new initiative, the Fusion Simulation Project,[2] whose goal is the fully integrated simulation of a toroidally confined burning plasma and the predictive comparison of simulation results to experiment by means of synthetic diagnostics. It is expected that the Fusion Simulation Project will well utilize the Office of Science's second-highest-priority new facility after ITER, that of leadership-class computing.

On the international front, the three U.S. facilities are major contributors. All three are participants in the International Tokamak Physics Activity (ITPA), which has expanded the coordination of international research. Results from the three U.S. facilities have provided a significant portion of the world's database on fundamental toroidal plasma physics. C-Mod and DIII-D have both had major impacts on the design of the current version of ITER. (NSTX operation started in 1999, subsequent to the ITER redesign effort.) All three facilities are involved in proposing upgrades to ITER operational scenarios. All three also carry out joint experiments with international partners.

From the perspective of fundamental fusion science, the portfolio of U.S. toroidal magnetic fusion facilities—with leverage from the world program—continues to make major contributions to the science of confined plasmas, providing detailed understanding of magnetohydrodynamics (MHD), particle and energy transport, plasma-wall interactions and divertors, heating and current drive, energetic particle physics, and dynamical plasma control.

The three U.S. facilities are also paving the way for effective burning plasma experiments on ITER. In the longer term they are expected to provide the physics basis for a Component Test Facility (especially NSTX), a Compact Ignition Experiment (especially C-Mod), and an improved DEMO reactor (especially DIII-D) and ultimately a commercially viable and environmentally benign Toroidal Fusion Power Plant (all three).

1.2. Importance of the Three U.S. Facilities

The three major U.S. facilities—C-Mod, DIII-D, and NSTX—represent massive national investments of talent, intellect, and financial resources directed to key issues of toroidal plasma confinement. The estimated replacement value of this portfolio is on the order of 1.2 billion dollars U.S. (see Appendix 1 Table B). Each of these facilities has made seminal contributions to the development of toroidal plasma confinement and the fundamental science and technology that underpin it; as an integrated program, they provide an unparalleled test bed for fusion energy sciences research.

One standard for judging progress in fusion research is the "triple product" of plasma density, temperature, and confinement time. This may be rephrased as the product of plasma β, magnetic field squared, and confinement time. A high value of the triple product implies access to a broad range of all relevant (dimensionless) plasma parameters and, thus the ability to investigate fundamental plasma phenomena in regimes that extend from laboratory fusion to wider reaches of astrophysical plasmas. In its rephrased form, a high value of the triple product is achieved within the U.S. toroidal portfolio. Specifically, Alcator C-Mod is the highest field tokamak in the world; DIII-D pursues the direction of high confinement through the development of advanced tokamak operational regimes with long confinement times; and NSTX pursues the direction of ultra-high β, resulting from its extreme toroidicity (i.e., low aspect ratio). Importantly, while these three facilities are distinctly different in some regards, they have a valuable degree of commonality that permits them to compare plasma behavior under similar dimensionless—but different dimensional—parameters (e.g., magnetic field strengths from 0.3 Tesla up to 8 Tesla, all at high performance), comparison which enables the development of fundamental predictive understanding.

The *combined* resources of the present three major U.S facilities provide an outstanding capability for answering critical scientific questions in magnetic fusion research. The U.S. is well positioned by having these three facilities, whose complementary capabilities enable unique research contributions. All three of the U.S. facilities are operationally very productive, with research results that have high impact worldwide. This is evidenced by the large numbers of refereed publications produced by the three facilities (more than one thousand combined in the years following 1999), and by the impressive citation counts for a number of these published journal articles. If only articles that have appeared in print after 1998 and that have citation counts in excess of 30 are considered, the highest citation counts are: for C-Mod, one publications with 70 citations, one publication with more than 50 citations, and two additional with more than 30 citations; for DIII-D, two publications with more than 50 citations, five additional with more than 40 citations, and two further with more than 30 citations; and, for NSTX, two publications with more than 40 citations. (Note that ohmic operations commenced on NSTX in CY2000.)

The loss of any of the three facilities would fundamentally jeopardize the ability of U.S. researchers to perform broadly relevant fusion research, and thus would undermine the current U.S. position of strength and leadership in this scientifically interesting and important international research area, for which the next major facility will be the offshore international experiment, ITER. To reach this conclusion, the Panel considered the contributions[3] from the major international toroidal facilities—JET, JT-60U, ASDEX-U, TCV, TEXTOR, Tore Supra, and MAST. From our understanding of the proposed international activities,[4] the Panel judges that the three U.S. facilities will be making important and unique contributions to the world's magnetic fusion program over the next five years and, furthermore, will jointly carry out important confirmatory experiments. The U.S. will also continue to collaborate with major international facilities through joint experiments.

The U.S. facilities are also very important for training the next generation of plasma physicists and fusion researchers. Their current complements are:

- C-Mod: 30 graduate students;
- DIII-D: 16 graduate students;
- NSTX: 16 graduate students.

This vital training activity is currently limited by funding constraints and insufficient machine operation time. Moreover, these three facilities and their host laboratories have each implemented significant elementary and high school outreach programs.

Extensive information about prior accomplishments and expected future contributions of the U.S. fusion energy sciences program may be found in the FESAC Program Priorities Report.[1]

The charge to the Facilities Panel is reproduced in Appendix 2. The Panel's responses to this charge are contained in the present document (Volume 1). A companion document (Volume 2) contains information submitted by each of the three major U.S. facilities.

1.3. Panel Activities Timeline and Report

The FESAC Facilities Panel (membership provided on page 2 of this report) is comprised of ten researchers representing expertise in all aspects of fusion energy sciences. Table 1.3-1 provides a work timeline of the three months of Panel activity.

Table 1.3-1. Facilities Panel 2005 Work Timeline.

April 5:	Charge Letter is issued to FESAC
April 19:	Panel membership is determined
April 28:	Panel requests information from the three facilities
May 28:	Panel receives requested input from the three facilities
June 2:	Panel receives unsolicited joint facilities document
June 13-17:	Panel holds review meeting [see Appendix 4 for schedule]
July 11:	Panel provides final report to the FESAC
July 19:	Panel briefs FESAC per report contents

The Panel's April 28 request for input from each of the three major facilities included an invitation to each facility program director to provide a document (30 pages maximum length) that addressed the panel charge in detail. Each of the three ‘30-page' documents that were received from the facilities in response to this request is reproduced in Volume 2 of this report, as background to the panel findings and responses. On June 2, the Panel further received an unsolicited joint document from the program directors of the three facilities the purpose of which was to describe for Panel consideration the unique research roles for the U.S. in the world program that are enabled by the combined contributions of the three facilities. This thoughtful and detailed document is reproduced with its cover letter in Appendix 3.

This report addresses the major toroidal magnetic facilities charge, with focus on the four specific charge questions, reproduced in Table 1.3-2.

Table 1.3-2. Facilities Panel Charge Questions (April 5, 2005).

1. What are the unique and complementary characteristics of each of the major U.S. fusion facilities?
2. How do the characteristics of each of the three U.S. fusion facilities make the U.S. toroidal research program unique as a whole in the international program?
3. How well do we cooperate with the international community in coordinating research on our major facilities and how have we exploited the special features of U.S. facilities in contributing to international fusion research, in general, and to the ITER design specifically?
4. How do these three facilities contribute to fusion science and the vitality of the U.S. Fusion program? What research opportunities would be lost by shutting down one of the major facilities?

Section 2, immediately following, provides the Panel’s response to charge Questions #1, #2, and the first part of Question #4. Charge Question #3 is addressed in Sec. 3. In Sec. 4, the second part of Question #4 is discussed primarily from the perspective of the Program Priorities Report fusion energy sciences research objectives.

2. Unique and Complementary Contributions to Scientific Campaigns

Scientific understanding of magnetic fusion plasmas comes from a combination of experimental and theoretical investigation. Crucial experimental verification of theory requires facilities that can vary plasma conditions over an extended range. No single facility can adequately span all desired variations. The U.S. fusion program has a complementary set of three major experimental facilities that are able to access a very wide range of variations. The high magnetic field approach of Alcator C-Mod, the advanced tokamak studies at intermediate field on DIII-D, and the low-field, high-β capability in NSTX complement each other technically, and also for collaborations. Furthermore, the healthy synergy of similarity experiments conducted between two or more facilities contributes to exploiting their full potential to test theory and techniques. With their unique and complementary machine capabilities and outstanding diagnostics, all three major U.S. facilities are carrying out world leading magnetic fusion plasma research.

In this section, we address the charge questions about characteristics and contributions (Questions #1, #2, and the first part of #4) in terms of the five scientific campaign areas that describe the magnetic fusion research program. These important campaign areas were identified in the 2005 FESAC Program Priorities Report as:

> "Macroscopic plasma physics: Understand the role of magnetic structure on plasma confinement and the limits to plasma pressure in sustained magnetic configurations;
>
> Multi-scale transport physics: Understand and control the physical processes that govern the confinement of heat, momentum, and particles in plasmas;
>
> Plasma boundary interfaces: Learn to control the interface between the 100-million-degree-C plasma and its room temperature surroundings;
>
> Waves and energetic particles: Learn to use waves and energetic particles to sustain and control high temperature plasmas;
>
> Fusion engineering science: Understand the fundamental properties of materials, and the engineering science of the harsh fusion environment."[1]

The unique and complementary characteristics and research contributions of the three major U.S. toroidal facilities – C-Mod, DIII-D and NSTX – that are pertinent to these five major topical research categories are overviewed in Secs. 2.1 – 2.5 following. These subsections highlight illustrative examples of ongoing research and capabilities of the three experimental facilities and are not intended to be complete surveys.

2.1. Macroscopic Plasma Physics

Research in the area of macroscopic plasma physics seeks to determine how to confine and sustain maximum plasma pressure efficiently in a magnetic field configuration. This quest is extremely important, since the production of fusion energy in a burning plasma facility (such as ITER) increases with the square of the plasma pressure. The science is complex, because plasma self-organizes into minimum energy states as it nonlinearly interacts with the confining magnetic field, internal electric fields, and plasma flows.

The three macroscopic plasma science topics of equilibrium, stability, and external control are addressed individually by the first three topical scientific questions, T1 – T3, defined in the FESAC Program Priorities Report. The balance of this section shows strengths of the major U.S. magnetic fusion facilities in these areas.

2.1.1. Plasma Equilibrium and Magnetic Field Structure

The plasma pressure and current profiles, rotation, and the confining magnetic field shape can be altered to increase the confined plasma pressure. An appropriate dimensionless quantity used for measurement is called β, which is the ratio of the plasma pressure to the magnetic field pressure.

All three major U.S. facilities are able to match the ITER boundary cross-sectional shape. Alcator C-Mod investigates plasma behavior at high magnetic field strength. It has the highest toroidal field (8. T) and the highest plasma pressure of any tokamak in the world. C-Mod plasmas have intrinsic plasma rotation from torque-free heating with radio frequency ion cyclotron waves. DIII-D has unparalleled ability to study the scientific advantages of plasma shaping. DIII-D can closely match the shape of most tokamaks in the world (Fig. 2.1-1), allowing most important, inter-device similarity experiments. Advanced tokamak studies to optimize β and extend lifetime can be performed on DIII-D through pressure, current, and rotation profile variation, with world-class diagnostics leading, hopefully, to conclusive comparisons to theory. Through boundary and profile shaping, DIII-D has achieved volume-averaged toroidal β of 12.5%, a world record for tokamaks with conventional aspect ratios. DIII-D can vary plasma rotation, through its co- and counter neutral beam injection capability. The low aspect ratio geometry (high plasma toroidicity) of NSTX allows study of the many critical macroscopic phenomena that depend on aspect ratio. Low aspect ratio enables very high shaping factor and elongation, enabling very high β. NSTX is the world's only tokamak to study plasmas in the range of zero to unity (local) β: volume-averaged toroidal β in NSTX has reached 40%, and normalized β has attained a world-record value of 7.2 through boundary shaping and pressure profile broadening. NSTX can produce rapidly rotating plasmas with co-directional neutral beams, allowing research at the uniquely high Alfvén Mach number (exceeding 0.45) reached in the device. Rotational effects on equilibrium are advantageous and particularly strong at low aspect ratio, causing the contours of constant plasma pressure to no longer be coincident with the flux contours of the magnetic field.

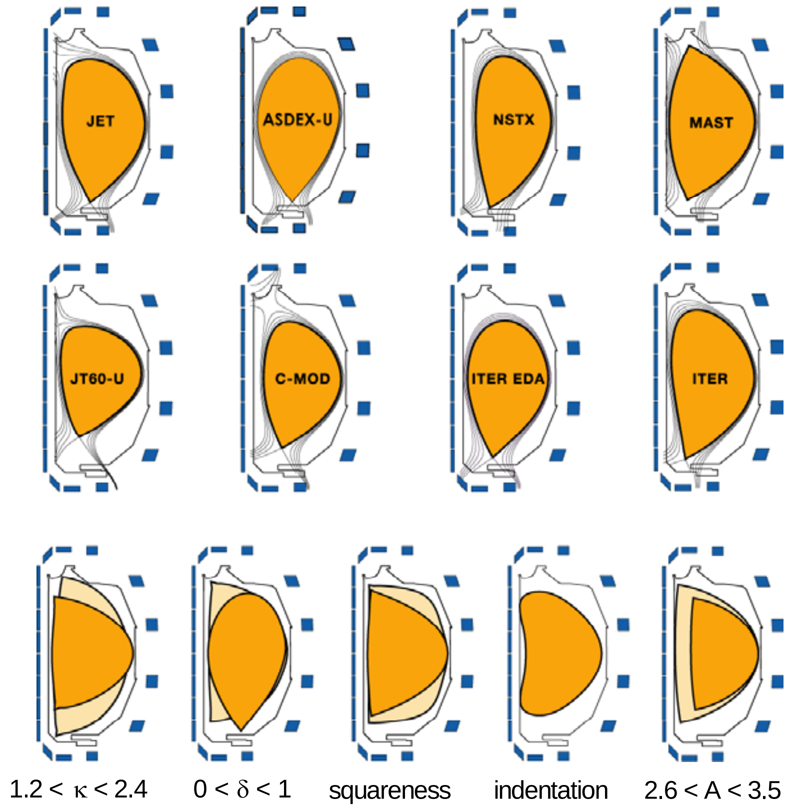

Figure 2.1-1: DIII-D can produce a wide range of plasma shapes and can match the shapes of most machines, including the ITER design.

2.1.2. Pressure-limiting Instabilities

When the β value of magnetically confined plasma exceeds an upper limit, large-scale unstable motions of the plasma can develop, leading to loss of confinement. β-limiting macroscopic plasma instabilities include the kink/ ballooning mode, the resistive wall mode (RWM), and the neoclassical tearing mode (NTM). The kink/ ballooning mode can manifest itself as an internal mode, a global mode, or an edge localized mode (ELM), depending on the area of the plasma that is affected. Understanding the science of the stability limits set by these modes is an essential goal of magnetic fusion research.

High β plasmas can be stabilized by sufficient rotation. DIII-D was the first device to show that it is possible to sustain stable plasma operation above the ideal MHD "no-wall" limit by rotating the plasma inside the electrically conducting vacuum vessel. Both DIII-D and NSTX use a combination of rotation and conducting walls to passively stabilize resistive wall modes and external kink modes. NSTX is the only spherical torus in the world with such capability. At low rotation values where these modes can become unstable, both facilities are instrumented to study their properties (Fig. 2.1-2).

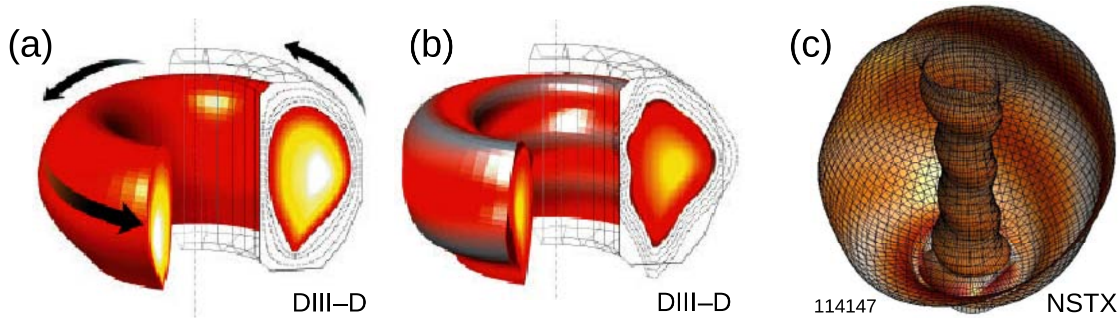

Figure 2.1-2: Resistive wall mode (RWM) in DIII-D and NSTX: (a) DIII-D plasma stabilized by rotation; (b) unstable n = 1 RWM in DIII-D; and (c) RWM with n = 1-3 components in NSTX (surface distortions shown 10x exaggerated).

U.S. facilities have a leadership role in the world program in investigating how amplitude-saturated plasma modes can create drag that slows down plasma rotation, thereby decreasing its stabilizing effect. DIII-D and NSTX are currently performing experiments to examine drag induced by saturated neoclassical tearing modes and edge localized modes, by stable resistive wave modes ("resonant field amplification"), and by the more catastrophic unstable resistive wall mode. Since the drag theoretically depends on the aspect ratio, the safety factor, and the toroidal mode number of the instability, NSTX (with its low aspect ratio, *A*, and high safety factor, *q*) complements experimental studies on DIII-D to validate theory.

Another key research topic is to understand the physics mechanism that sets the critical plasma rotation frequency below which the resistive wall mode goes unstable. Joint experiments between DIII-D and NSTX, abetted by theory and numerical simulations, have recently discovered that the critical rotation frequency normalized to the Alfvén speed scales as $1/(Aq^2)$. Theoretical explanations for the stabilizing mechanism yielding the critical rotation frequency depend either on the plasma sound speed or the Alfvén speed. The separation of the two speeds in NSTX permits their effects to be distinguished.

DIII-D and NSTX are also studying the stability threshold for neoclassical tearing modes, which is an area with important contributions from ASDEX-U as well. These studies take into account the related behavior of large internal "sawtooth" modes with toroidal mode number n = 1. C-Mod plans to study issues of so-called giant sawteeth and sawtooth control with localized ion cyclotron heating; DIII-D is undertaking similar studies, taking advantage of its ability to manipulate the fast ion and current profiles. NSTX is examining the combination of rotation and kinetic stabilization effects on these modes, since these effects are amplified at high β and Alfvén Mach number.

DIII-D experiments, guided by theory, have led the way to establish the kink/ ballooning mode with intermediate toroidal mode numbers as the leading candidate to explain edge localized modes. Its comprehensive diagnostics for the edge pedestal region allow accurate stability predictions, while flexibility in boundary shape and in edge collisionality (varying over two orders of magnitude) permit the analysis of a wide range of edge plasma conditions.

Static, relatively small "error" magnetic fields pose a significant limitation on plasma confinement by aggravating macroscopic instabilities and also by reducing rotation. These error fields can be amplified through resonant response of the plasma. Non-rotating "locked" modes can grow and terminate plasma confinement. C-Mod, DIII-D, and NSTX are all equipped with non-axisymmetric coils for complementary studies (with JET) of the plasma response to error fields. Interestingly, C-Mod results predict a significantly more favorable (five times higher) error field threshold for locked modes in ITER, consistent with results in DIII-D and JET. NSTX contributes studies of the sensitivity of plasma rotation damping and instability threshold to the magnitude of the error field at low aspect ratio.

A high-pressure plasma is potentially susceptible to rapid loss of its stored energy and current if macroscopic instabilities grow unchecked. C-Mod and DIII-D have unique capability to measure toroidal asymmetries of the halo currents that are induced during "disruption" events. Strong asymmetries are critical because they increase design limits on forces that a burning plasma system must withstand and on damage caused by high-power losses onto the plasma-facing components. Another key part in the complementary study of disruption dynamics is control and mitigation. DIII-D pioneered the technique of high-pressure gas injection for mitigation of disruption damage, which is now being implemented jointly on C-Mod for experiments at ITER-like high energy density and pressure values. This technique of massive gas puffs will be implemented on ITER.

2.1.3. External Control and Self-organization

Plasmas that are above instability limits can often be stabilized by external means, with a self-organizing plasma response. Burning plasmas will add the additional influence of strong self-heating due to alpha particles and will provide new opportunities to study highly coupled plasma dynamics and optimized control. All three national facilities are involved in addressing external plasma control and self-organization by studying the proper balance between internal and external control. The objective is to understand the fundamental science that will allow sustained, optimized fusion power production.

The plasma current is most efficiently sustained by non-inductive current that is self-generated by pressure gradients; this is known as the bootstrap current. Large fractions of the total plasma current—up to 85% in DIII-D, and 60% in NSTX (the highest among all spherical tori)—have been created in this way. To supplement the bootstrap current, the U.S. facilities employ non-inductive currents that are externally driven (typically in the outer region of the plasma) by a mix of neutral beams and radio frequency waves. (Current drive capabilities of the three facilities are described in Section 2.4.) DIII-D, which has achieved up to 100% non-inductive operation, along with ASDEX-U, JET, and JT-60, is among the world leading programs in the use of non-inductive current to develop quasi-stationary high-β discharges and high-performance operating scenarios that can be used in present-day and future tokamak fusion facilities, including ITER.

Sustained, self-organized plasma current and pressure profiles must yield a stable plasma equilibrium. With insufficient plasma flow to stabilize instabilities (such as the resistive wall mode), active control coils can be used for this purpose. Calculations using the present ITER design show that ITER would rotate at one-half of the required rotation rate for RWM stabilization. Proposed improvements of the ITER design to include active control coils in the midplane access ports (Fig. 2.1-3) depend on demonstration of the effectiveness of such coils in the next few years. There are existing external and internal coils on DIII-D. NSTX has recently added a set of external coils for error field and RWM feedback control of strongly rotating, high β plasmas. DIII-D and NSTX will operate plasmas above the no-wall limit by means of wall stabilization with rotation. The RWM feedback coils inside the DIII-D vacuum vessel will be used to study fast feedback stabilization of the RWM in plasmas with controlled low rotation, as is expected in ITER. The rotation in DIII-D will be controlled by toroidally balanced neutral beams. Initial results of feedback experiments have been positive. DIII-D and NSTX plan to carry out complementary studies that will complete the validation of active stabilization physics, for extrapolation to the burning plasma regime of ITER.

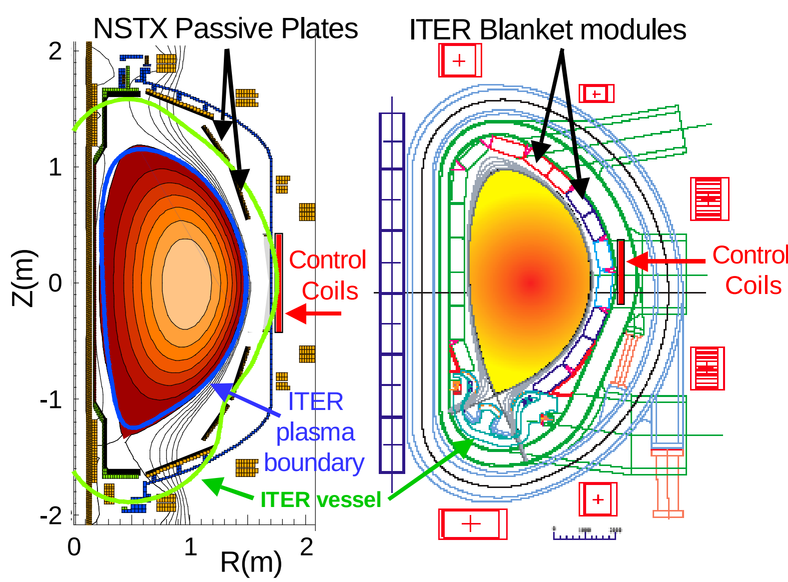

Figure 2.1-3: NSTX passive stabilizers and control coils compared to stabilizing blanket modules and a proposed control coil location in ITER.

DIII-D is a world leader in active mode control systems for both resistive wall modes and neoclassical tearing modes. Its resistive wall mode control system includes both internal and external active control coils, a high-speed digital feedback control processor, and an audio amplifier system for wide bandwidth and reduced system delay (latency). NSTX complements this research through studies of aspect ratio dependency and resistive wall modes with toroidal mode numbers greater than unity.

For active stabilization of neoclassical tearing modes – an area first tried on ASDEX-U – DIII-D uses localized current drive from externally applied electron cyclotron waves. The DIII-D integrated control system uses real-time measurements of the pitch angle of the magnetic field line to track the position of the rational surface where the neoclassical tearing mode resides. With this capability, DIII-D has demonstrated pre-emptive stabilization of neoclassical tearing modes.

The same active coils used for resistive wall mode control in DIII-D and NSTX can also be used for control and mitigation of edge localized modes. Their suppression, critically important for ITER, has already been demonstrated using these coils in DIII-D, for ITER-like edge collisionality, with the application of resonant magnetic fields. Initial results on NSTX are not inconsistent with results from DIII-D.

Due to its high magnetic field, C-Mod generally achieves high performance plasmas while remaining below the macroscopic instability thresholds described above. Within the next five years, the C-Mod advanced tokamak program enabled by the implementation of lower hybrid current drive (see Sec. 2.4) will address issues of significant bootstrap current generation and beta limits in higher pressure plasmas.

2.2. Multi-scale Transport Physics

In plasmas that are relatively free of large-scale instabilities, the transport of energy, momentum, and particles along and across the confining magnetic field lines exhibits complex dynamics and rich phenomenology. Hence a deep understanding of the turbulence processes that govern plasma transport has been described as one of the "grand challenges" of fusion science and, indeed, of physics in general. Small-scale turbulence is driven by plasma free energy or rotation. It is the organization of this small-scale turbulence into larger structures that can cause substantial leakage of plasma out of the "magnetic bottle" at anomalously high rates, faster than what would be expected from simple Coulomb collisions in toroidal geometry (neoclassical transport).

Development of a predictive model for turbulent transport in magnetized plasmas is, therefore, a major goal of the international fusion program, motivated by the need to increase the confidence in extrapolating confinement predictions to larger, more powerful facilities. With advances in fundamental turbulence models, the invention of new turbulence diagnostics, and the economical availability of computing power, development of a predictive transport model is within reach for the first time. Core plasma transport is discussed here, and edge transport is discussed in the boundary section.

The U.S. is the leader of the international research effort to determine the underlying instabilities responsible for plasma transport. The U.S. program focuses on measuring plasma instabilities over the wavelength ranges predicted by theory to be responsible for the cross-field transport, as well as identifying the relevant processes that can damp the turbulence and reduce the transport. The wavelength ranges of the underlying turbulence instabilities are shown in Figure 2.2-1. The longer wavelength range (left) corresponds to instabilities driven by the ion temperature gradient (ITG), whereas the shorter wavelength ranges (center and right) correspond to instabilities driven by the trapped electron population (TEM) and the electron temperature gradient (ETG). Spanning between the ITG and TEM wavelength ranges are micro-tearing instabilities (μtearing), related to plasma resistivity.

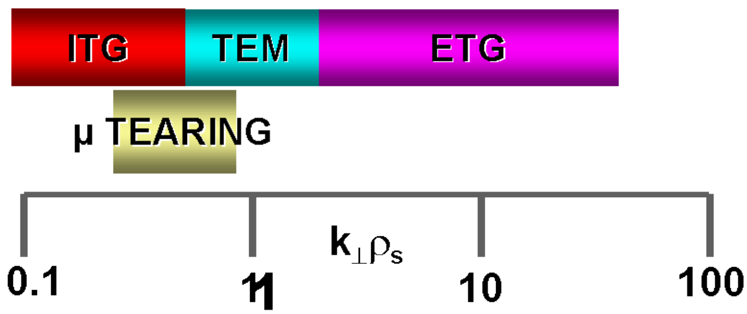

Figure 2.2-1: Microscopic instabilities and their ranges in normalized inverse wavelength ($k_\perp \rho_s$).

Each of the three U.S. facilities contributes important elements to a vibrant domestic program. The U.S. transport program is made strong by DIII-D, which has an integrated program with measurements over the important wavelength ranges, as well as the most comprehensive gyrokinetic transport model (see Fig. 2.2-2).

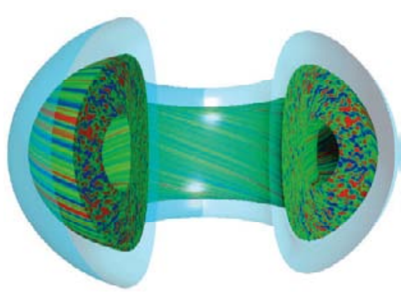

Figure 2.2-2: GYRO code calculation of turbulence in realistic DIII-D geometry and with much of the essential physics.

Although the goals of the NSTX transport program are similar to those of DIII-D, NSTX plasmas offer a different parameter range over which to investigate these instabilities—in particular, a wide range of β (ratio of plasma pressure to magnetic field pressure), as shown in Fig. 2.2-3. High β enables examination of whether electromagnetic turbulence is important, as compared to the baseline electrostatic turbulence predicted to drive transport in lower β plasmas.

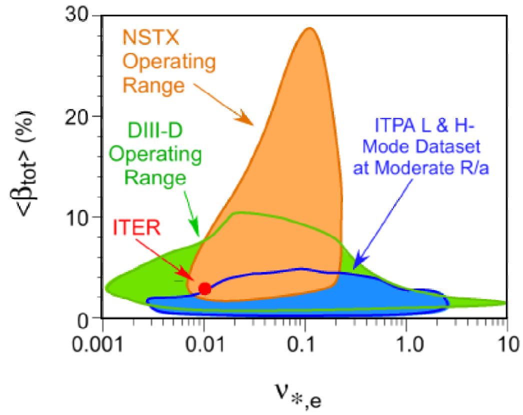

Figure 2.2-3: Extension of operating space provided by NSTX in the dimensionless parameter space of total β (including fast ion pressure) and $\nu_{*,e}$, compared with the international ITPA database at moderate R/a (which includes C-Mod data) and a portion of the DIII-D database.

In C-Mod plasmas, the studies of thermal transport can be extended to different dimensional parameter regimes—e.g., power density (Fig. 2.2-4) and toroidal field B_t—while overlap with other tokamaks is maintained in dimensionless parameter space, e.g. β_N and ρ_*. In addition, C-Mod extends the transport studies to the reactor-relevant regime for uniformly equilibrated temperatures of electrons and ions.

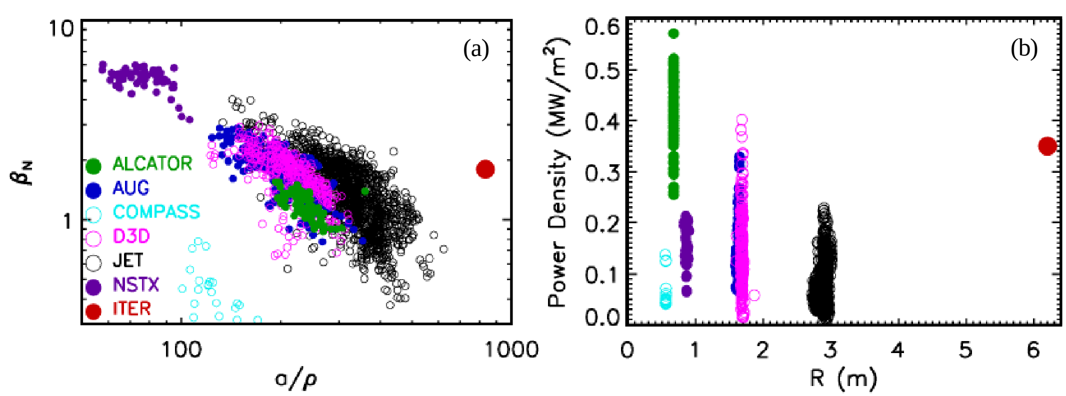

Figure 2.2-4: Extension of operating space provided by C-Mod in the dimensional parameter space of power density (Power/ surface area) and major radius (R) in (b); as overlapping in the dimensionless parameter β_N and $1/\rho^*$ in (a), from the ITPAELMy H-mode database 'DB3v12'.

The combination of the three machines improves the ability to separate out the controlling variables through cross-machine studies of energy confinement; in this way, for example, the importance of aspect ratio in confinement and transport can be studied. This combination of facilities also makes it possible to perform identity experiments (in which all dimensionless parameters are held fixed) and similarity experiments (in which one dimensionless parameter is scanned over a wide range) on two or more domestic (and international) machines.

2.2.1. Thermal Transport (Ions and Electrons)

DIII-D pioneered the research on the role of radial electric field shear (i.e., E×B shear) in suppressing ion turbulence and thus reducing cross-field *ion thermal transport* to neoclassical levels. The transport rate is not reduced everywhere across the plasma, but rather within a distinctive region bounded by a "transport barrier," where the density and temperature sharply fall off. A plasma with a barrier near its edge is termed an "H-mode" plasma, whereas a barrier located farther into the core of the plasma is called an "internal transport barrier" (ITB). The beam emission spectroscopy diagnostic on DIII-D enables local comparisons of the E×B shearing rate and the turbulence de-correlation rate, which provide additional corroboration for the need to have E×B shear in the vicinity of internal and edge transport barriers. NSTX offers a different view of the problem. Because of the high edge magnetic shear in NSTX endemic to low-aspect-ratio geometry, theory predicts that ion-scale instabilities should be less important than in usual-aspect-ratio tokamaks, leading to larger spatial regions of ion thermal transport at the neoclassical level in H-mode plasmas. Investigation of the effect of the ion-channel instabilities in NSTX is facilitated by its charge-exchange recombination diagnostic, whose high resolution can measure the ion temperature profile and flow profiles down to the ion gyro-radius scale length. NSTX also plans to install a new microwave turbulence imaging diagnostic with a large 2-D image for turbulence measurements. An important issue for ion thermal transport that is being investigated in all three facilities is the existence of zonal flows, which can be described as large-scale poloidal rotation along magnetic flux surfaces in narrow radial bands (similar to planetary-scale flows observed on the surface of Jupiter). Zonal flows are theoretically predicted to grow out of turbulence and then to regulate that turbulence and its concomitant transport. Here, C-Mod complements the DIII-D and NSTX data by extending the collisionality range for a better assessment of collisional zonal flow damping of turbulence.

Electron thermal transport is now a frontier for transport studies, in part because of new and improved diagnostics that enable comprehensive measurements. It appears that electron transport exceeds neoclassical values even with electron transport barriers. It is not known for certain if electron transport is dominated by short wavelength turbulence such as electron temperature gradient (ETG) modes, which are on the scale of the electron gyroradius, or longer wavelength turbulence such as trapped electron modes (TEM) and μtearing modes. Domestically, this research direction has been elevated in importance through inter-facility cooperation promoted by the U.S. Transport Task Force and, more recently, by the designation of electron transport as an important research thrust in the report of the FESAC Program Priorities Panel (2005). New backscattering and tangential scattering diagnostics (on DIII-D and NSTX, respectively), developed primarily through university collaborations, are adding to the phase contrast imaging diagnostic (on C-Mod and DIII-D) in assessing fluctuations at intermediate and short wavelengths. The three facilities can access complementary plasma regimes since ion transport can sometimes dominate total heat loss in high-performance C-Mod and DIII-D plasmas, whereas electron heat transport always dominates heat loss in the highest performance NSTX plasmas.

The damping of both ion and electron turbulence can be strongly affected by the ratio of ion temperature to electron temperature: a high ratio is stabilizing for ion transport gradient modes, whereas a low ratio is stabilizing for electron temperature gradient modes. The three facilities offer a flexible range of conditions to quantify the importance of this ratio. Transport studies on C-Mod are conducted in plasmas of high density (afforded by its operation at high magnetic field), which enforces ion and electron temperatures to be everywhere equal across the radial profile, a situation likely to occur in fusion reactors. This strongly coupled ion-electron regime in C-Mod is complemented by the flexibility to vary the value of the temperature ratio by changing the fraction of neutral beam heating electron heating via fast wave in NSTX or the fraction of ion cyclotron and electron cyclotron radio frequency heating on DIII-D.

2.2.2. *Momentum and particle transport*

Momentum transport is another important research frontier. Momentum transport research is needed to help bridge the gap between present-day experiments, which tend to have relatively high rotation rates, and fusion reactors, which are expected to rotate significantly more slowly. Recent C-Mod results demonstrate a need for basic understanding: C-Mod ohmic and radio frequency wave-heated plasmas rotate rapidly, which is surprising because radio frequency wave heating (in contrast to neutral beam heating) does not impose external torque. These findings have stimulated theoretical speculation and new experiments that suggest that the plasma edge region plays a crucial role in core plasma flow. A new rotation diagnostic (active charge exchange) will add to the existing core rotation measurements by determining the importance of the edge plasma in setting the boundary conditions for rotation of the core plasma, and also in affecting the power level required to force the formation of an edge H-mode transport barrier. For such studies, DIII-D will soon have important flexibility enabled by the rotation of one neutral beam line. The reconfigured beam line will allow up to 10 MW of heating, while imparting zero net momentum to the plasma. An additional 10 MW of unidirectional neutral beam heating and the ability to reverse the direction of the plasma current will give DIII-D the ability to vary the direction of the torque and the ratio of heating power to torque magnitude. NSTX can complement these momentum transport studies by comparing forward and reversed plasma current in order to investigate the origin of momentum transport with rates approaching the neoclassical level.

Finally, a main goal of *particle transport* research is to predict the residence time of fuel and impurities, as well as the build-up of helium ash in a burning plasma. Historically, particle transport studies in tokamaks have been conducted with the use of spectroscopy to track helium or other injected impurities. Those studies yielded effective diffusion coefficients and inward convective velocities, for comparison with neoclassical predictions and scaling laws. Substantial progress was made in the 1990's on important issues such as transport and exhaust of helium and impurities, impurity screening, and density limits. Consequently, this area of research presently receives less emphasis than the other transport topics. New techniques to infer the turbulent radial particle flux directly from beam emission spectroscopy data are being pursued in DIII-D, while NSTX is focusing on impurity transport studies, which have indicated diffusion rates comparable to neoclassical values.

2.3. Plasma-Boundary Interfaces

The science of the transitional plasma that connects the hot (100,000,000 degrees C) thermonuclear "core" where fusion occurs with the "terrestrial" material walls (20-1000 degrees C) is the subject of plasma-boundary research. Confined and stable core plasmas must still leak particles and power through this boundary layer. This layer is only a few centimeters wide, and yet contains the largest gradients in plasma pressure and magnetic field topology, making it a critical region for stability and confinement. The prevailing magnetic topology, and the design chosen for ITER, involves a "divertor" that intentionally diverts boundary field lines to a dedicated target surface in order to provide heat and particle exhaust. These field lines are termed "open" because they strike material surfaces before having the chance to close upon themselves. The dividing surface between the open field lines and "closed" field lines in the core is called the magnetic separatrix and the region outside of the separatrix is called the scrape-off layer (Fig.2.3-1). The characteristics of the scrape-off layer plasma are determined by the balance between transport along the open field lines to the divertor target and transport across the open field lines to the wall.

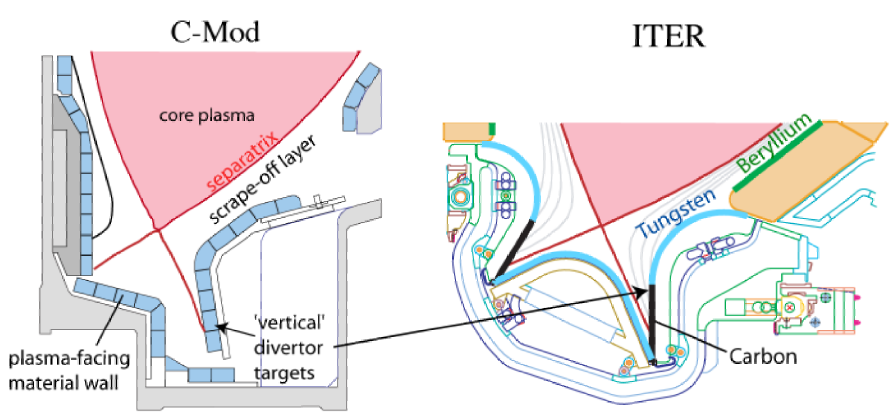

Figure 2.3-1: Cross-sections of C-Mod (x10 scale) and ITER divertor. Boundary plasma terms are labeled in the left figure, and the choice and location of ITER wall materials is indicated at the right.

Plasmas with edge transport barriers, i.e., H-mode plasmas, exhibit steep density and temperature radial gradients within centimeters of the separatrix. The end of the steep gradients as one moves further into the core plasma is called the H-mode pedestal. In this plasma-boundary interface section, we discuss plasma transport from the pedestal into the scrape-off layer, the divertor and the first wall, including the topics of edge plasma optimization and wall material selection.

Plasma transport to wall surfaces, both along and across magnetic field lines, represents a complex interaction between multiple states of matter (gas, liquid, solid) and the highly turbulent plasma. Severe and simultaneous demands are placed on the plasma-facing component (PFC) materials for heat removal, plasma cleanliness, and compatibility with fusion hydrogen fuels. Small changes in the highly non-linear equilibrium of the boundary plasma can dramatically affect the overall performance of a burning plasma. Hence, the study of plasma-boundary interactions aims to answer the following scientific question: *How can a 100-million-degree burning plasma be interfaced to its room temperature surroundings?*

Understanding the physics of the boundary region with sufficient clarity to reliably predict the behavior of a burning plasma experiment presents one of the greatest challenges to the fusion community.

2.3.1. Boundary Characteristics of U.S. Devices

Prediction of boundary plasma behavior is more complex than for the core plasma. The added complexities of the boundary plasma, such as atomic physics and gas-plasma interaction, require extra dimensionless parameters that cannot be matched simultaneously. Hence, there is a strong necessity to have access to a wide range of boundary plasma parameters in present fusion facilities for extrapolation to future devices. Simultaneously, significant cross-device comparisons occur between facilities due to both their common use of divertor topology (as in ITER) and their strong links to a comprehensive suite of U.S. edge models. The high-field C-Mod features local plasma parameters of power and particle density nearly identical to those in the ITER divertor. This allows for unique tests of radiation trapping and neutral/ gas physics critical to ITER divertor operations. DIII-D features upper and lower divertors that operate at lower density due to efficient in-vessel pumping. This allows for unique control of plasma density in highly shaped plasmas in an integrated manner with advanced tokamak research. NSTX features the unique aspects of low-aspect-ratio edge plasmas (including high expansion ratios of the poloidal flux to broaden the heat footprint), short magnetic field connection lengths and high heat flux to the divertor target. In combination the three U.S. facilities have a strong focus on divertor and edge plasma diagnostics, which provides a detailed set of measurements for model benchmarking. Taken together, the U.S. facilities and computational modeling effort thus provide an effective capability in edge and divertor science.

The U.S. facilities have made seminal contributions to the current ITER boundary design. An operational regime known as detachment was identified which greatly reduces divertor plasma temperature with the goal of controlling the localized heat flux to the divertor targets. ITER adopted the radiative divertor concept from C-Mod and DIII-D that facilitates detachment. DIII-D's divertor Thomson scattering diagnostic to measure the electron temperature verified the ultra-low temperature plasma. C-Mod showed that the plasma recombines into neutral states before it reaches the wall, in a sense demonstrating "detaching" the plasma by suspending it off of the material surface. Divertor detachment is the standard ITER operation scenario. A continuing unique U.S. contribution in detachment physics is carried out on C-Mod on the effect of line radiation trapping, which can make detachment more difficult to be obtained.

2.3.2. Plasma-facing Component Materials

The design choice of the ITER wall material is strongly interlinked with all the other boundary research topics, and remains an issue that requires timely contributions in the next few years for ITER construction. The existing ITER design uses a mixture of plasma-facing component materials: beryllium metal main chamber walls, tungsten metal near the divertor entrance, and carbon divertor plates (see Figure 2.3-1). These choices are based on the expected compatibility of each material in different locations: beryllium to ensure only low-atomic-number (low-Z) impurity in the core, high-Z tungsten for its low sputtering yield, and carbon (which sublimates rather than melts) for its resilience to transient heating. Present boundary plasma research is identifying questions about the validity of these trade-offs. In particular, carbon is favored for heating resilience, but tritium retention and recovery from carbon could impact its operational availability. Significant research is required in the next five years to ensure an optimized design of the complex and expensive water-cooled ITER wall components.

Due to their present diversity of wall materials, the U.S. fusion facilities are well positioned to make key contributions to ITER. C-Mod is the only divertor tokamak in the world with an all-metal, high-Z plasma-facing component wall. With significant operational experience in this area, C-Mod is playing a central role internationally in assessing the effects of its molybdenum/ tungsten walls on high-Z impurity radiation in the core, on surface melting, and on mitigation of tritium retention without carbon. DIII-D and NSTX presently have all-carbon plasma-facing component walls, allowing them to study carbon tritium retention and recovery issues without the complicating aspects of metals. This portfolio is well integrated with other facilities: JT-60U and JET have carbon walls, with JET installing beryllium/ tungsten walls in four years, and ASDEX-U is gradually switching from carbon to tungsten-coated graphite.

2.3.3. Pedestal Physics

A high level of energy confinement is required for ITER to reach its goal of fusion gain Q=10. A U.S. physics insight had identified the controlling role of the plasma parameters at the top of the H-mode pedestal. The pedestal temperature sets the global energy confinement due to critical temperature-gradient transport in the core; however there presently exists no first-principles model to predict the pedestal parameters for ITER. The U.S. is leading an effort to understand pedestal physics with comprehensive and complementary diagnostics and theoretical models applied across the three facilities. C-Mod, DIII-D, and NSTX all have spatially detailed measurements of the electron density and temperature and ionization in the pedestal, along with measurements of fluctuations that can drive transport. DIII-D has a unique diagnostic for the edge plasma current. Measurements from all facilities are currently being compared with common stability models (e.g., ELITE) and edge turbulence models (e.g., BOUT). Importantly, the three facilities span a large range of edge parameters: C-Mod operates with high density and neutral opacity and is adding cryopumping for edge density control for the FY06 campaign; DIII-D can lower the edge density with cryopumps; and NSTX has very large magnetic shear. The U.S. goal in the next five years is to provide the first predictive model of the edge pedestal in a tokamak. These activities are focused and coordinated through the Transport Task Force in the U.S. and the International Tokamak Physics Activity (ITPA) internationally. Pedestal physics is a high-leverage burning plasma science issue, and the U.S. is well positioned to be a leader in this area during the next five years.

2.3.4. Edge Localized Modes

As a consequence of the need for steep pressure gradients at the pedestal, most high confinement regimes are characterized by edge localized modes, which are MHD instabilities that cause repetitive, sudden pulses of heat and particles through the edge plasma to material surfaces. Baseline ITER operation is an H-mode plasma with edge localized modes, the size of which are presently predicted to transiently heat divertor materials past their melting and ablation limits, risking unacceptable lifetime for the material targets. The U.S. has a concerted cross-facility U.S. effort to: (1) identify the modification to the edge stability that maintains good confinement and high edge gradients without large edge localized modes; and, (2) identify the processes responsible for residual steady-state particle and heat transport. DIII-D and NSTX have high-resolution measurements of edge localized modes, which are being compared in detail with results from stability codes. C-Mod routinely accesses a high-collisionality small ELM regime. DIII-D has identified a low-collisionality ELM-free regime and uses an internal coil set for suppression of edge localized modes. NSTX has identified a different small edge localized mode regime that is compatible with very high β. Advances in understanding and controlling edge localized modes is a high-leverage issue for the U.S. relevant to the success of ITER.

2.3.5. Tritium Retention and Plasma-facing Components

Tritium retention is a critical issue for ITER construction, nuclear licensing and operations. The radioactive tritium fuel used in ITER (or any burning plasma) can be trapped in films caused by erosion of and deposition on the walls by the boundary plasma. Extrapolation and modeling suggest the ITER tritium in-vessel limit, mainly in carbon film deposits, may be reached in only one to five days of operation, necessitating tritium removal intervention that may compromise the scientific productivity of ITER. However this prediction is uncertain since it is based on data from all-carbon tokamaks, and modeling is hampered by a lack of fundamental understanding for plasma particle transport. To address tritium retention the U.S. facilities are examining different approaches that are complementary to each other, and to efforts in other tokamaks in the world. C-Mod has all-metal walls (molybdenum/ tungsten) without any carbon surfaces. C-Mod has shown that molybdenum walls have significantly less tritium retention (by factors of ten) than tokamaks with carbon plasma-facing components. Largely based on these C-Mod results, other facilities (ASDEX-U and JET) are investigating whether to switch from carbon as a plasma-facing component in order to solve the tritium retention problem. However molybdenum and tungsten have other issues, namely melting during edge localized modes and disruptions, and strong radiation in the core plasma, which affects core plasma performance. C-Mod has also observed that frequent application of low-Z films on top of the high-Z metal may be necessary to achieve good plasma performance. C-Mod, with its ITER-relevant edge power density, should continue to make vital contributions on these topics in the next five years. DIII-D has all-carbon walls and is therefore an ideal facility to pursue a strategy of understanding tritium retention in order to maximize recovery of tritium from carbon films. DIII-D is exploiting the unusually high toroidal symmetry of its wall surface to carefully diagnose and account for carbon-13 isotope tracers injected into the plasma, with the goal of understanding the underlying carbon sources and transport that sets tritium retention. This is complementary to similar studies on JET and ASDEX-U. DIII-D will further explore the efficiency of oxygen baking to recover tritium by laboratory studies and possibly in-vessel baking. Since oxygen baking is presently the only tritium recovery technique planned for ITER, controlled tests of its efficiency are critically required in the next five years. NSTX has carbon walls and has presently installed a unique in-situ diagnostic to measure the presence of dust containing carbon. Tritium-laden dust is a safety concern in ITER. NSTX is also proposing to test a liquid lithium divertor, which would provide both particle pumping and heat flux control. Because it is liquid, lithium can form a “renewable” divertor, and the tritium can be removed by lithium recycling. However, liquid lithium may also experience MHD forces; NSTX will test the stability of the liquid metal against such forces, an activity that complements melt-layer studies of high-Z metals on C-Mod.

2.3.6. *Boundary Transport*

Reliable predictions of the boundary plasma and plasma-facing component behavior require an understanding of the controlling mechanisms for the complex turbulent transport of plasma to the surfaces. The U.S. facilities have recently carried out several experimental and theoretical research efforts on this subject. All three facilities have documented the existence, first reported on C-Mod, of intermittent plasma ejections by convective transport across magnetic fields. The three facilities feature common diagnostic techniques, such as rapid imaging and probes, across a large span of edge plasma parameters, thus providing critical tests for the strong U.S. modeling effort on turbulent transport. The presence of near-sonic plasma flow along field lines is another topic of interest that is commonly studied across the three facilities. This edge flow appears to be one of the explanations for the prevalent retention of deposited films with tritium at the inboard divertor, but it has eluded theoretical explanation to-date. C-Mod research has recently suggested that the flow is caused by ballooning transport at the outboard side of the plasma, and NSTX is planning to image the 2-D flow patterns at different poloidal locations. Toroidally symmetric injection of carbon-13 into DIII-D has confirmed the link between the inboard-directed flow and carbon deposits at the inner divertor. Taken together, the three U.S. facilities will play a leading role in developing a better understanding of edge plasma transport, which will lead to better predictability for ITER.

2.4. Waves and Energetic Particles

There is a large variety of electromagnetic waves that propagate in magnetized plasmas, and their behavior is important for achieving high-temperature plasmas. Waves, particularly in the radio frequency range, that are applied to a plasma from the outside can accelerate plasma ions and electrons through resonant interactions, thereby heating the plasma and/ or driving non-inductive current. The applications of such waves are becoming increasing important and precise tools for stable high performance plasmas, and are essential for achieving steady-state, advanced tokamak scenarios. Externally launched waves are also useful for diagnosing plasma properties. Waves that arise spontaneously inside plasmas can develop into eigenmodes that may cause large-scale unstable disturbances in plasma motion. A “sea” of turbulently fluctuating waves may enhance plasma transport, reducing confinement. Thus, understanding and manipulating wave-particle interactions are important thrusts in the U.S. and international fusion research programs and will remain so during the next five years.

In addition to thermal ions and electrons, plasmas often contain a special population of highly energetic (supra-thermal) particles, which can strongly influence plasma behavior. The energetic particles may be fast ions injected via neutral beams or generated via radio frequency applied waves. Runaway electrons are another example. In burning plasmas there arises an important class of energetic particles, viz., alpha particles (helium-4 ions) born from thermonuclear reactions. These alpha particles transfer their energy mostly to the thermal electrons, which then through collisions heat the plasma ions that are the fusion “fuel.” Since this process represents the main heating mechanism in a self-sustained fusion plasma, maintaining good confinement of the alpha particles is vital. However, non-Maxwellian energetic ions and alpha particles can excite unstable plasma waves that may degrade their confinement; also, fast particles can affect the macro-stability of the bulk plasma. Hence, understanding the behavior of fast particles will be critical for successful operation of ITER and also is necessary for fusion research today, particularly as studies of plasma confinement push into new regimes in which the plasma pressure is high, the magnetic field shear is reversed, and rotation is strong.

The three major U.S. facilities contribute significant and distinct efforts toward advancing our knowledge about energetic particle behavior and application of wave heating and current drive in fusion-grade plasmas.

2.4.1. Wave Heating and Current Drive

The specific techniques for radio frequency wave heating and non-inductive current drive are different in each of the facilities, due to the different properties of their plasmas and to the different tasks for the waves (Table 2.4-1). Some of these techniques have been investigated since the early years of controlled fusion research and are now reliable and flexible, while others are relatively new. The wide range of frequencies—from 30 MHz (characteristic of ion cyclotron motion) up to 120 GHz (characteristic of electron cyclotron motion)—means that quite different types of antennas must be used to couple high-power radio waves into the plasma from its edge.

Table 2.4-1. Wave heating and current drive techniques on the three facilities.

	C-Mod	**DIII-D**	**NSTX**	**ITER**
Heating	Ion cyclotron resonant heating (minority ion & mode conversion) Lower hybrid	Neutral beam injection Fast wave heating Electron cyclotron heating	Neutral beam injection Fast wave	Neutral beam injection (33 MW) Ion cyclotron resonant heating (20 MW) Electron cyclotron heating
Current Drive	Lower hybrid Fast wave current drive Mode conversion current drive	Neutral beam injection Electron cyclotron current drive Fast wave current drive	Neutral beam injection Electron Bernstein waves Coaxial helicity injection Fast wave current drive	Electron cyclotron current drive (20 MW) Lower hybrid (option–20 MW) Neutral beam injection (33 MW) Negative ion neutral beam current drive

DIII-D is the only U.S. facility to use electron cyclotron wave heating, as a supplement to neutral beam injection, its primary heating method. It also uses electron cyclotron waves to drive non-inductive current for two purposes: (1) broadening the radial profile of the current, in order to access high-gain, steady-state advanced tokamak operating modes; and (2) suppressing resistive instabilities known as neoclassical tearing modes at targeted radial locations. The theory of electron cyclotron current drive has been extensively validated by experiments. These instabilities are known to limit plasma β, and hence fusion performance, if they grow to large amplitude, and may even cause major disruptions. DIII-D is a leader in the worldwide effort to control neoclassical tearing modes. Following initial international work, experiments on DIII-D showed that electron cyclotron current drive can successfully suppress neoclassical tearing modes. Over the next five years DIII-D plans to double the power of its electron cyclotron waves, up to 6 MW, making it the most powerful such system in the world. The efficiency of neoclassical tearing mode suppression will be improved through power modulation techniques and full integration of the electron cyclotron current drive into the DIII-D advanced plasma control system; this will permit the method to be validated for use on ITER, which will have 20 MW of electron cyclotron current drive and heating. This is an area in which ASDEX-U and TCV are also expected to make important contributions.

DIII-D also employs fast waves in the ion cyclotron frequency range to drive non-inductive, steady-state current in the core of the plasma (in contrast to electron cyclotron wave-driven current, which is mainly off-axis). It is believed that extended-discharge scenarios will benefit from control of the central current profile, and DIII-D will explore this as an element of their integrated advanced tokamak research thrust.

Spherical tori, due to their innovative configuration, have limited capability to produce purely inductively driven long-pulse discharges, and therefore NSTX is exploring non-inductive current drive for current sustainment and current profile control (the latter to achieve optimally stable current and pressure profiles). However, the high density and relatively low magnetic field in NSTX prevents electromagnetic waves at the electron cyclotron frequency from penetrating the plasma, rendering current drive by electron cyclotron waves unfeasible. Within the next five years, NSTX will initiate current drive via electron Bernstein waves: possibly the only method suitable for off-axis current driven in over-dense plasmas. Electron Bernstein wave heating and current drive were demonstrated on the W7-AS stellarator and the COMPASS-D tokamak. Presently, NSTX is collaborating with the MAST facility (U.K.) to optimize plans for its own studies. Because of their higher magnetic fields, neither of the other two major U.S. facilities proposes—or needs—to implement this scheme. Also, NSTX is unique in investigating coaxial helicity injection, a non-inductive method (albeit not based on radio frequency waves) to initiate plasma current in spherical tori. Successful non-inductive localized current drive on NSTX will supplement the bootstrap current to complete the current drive requirement for steady-state advanced tokamak operation at high values for the plasma β. Finally, like DIII-D, NSTX presently uses neutral beam heating and current drive and is currently exploring the use of fast wave current drive for central heating and non-inductive current generation in the plasma core.

Alcator C-Mod is unique in that the entirety of its auxiliary heating and current drive is obtained from the application of radio frequency waves. Unlike DIII-D, NSTX, and major international facilities, C-Mod does not use neutral beam heating. Instead, it exclusively employs ion cyclotron resonant heating and has succeeded in making this method a reliable heating tool, with a heating efficiency comparable to that of neutral beams. This is a notable achievement that follows from decades of multi-institutional work to understand the coupling, propagation, and absorption of these waves and their antenna technology. Active research on this subject during the next five years will continue to be a priority in the U.S. program because ion cyclotron resonant heating is more flexible than neutral beams in providing localized heating (and current drive); this is particularly important for large fusion-grade plasmas in which neutral beams cannot penetrate to the core. Moreover, the ion cyclotron heating deposition profile is independent of density, and can be tailored by the use of multiple sources with different frequencies to make it extremely useful in confinement studies and optimizing the generation of transport barriers. C-Mod will also use ion cyclotron waves to explore novel methods of local current drive and flow drive (i.e., momentum transfer to the plasma), involving the mode conversion of the incident fast waves into ion Bernstein and ion cyclotron branches. Since ion cyclotron resonant heating will be the main radio frequency heating method on ITER, these studies on C-Mod are crucial to U.S. participation in the development of key elements for burning plasma experiments.

C-Mod has also recently begun non-inductive current drive experiments with lower-hybrid waves that are intermediate in frequency between ion cyclotron and electron cyclotron frequencies. Lower-hybrid waves have the highest efficiency of all available (and proposed) methods of wave-driven current drive, and they are particularly suited to produce the current profiles needed for long-pulse, moderate β_N advanced tokamak scenarios that need significant non-central current. On ITER, lower hybrid is being held in reserve as the primary method for auxiliary current drive in future advanced tokamak studies. Among the three major U.S. facilities, lower hybrid current drive is practical only for the high-density, moderate- β plasmas achievable in C-Mod. Its 3 MW lower-hybrid system will be key to this facility's contributions over the next five years to the knowledge base for high-power-density lower-hybrid current drive at ITER-relevant densities, made accessible by the uniquely high magnetic field capability of C-Mod.

A particular strength of U.S. fusion research that compares favorably with the international program is the capability to compare detailed experimental measurements of wave propagation and absorption with comprehensive full-wave modeling of radio frequency heating and current drive scenarios. In C-Mod studies, excellent agreement of density fluctuations associated with ion cyclotron heating, measured with the novel phase contrast imaging diagnostic, is obtained with numerical modeling from a state-of-the-art SciDAC code simulation of the launched wave spectrum. Similarly advanced modeling and experimental validation capability will available for the lower hybrid current drive studies on C-Mod.

As a whole, the diversity of research on wave heating and current drive being carried out on these three facilities solidifies U.S. participation in the international burning plasma program and is also essential for the development of innovative advanced tokamak confinement scenarios, in which effort the U.S. fusion science program is a world leader.

2.4.2. Energetic Particles

All three major U.S. facilities pursue fast particle investigations with different emphases, related to their characteristic plasma properties and machine capabilities. To investigate the behavior of Alfvén-type eigenmodes destabilized by energetic particles, a potential threat to confinement in burning plasmas, the *fast ion velocity* should be comparable to the plasma Alfvén velocity. NSTX, due to its low magnetic field, can access values for the fast ion velocity that are several times above Alfvénic, allowing excitation of high-frequency Alfvén instabilities. The *fast ion* β (normalized pressure) measures the strength of the excitation by fast particles; DIII-D, with its multiple neutral beam ion sources, achieves high values for this parameter. The *thermal plasma* β value is also relevant to fast particle modes, since it skews the spectral gap where the modes occur and also affects their damping rate. With NSTX (high β), DIII-D (moderate β), and C-Mod (low β at high performance), a large variation in plasma β is available to the U.S. program for such studies. As shown in Figure 2.4-1, the U.S. facilities have fast ion to Alfvén velocity ratios (V_{fast}/V_A) and fast ion to plasma β ratios ($\beta_{fast}/\beta_{plasma}$) relevant to the projected operating space for ITER.

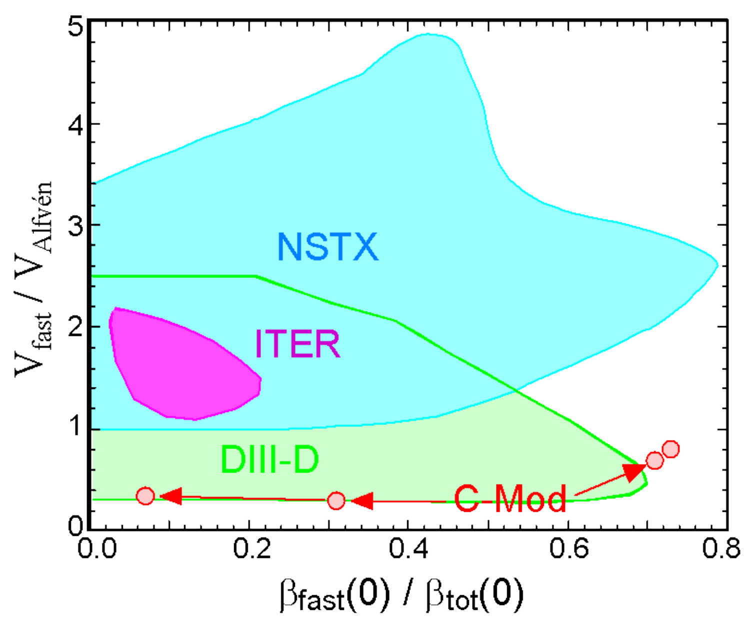

Figure 2.4-1: Operating regimes in fast particle normalized velocity and pressure for the three U.S. facilities and ITER.

With regard to other key energetic particle parameters, the experimental conditions in the three facilities are different from those expected in ITER. A burning plasma like ITER will contain energetic alpha particles, whose velocity distribution is isotropic, whereas fast ions created by neutral beam injection (DIII-D and NSTX) or by ion cyclotron wave heating (C-Mod) are anisotropic, with fast ion energy primarily in the direction of the toroidal magnetic field for neutral beams or in the cross-field direction for wave heating.

The *degree of anisotropy* can affect the strength of the Alfvén eigenmode instability; fortunately, for equivalent energies, the prediction for isotropic alpha particles in ITER will probably be more optimistic than the result for either limit of anisotropy. Another key parameter is the ratio of the *fast ion gyro-radius* to the plasma minor radius ($\rho*_{fast} = \rho_{fast}/a$); this determines which are the most unstable mode numbers. C-Mod performs studies at $\rho*_{fast}$ values that are smaller than in NSTX and DIII-D and closer to the values expected in ITER.

In other respects, the three U.S. facilities go beyond the standard ITER operating space and explore features of energetic particle-excited instabilities in broader regimes of operation. The regime of ultra-high β is accessible in NSTX, as is also the issue of strong poloidal coupling among Alfvén eigenmodes, thanks to NSTX's low aspect ratio configuration. All three facilities can access the advanced tokamak regime of flat and reversed magnetic field shear; in particular, they plan to study how the known nonlinear properties of Alfvén cascade waves can serve as an indirect diagnostic of the internal magnetic field. Also, the stability of fast ion-excited instabilities can be examined in regimes of rapid plasma rotation (NSTX and DIII-D) and low rotation (C-Mod, DIII-D and NSTX), to see if fast ion losses are thereby minimized.

A characteristic of all three U.S. facilities is that they have very good diagnostics for energetic particle physics studies. C-Mod is unique within the U.S. program for having active external antennae, by means of which the damping of Alfvén eigenmodes can be measured in the absence of fast ions; these studies are collaborative with JET, where this method originated. C-Mod is using its new phase contrast imaging diagnostic (which it has also installed on DIII-D) to observe the modes deep inside the plasma. DIII-D is the only machine using beam emission spectroscopy to see the internal modes. DIII-D is developing a new diagnostic, based on D_α emission, to measure the distribution of fast ions. NSTX has an especially good set of fast ion diagnostics, including a scanning particle analyzer, lost ion probes, and plans a neutron collimator. All three machines either now or will soon have the powerful motional Stark effect diagnostic to measure the internal magnetic field profile, which is sensitive to localized beam current drive and on which the Alfvén eigenmodes have a sensitive dependence. The NSTX is unique in being able to measure q(r,t) at B fields where $V_{beam} > V_{Alfven}$.

In the world arena, the three U.S. facilities are able to make significant contributions to energetic physics research. Since the shutdown of the U.S. tokamak TFTR, experimental leadership in this area has belonged to the world's largest tokamaks, JET (in Europe) and JT-60U (in Japan). JET produces fast ions with either neutral beams or ion cyclotron waves; it has also carried out deuterium-tritium campaigns with alpha particles (following the initial such investigations on TFTR); it also injects helium ions to mimic alpha particles. JT-60U likewise creates fast ions with both beams and waves; in particular, it has implemented the technology of neutral beam injection for negatively charged ions, with much increased beam ion energy. To diagnose the fast ions, both JET and JT-60U use neutron measurements, and JET also has two-dimensional gamma-ray tomography. Healthy collaborative links have been established by the U.S. facilities with these energetic particle efforts abroad, with U.S. scientists installing new diagnostics and running experiments.

The three U.S. facilities provide key, distinct capabilities for the international effort in this research area. Flexibility in manipulating plasma shape and profiles is a major strength. DIII-D, for example, can be configured as a quarter-size ITER. Shape flexibility permits cross-machine similarity experiments; a recent example was the confirmation on DIII-D and NSTX of the predicted scaling with fast ion gyro-radius for toroidal Alfvén eigenmodes driven by beam ions (as demonstrated on NSTX). Flexibility also extends to controlling the fast particles. With its co- and counter-directional neutral beam injection ability and also fast wave coupling to beam ions, DIII-D can change the isotropy, energy, and pressure of the fast particle population, enabling the facility to access a wide variety of energetic ion phenomena. Another strength is that energetic particle experiments on the three major facilities benefit from close interaction with theory and numerical simulations—a traditional characteristic of the entire U.S. fusion science program.

Since the behavior of burning plasmas will be dominated by a large population of fusion-product energetic alpha particles, as well as beam- and wave-generated fast ions, understanding their effects is vitally important for the successful operation of ITER. The three U.S. facilities are poised to make useful contributions to this knowledge base.

2.5. Fusion Engineering Science

The FESAC Program Priorities Report[1] states that the objective of fusion engineering science is to "Understand the fundamental properties of materials and the engineering science in the harsh fusion environment." Much of the research activity in this area is outside the scope of the experimental programs of the three major U.S. toroidal facilities; hence this section will be brief. (A related discussion can be found in Sec.3).

The aspects of fusion engineering science that are relevant to the three facilities are primarily concerned with providing the knowledge base in plasma technologies for heating, current drive, fueling, and plasma-facing materials (particle and impurity control). In addition, there is the goal of supporting the construction and safe operation of ITER, including disruption mitigation. This research, overall, is highly collaborative and productive; see Fig. 2.5-1 for examples of fusion engineering science capabilities developed by collaborating institutions.

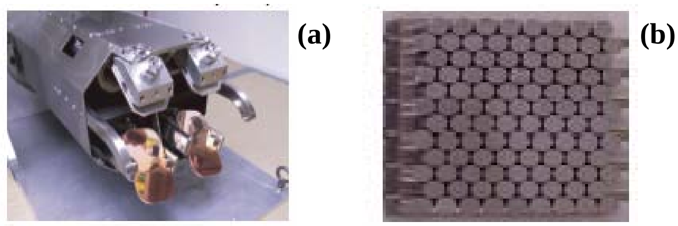

Figure 2.5-1. Collaborative fusion engineering science research: (a) steerable electron cyclotron launcher built for DIII-D by PPPL; and (b) tungsten divertor wall element built for C-Mod by Sandia National Laboratories.

The wide range of complementary wave heating schemes used in the three U.S. facilities are important for both profile and instability control: C-Mod uses ion cyclotron and lower hybrid waves; DIII-D used electron cyclotron and ion cyclotron waves; and, NSTX uses electron Bernstein and plans to use electron Bernstein waves. Figure 2.5-1(a) shows the steerable electron cyclotron wave launcher built for DIII-D by PPPL.

NSTX is studying coaxial helicity injection in order to enable solenoid-free startup, which is important for future embodiments of the spherical torus concept.

The choice of plasma materials for ITER remains an issue. Presently ITER has plans to use beryllium walls and carbon fiber composites and tungsten brushes for the divertor. JET is undertaking tests of beryllium and will also test tungsten. C-Mod has molybdenum walls and in-situ boron coatings and is testing tungsten brushes; the tungsten divertor wall element built for C-Mod by Sandia National Laboratories is shown in Fig. 2.5-1(b). DIII-D uses all carbon and may test hydrogen recovery with oxygen baking. NSTX uses carbon; it plans to test liquid lithium for pumping and as a divertor target. NSTX is also using lithium pellets and plans to use a lithium evaporator. Both C-Mod and NSTX have ITER-level divertor power densities.

For density control in H-modes, DIII-D uses divertor cryopumping. C-Mod plans to install a cryopump and the neutrals will be in a fluid regime, as compared with the kinetic regime in DIII-D. C-Mod operates at ITER-level power density and opacity in the scrape-off layer. A variety of fueling schemes is in use, including pellet injection on all three facilities . NSTX plans to install a compact torus injector.

The detection, avoidance, and mitigation of disruptions in modes of operation important for ITER are being studied in C-Mod and DIII-D.

Significant contributions have been and will continue to be made by the U.S. facilities in demonstrating integrated operational scenarios that maximize performance – β, transport, impurity minimization, steady state – while minimizing undesirable wall and divertor interactions. The Multiple Input Multiple Output dynamic control system in DIII-D uses inputs from profiles and instability mode behavior; this work is very important for ITER.

The MDSplus data management system developed at C-Mod is now used at 30 sites internationally, as well as for ITER databases and for remote access to JET. This system could be very useful for international operation of the ITER facility.

The SciDAC-funded Fusion Collaboratory tools have been implemented on all three facilities to enable remote participation and serve as a prototype for U.S. participation in ITER operation.

There are areas in which the U.S. facilities will not make a contribution. Key are:

- Operation with burning deuterium-tritium plasmas. Following the shut-down of TFTR, JET remains the only experiment with this capability.
- Operation with ultra-long to "steady state" plasmas. Tore Supra operates for 100's of seconds with a limiter plasma. The superconducting divertor tokamaks – EAST and K-STAR – will make important contributions after they start-up and have had time to optimize their complement of heating, current-drive, and diagnostic system.

3. Contributions to and Cooperation with the International Community

This section presents an overview of the research contributions made by the U.S. facilities to the international program in burning plasma research and of their future plans with emphasis on the near term (i.e., ITER). The research on the U.S. facilities will be placed in the context of the cooperative research effort carried out with international facilities. This section, therefore, addresses Question #3 of the charge to the Panel.

As background to this section, key plasma parameters and machine parameters of C-Mod, DIII-D and NSTX are provided in Appendix 1 Tables A1-1 and A1-2 respectively. Table A1-3 of Appendix 1 provides the same key plasma and machine parameters for a subset of operating international facilities (ASDEX-U, FTU, JET, JT-60U, MAST, TCV), and for ITER.

The current framework for coordinating much of the international tokamak research is the International Tokamak Physics Activity (ITPA), which aims at cooperation in development of the physics basis for burning tokamak plasmas. The ITPA includes researchers from several countries (including the European Union, Russia, U.S., and Japan) and covers several toroidal devices (viz., tokamak, stellarator, and spherical torus). Over 50 U.S. scientists are members of the various ITPA working groups, and the overall head of the ITPA is also a U.S. scientist.

Each year, the ITPA agrees on a list of experiments, analyses, and modeling efforts that will be completed on U.S. and international facilities in the following areas: confinement database and modeling; transport; pedestal and edge physics; divertor and scrape-off layer; MHD disruptions and control; steady-state operation; and diagnostics. It is important to note that the U.S. research for ITPA comprises a strong subset of the fundamental science and technology for a robust fusion energy *sciences* program, independent of any particular confinement approach.

The combined contributions of the three major U.S. facilities constitute an important part of nearly all of the ITPA tasks. At a recent meeting, the ITPA enumerated several advancements in the tokamak physics basis. The U.S. facilities played important roles in each of these advancements:

- Achievement of good H-mode confinement at ITER plasma density;
- Development of alternative H-mode regimes with ITER-required performance at lower plasma current and higher edge safety factor;
- Demonstration of enhanced helium exhaust by elastic scattering;
- Improvements in theory-based confinement validation and projection;
- Confirming scaling-based predictions of fusion amplification factor;
- Progress in elucidating the coupling between plasma edge conditions and core confinement;

- Improved understanding of edge localized mode (ELM) physics, developments in ELM mitigation techniques, and alternative ELM regimes with high confinement;
- Demonstration of active feedback control of neoclassical tearing modes by electron cyclotron wave current wave and resultant improvement in the sustainable β values;
- Demonstration of increased β-limit by stabilization of resistive wall modes;
- Demonstration of disruption mitigation by strong impurity puffing.

To draw from the FESAC Program Priorities Report, burning plasma research encompasses the two themes of “create a star on earth” and “develop the science and technology to realize fusion energy.” It involves the *integration* of the fundamental science and technology issues that were discussed in Section 2. In the near term, the research will have application to ITER, in terms of qualifying its existing design and motivating possible improvements. Research for the longer term will include a broad range of generic burning plasma issues, which will help validate the physics of a demonstration power plant (DEMO).

The ITER schedule envisions first plasma operation ten years from the start of construction. During those ten years, the U.S. will contribute about ten percent of the funds for construction of ITER, building several of its major systems. At the same time, coordinated research on the U.S. facilities will prepare us for full and effective participation in ITER operation and research by: (1) increasing confidence in the current ITER design, e.g., the choice of wall material; (2) providing information for design decisions not yet finalized, e.g., details of heating systems; (3) suggesting possible improvements to the baseline design, e.g. magnetic control coils for reducing field errors and stabilizing MHD mode; (4) developing new measurement techniques, diagnostics, and control systems controls, e.g., feedback sensing and mitigation of plasma disruptions; and (5) enhancing theory and integrated modeling, so that advanced simulation capability will be available to design experiments on ITER. More general research for a burning plasma experiment to provide scientific understanding and concept improvements will also be pursued (e.g., advanced scenarios in highly shaped plasmas, high toroidal magnetic fields, different aspect ratios), even though not immediately applicable to the present ITER design.

The strength of the three U.S. facilities *in combination* will be demonstrated by means of a few illustrative examples of how they have directly contributed to fusion science understanding, with impact on the ITER design.

3.1. Transport Basis for ITER Operating Scenarios

The U.S. leads the world in transport research, including the invention of new diagnostics to measure the underlying turbulence at the important wavelength scales and the development of comprehensive first-principles transport models. This work has focused around the U.S. Transport Task Force since the 1980’s and, more recently, around the

ITPA. C-Mod examines confinement at high plasma density, with equilibrated electron and ion temperatures, and has novel diagnostics, including the phase contrast imaging system. DIII-D has a comprehensive diagnostic set for measuring fluctuations over a wide range of wavelengths, and has developed the most comprehensive transport code in the world for detailed comparisons of experiment with theory. NSTX examines how transport scales with aspect ratio at high β, and is commissioning fluctuation diagnostics that cover the relevant wavelength regimes.

The integrated theory and diagnostic programs of the U.S. have identified important issues not evident from multi-machine scalings. In the mid-1990's, core transport models such as IFS-PPPL and GLF23 demonstrated that ITER's performance projections are very sensitive to the assumed electron temperature at the top of the H-mode pedestal, prompting a world-wide emphasis on pedestal research. C-Mod (in June, 1997) and ASDEX-U (in July 1997) were the first devices to publish the confirmation of a strong correlation between the pedestal electron pressure and the confinement enhancement factor. Subsequent similarity experiments among C-Mod , DIII-D and ASDEX-U and JET have sought to isolate the dependence of the density pedestal height and width on the neutral penetration depth, and a recent experiment among NSTX, MAST, and DIII-D seeks to isolate the effect of aspect ratio on the pedestal parameters.

Despite the progress in core and pedestal research, more work remains to be done in transport science. For example, the physics of electron and momentum transport is not well understood and represents a leading frontier in transport research, with each machine making unique contributions. Still, the U.S. transport program has improved—and will continue to improve—the reliability of such transport predictions for ITER.

The projections for ITER concerning its plasma performance and access to H-mode confinement are based on an international database, for which data from C-Mod, DIII-D, and NSTX are important for an accurate determination of the relevant scaling variables. In combination with the larger international machines such as ASDEX-U, JET and JT-60U, C-Mod and DIII-D provide data crucial for an accurate size extrapolation to ITER. In addition, NSTX data facilitates an evaluation of the effect of aspect ratio differences among machines in scaling up to ITER.

3.2. Wall Lifetime with Bursts of Heat and Particles

The heat loads from transient phenomena such as edge localized modes (ELMs) are predicted to cause ablation of plasma-facing components in ITER. This has stimulated world-wide research on high-performance regimes with small or no edge localized modes. The U.S. facilities have (1) developed and benchmarked models of the edge localized mode, (2) developed operational regimes with tolerable edge localized modes, and (3) pioneered the use of edge resonant magnetic perturbations to avoid edge localized modes altogether.

- For (1), all three machines have world-class diagnostics for characterizing edge localized modes and measuring intermittent, cross-field transport.
- In (2), DIII-D developed a high-performance, low-collisionality “quiescent H-mode” (QH), which has no edge localized modes but achieves an acceptable cross-field particle transport rate through edge MHD activity. The quiescent H-mode has been subsequently reproduced in several international machines, and its extrapolability to ITER is being assessed. The direction of neutral beam injection is important in this mode, and JT-60 has done balanced injection, i.e., with no torque input. One of the DIII-D beam lines is presently being reversed, in order to extend the JT-60U experiments. C-Mod discovered the “enhanced D-alpha H-mode” with no edge localized modes, which also relies on edge MHD activity for particle control. In addition, NSTX found a new small-ELM regime (Type V edge localized modes), which allows sufficient particle control with minimal thermal transients; the similarity between the NSTX regime and small-ELM regimes in C-Mod and MAST is being studied through the ITPA.
- In (3), DIII-D pioneered the use of internal coils to apply an n=3 magnetic perturbation, which suppresses edge localized modes but enhances cross-field diffusive particle transport. Similar experiments are being conducted in NSTX, with edge localized mode mitigation sometimes observed. The DIII-D success with resonant magnetic perturbations has led to the planned installation of edge localized mode control coils in JET. The U.S. is also developing the physics basis to propose this system for ITER. Just as international researchers will participate in experiments on DIII-D, U.S. researchers are involved in upcoming JET experiments. An international database is being developed, which could eventually be incorporated into the ITER design.

3.3. Machine Damage due to Sudden Disruptions

The plasma in ITER will contain a large amount of stored energy. If MHD instabilities grow, the plasma pressure can – either locally or globally – exceed stable operating limits of the confining magnetic fields. The plasma can be quickly lost (“disruption”), and this can result in large forces on the tokamak mechanical structure and also large heat loads on the walls. The U.S. has been the leader in experiments and modeling to detect an impending disruption and then apply a large puff of gas to safely extinguish the disrupting plasma. DIII-D did the first such mitigation experiments, along with detailed measurements and modeling of the physical processes. This work has now been expanded to C-Mod, which provides a critical test of the gas puff technique due to its high absolute plasma pressure. The effort to determine the scaling and distribution of disruption forces has been led by C-Mod. The ITPA has a joint program of disruption experiments among C-Mod, DIII-D, and JET; these machines span the parameter range from high absolute plasma pressure in C-Mod to large plasma size in JET. NSTX has contributed in a somewhat unexpected way: because spherical tori exhibit moderate resilience against disruptions, understanding this property may allow application in higher aspect ratio devices. Reducing the severity of disruptions in ITER is important for allowing a safety margin in the engineering design.

3.4. Tritium Inventory and Choice of Wall Materials

The present ITER design calls for beryllium main walls and a combination of tungsten and carbon walls in the divertor region. Most experience on international tokamaks has been with carbon surfaces at high heat and particle fluxes. Carbon is fairly forgiving with respect to large transient heat loads, such as disruptions, and it does not radiate strongly if it migrates to the core plasma. However, carbon that is eroded from the wall can co-deposit with the tritium fuel on the vessel walls, resulting in an undesirably large inventory of trapped radioactive tritium. The U.S. facilities, along with international tokamaks, are attacking various parts of this problem with two different and complementary approaches: namely, (1) to further characterize carbon erosion and re-deposition, carbon transport, and surface film and dust generation, as well as to study tritium removal techniques; and (2) to examine other materials, such as molybdenum and tungsten, which do not lead to tritium accumulation, but which can melt with disruptions and can radiate strongly if they penetrate into the core plasma. DIII-D is an all-carbon machine and is focusing on carbon transport experiments to determine where the tritium would be co-deposited and on in-situ techniques to remove the tritium. NSTX is mostly carbon at present; since it plans to add lithium to the divertor region for an integrated power and particle control solution, it will provide a good test of mixed materials. C-Mod is an all-metal molybdenum machine. It is adding tungsten in a unique “brush” structure, similar to the ITER design, to address the melting issues. The U.S. machines can address a comprehensive set of issues relevant to the selection of the wall material for ITER. They also complement international facilities: ASDEX-U (gradually converting to tungsten-coated carbon), JT-60U (carbon), and JET (beryllium and carbon, with plans in the next few years for beryllium walls with a tungsten divertor). In as much as the ITER divertor is a modular “cassette,” there is still time to influence its design and choice of materials. It may also be the case that different wall materials will be used during the startup phase (without tritium) and the later operational phase of ITER.

3.5. MHD Stability, Plasma Rotation, and Feedback Control with Magnetic Coils

The U.S. has been the leader in research on the characterization and control of macroscopic MHD instabilities, particularly the resistive wall mode and the neoclassical tearing mode. DIII-D and NSTX have demonstrated that the resistive wall mode can be stabilized with sufficient plasma rotation. NSTX revealed the safety factor dependence of the rotation required for resistive wall mode stabilization, and joint work between NSTX DIII-D, guided by theory, is revealing the aspect ratio dependence. Some uncertainty remains concerning the level of rotation for mode stabilization in ITER. Research to be conducted on DIII-D and NSTX in the next five years should clarify the physics mechanism for resistive wall mode stabilization, thus allowing more accurate calculation of the critical rotation speed required for stabilization. C-Mod was the first to discover plasma rotation with radio frequency wave heating, even though it provides no external momentum input; understanding this phenomenon could be a key element in determining the level of rotation in ITER. DIII-D and NSTX will conduct experiments in slowly rotating target plasmas with external control coils in order to study the requirements for stabilizing the resistive wall mode if ITER does not have sufficient rotation. DIII-D has demonstrated feedback stabilization of the mode; the installation of a counter-directed neutral beam will allow more variation in rotation and further exploration of feedback stabilization scenarios.

Stabilization of neoclassical tearing modes by means of localized current driven with radio frequency waves was pioneered by ASDEX-U and DIII-D. DIII-D has recently demonstrated closed-loop feedback stabilization of this mode, resulting in higher β plasmas; planned modulation experiments will attempt to reduce the amount of radio frequency wave power that would be required in ITER. Scaling studies among DIII-D, JET, and ASDEX-U have been used to understand the underlying physics mechanism and to extrapolate the results to ITER. NSTX will use a different form of radio frequency current drive (based on electron Bernstein waves) for neoclassical tearing mode stabilization.

Departures from axisymmetry in the structure of the tokamak magnetic field can destabilize non-rotating tearing modes ("locked modes") that may significantly impact plasma operation. Locked modes induced by such error fields can lead to disruptions, as well as degradation of confinement. The impact of these modes on the operation of future burning plasma experiments, such as ITER, has been a matter of concern; key issues are the prediction of the error-field sensitivity (threshold perturbation) in such devices and the requirements for corrective measures. Joint experiments through the ITPA that make use of non-axisymmetric coils to apply known field perturbations are being undertaken among C-Mod, DIII-D, and JET in order to provide a basis for extrapolative prediction (in field and size) of the symmetry requirements for ITER. These experiments are conducted with matching shape and normalized plasma parameters in the three devices, spanning a range of size of nearly a factor of five. Preliminary results comparing C-Mod and JET indicate that the required symmetry is on the order of one part in 10^4, which is within the capability of the planned ITER correction coil system. Earlier projections had indicated a more stringent requirement, in the range of one part in 10^5. NSTX has also conducted similar experiments, using the new trim control coil capability of the device. JT-60U, which is installing iron inserts in the toroidal field coils to better symmetrize its toroidal magnetic field, will also add to this database.

3.6. Energetic Particles, Heating, and Current Drive

The combined capabilities of the U.S. facilities lead the world in the ability to vary the fast particle energy over a wide parameter range, in unique diagnostics, and in theoretical modeling. The normalized gyro-radius of the fast particles ($\rho*_{fast}$) varies from low values on C-Mod to very high on NSTX. In addition, NSTX has a very wide operating space in terms of V_{fast}/V_{Alfven} and $\beta_{fast}/\beta_{tot,}$ with a comprehensive diagnostic set including the motional Stark effect operating in the low magnetic field regime; its operating space contains the predicted ITER operating range in terms of these two parameters. C-Mod has active MHD spectroscopy diagnostics, and DIII-D has a comprehensive diagnostic set, including a new fast ion diagnostic and the ability to vary the fast ion population. As a combined package, the U.S. machines provide important information for ITER.

In most cases, the density and energy of fast particles are related to the auxiliary heating and current drive scenarios. The U.S. facilities can examine mixed regimes with neutral beam and radio frequency waves (DIII-D and NSTX) and with radio frequency waves only (C-Mod). Furthermore, the U.S. has a strong capability for modeling these results and developing validated, predictive models for ITER, particularly with electron cyclotron current drive in DIII-D and fast wave current drive in DIII-D and NSTX, and both ion cyclotron heating and lower hybrid current drive on C-Mod. Adding to the physics understanding of radio frequency heating and current drive, NSTX will also carry out experiments with electron Bernstein waves, and C-Mod with lower hybrid waves.

3.7. Advanced and Hybrid Operational Modes

The U.S. is a major contributor, with ASDEX-U, JET, and JT-60, in the development of advanced tokamak operational modes. DIII-D has obtained highly shaped, double-null diverted plasmas at high normalized pressure β_n and high confinement (H-factor). C-Mod has achieved the highest absolute pressure plasmas in the world, and NSTX has the highest normalized pressure (i.e., relative to the magnetic field pressure) with its strong plasma shaping and broad pressure profile. All of these advanced operational modes are possible directions for improving the tokamak concept. The U.S. program has emphasized tokamak innovations and has already made improvements in the baseline ITER design. Several of the advanced operational modes are being developed to a sufficient extent that they will eventually be part of the ITER design. In addition, DIII-D and ASDEX-U have led the effort to obtain "hybrid" discharges, which rely less strongly on inductive current and have parameters that lie between the conventional ITER and an advanced tokamak. If realized, these operating modes would provide easier access to high-Q plasmas in ITER with lower plasma current.

4. What Would Be Lost, and Recommendation

The three U.S. experimental programs have a superb track record of collaborative discovery and innovation within the U.S. and within the international program. They have the most extensive set of diagnostics, heating and fueling systems, and control systems, as well as a strong connection to theory and superb modeling capability. Alcator C-Mod is the world's only diverted, high field tokamak, featuring all radio frequency heating and high power flux to a unique metal wall/ divertor. DIII-D is the best equipped and most flexible tokamak in the world. NSTX has the most capabilities of any spherical torus facility, and it can study electromagnetic effects at ultra-high β values. In total, they have a unique set of capabilities for a coordinated effort to explore advanced plasma science relevant to fusion in the range of all key parameters – except machine size and alpha heating power. Moreover, each program plays a major role in the education of the next generation of fusion energy scientists and engineers.

The charge Question #4 (part 2) – "What research opportunities would be lost by shutting down one of the major facilities [during the next five years]?" – is addressed in terms of the U.S. fusion energy sciences program. In Sec. 4.1, this Question is treated from the perspective of the generation of essential knowledge for fusion energy sciences, in terms of goals and accomplishments that are expected for scientific campaigns. In Sec. 4.2 this question is treated from the perspective of each of the three major U.S. facilities. In conclusion, Sec. 4.3 presents the Panel's central recommendation. The Panel notes that it is important for the U.S. to be viewed as a major player in the world program in order for the U.S. to effect the science on and reap full benefit from ITER, and that only a folio of good research results buys a seat at that sort of decision table.

4.1. Scientific Campaign Goals and Expected Accomplishments

For each of the scientific campaigns identified in the 2005 Program Priorities Report[1] we summarize the principal elements related to accomplishments that are expected during the next five years, for a constant level of effort. In some of these elements, good progress has been made already. Cancellation of one or more facilities (parenthetically identified by name, with "All" meaning all three of the facilities) would lead to either the loss of an element or the weakening of a unique coordinated effort.

<u>Macroscopic Plasma Physics</u> *(cf. this report Sec. 2.1):*

Substantial progress will be achieved in understanding the role of magnetic field structure on plasma confinement, both for fundamental science and in direct preparation for ITER [**All**]. Experiments to understand pressure limits in rotating plasmas with resistive walls should be completed, and studies of active stabilization without rotation will be underway [**DIII-D, NSTX**]. The understanding of neoclassical tearing instabilities and edge localized modes will be complete and documented, including methods of control and suppression [**All**]. An understanding will be developed of mode stabilization due to plasma rotation, as well as the physical processes responsible for increased drag due to plasma modes **[DIII-D, NSTX]**. Studies will be performed of different types of plasma self-organization and their interaction with external control methods [**All**]. Methods will be developed to sustain the plasma duration, including "hybrid operating modes" for ITER **[All].** Scientific issues associated with the integration of high plasma pressure, good confinement, and efficient sustainment of plasmas, including bootstrap current, will be investigated [**All**]. The mechanisms for relaxation and reconnection of the magnetic field will be identified [**DIII-D, NSTX**].

<u>Multi-scale Transport Physics</u> *(cf. this report Sec. 2.2):*

The goal of developing a predictive capability for ion thermal transport is within reach [**All**]. Turbulence-driven electron cross-field thermal transport remains to be understood at the fundamental level. As with ion transport, studies over a parameter range broader than that accessible by one machine will be required. The three U.S. facilities are sufficiently distinct to provide this range, and are experienced in carrying out the necessary comparisons. Small-scale turbulence diagnostics are being implemented to explore the role of turbulence on electron thermal transport, and within ten years the full wavelength and frequency ranges of the dominant turbulence should be identified. Loss of any of the three facilities will severely compromise this research area [**All**]. Particle and momentum (rotation) transport studies are less mature, and there are known surprises that must be understood; within five years, some convergence can be expected on which physical mechanisms are most important [**All**]. A more complete understanding of the conditions and thresholds for the formation of edge and core transport barriers, and their dynamics, will be obtained. The expectation is that transport barriers can then be used as a tool to control the levels of core transport [**All**]. Within ten years it will be possible to compare theoretical and numerical models with experimental data on zonal flows and their effects [**All**]. Large scale flows in plasmas without dominant external flow drive will be documented and analysis should enable predictions for ITER [**All**].

<u>Plasma-Boundary Interfaces</u> *(cf. this report Sec. 2.3):*

First-principles models of the edge pedestal will be available to reproduce many of the measured characteristics of the plasma boundary in one or more fusion experiments [**All**]. Experiments will be carried out to validate these plasma-boundary models for ITER [**All**]. The physics of edge localized modes and their impact on the scrape-off layer will be largely understood, enabling the development of new mitigation techniques and operational regimes to minimize their impact [**All**]. The fundamental characteristics of scrape-off layer turbulence in high-confinement plasmas will be determined [**All**]. There should be sufficient data to quantify the major impurity sources, including divertor and first wall erosion and deposition rates [**All**]. The distribution of tritium that is expected to be trapped in carbon-based plasma-facing components will be largely understood in terms of plasma conditions observed in present experiments. Candidate techniques for tritium removal will be developed in carbon-based components [**DIII-D, NSTX**]. Operation with potentially low tritium retention will be performed with a purely metallic wall [**C-Mod**]. Alternative high-heat flux components such as liquid lithium divertors will be qualified [**NSTX**]; tungsten brush divertor components and all-metal walls will be utilized [**C-Mod**].

<u>Waves and Energetic Particles</u> *(cf. this report Sec. 2.4):*

Externally launched waves are the only controllable means of providing core plasma heating and the necessary precision control of the current profile in advanced tokamak scenarios in ITER and in future experiments. Considerable advances are expected in developing the fundamental physics models for wave coupling, propagation, absorption, and plasma responses that will enable coupling on critical time scales with plasma stability and transport models. Experiments with lower hybrid and ion cyclotron waves [**C-Mod**], electron and ion cyclotron waves [**DIII-D**], fast waves and electron Bernstein waves [**NSTX**] will validate these models. Improved understanding of radio frequency sheath effects and antenna loading will be achieved [**All**], and with an ITER-like scrape-off layer [**C-Mod**]. Wave heating and current drive simulations will be benchmarked in support of ITER [**C-Mod, DIII-D**]. Prior to the advent of full burning plasma operation in ITER, progress is expected in understanding the behavior of energetic particles (generated by auxiliary heating) and of the unstable waves that can be excited by them with the use of existing facilities, and in improving theory and simulations for energetic particle dynamics [**All**].

<u>Fusion Engineering Science</u> *(cf. this report Sec. 2.5):*

Considerable progress is expected in developing the knowledge base required to determine performance limits and identify innovative solutions for the plasma-facing components and divertor materials [**All**]. Advances will be made in fueling, wave heating, current drive, and plasma control systems [**All**]. Support will be provided for the construction and reliable operation of ITER, including disruption mitigation [**C-Mod, DIII-D**] and support for a candidate Component Test Facility [**NSTX**].

<u>Burning Plasmas</u> *(cf. this report Sec. 3):*

The three U.S. machines taken together provide a wide set of parameters to determine relevant physics parameters for ITER and future burning plasma devices [**All**]. The physics of the edge pedestal is critical for ITER, and the three machines all contribute to this ongoing effort [**All**]. Control of exhaust heat and particles is being approached with different wall materials in the three machines [**All**]. Experiments on the understanding and control of ELMs, a short burst of heat flux, will be undertaken on all three machines, along with coordinated scaling experiments on international machines [**All**]. The physics of plasma rotation and its role in MHD stability will be investigated; this is important in determining whether ITER will be stable to various modes such as the resistive wall mode [**DIII-D, NSTX**]. Experiments on facilities with no apparent torque input when only radio frequency waves are used for heating and current drive [**C-MOD**] will be compared with experiments in facilities in which the torque can be varied [**DIII-D, NSTX**].

4.2. What Research Opportunities Would Be Lost

Here we address the charge Question #4b specifically in reference to each of the three major U.S. facilities.

C-Mod is distinguished by the following salient characteristics: (1) it operates at higher magnetic fields than any other existing divertor tokamak, and over a range that spans the ITER magnetic field value; (2) it has all-metallic (high atomic number) tungsten and molybdenum wall armor and divertor plates; and (3) its non-inductive heating and current drive are supplied entirely by flexible radio frequency techniques. C-Mod is also distinguished by its location at a major research university and, as such, hosts the largest number of graduate and undergraduate students.

The loss of C-Mod would:

- Eliminate studies of tritium fuel retention and power flow to the wall and divertor in ITER-relevant edge conditions, which have potentially serious consequences for ITER.
- Eliminate ITER-relevant tests of thermal, particle and momentum transport with radio frequency heating and current-drive techniques in plasmas with the ITER-like characteristics of ions and electrons coupled at the same temperature through collisions, no core external momentum-drive, and high-pressure edge conditions.
- Compromise development of ion cyclotron and lower hybrid RF capabilities for controlled plasma heating and edge current drive essential for advanced tokamak scenarios in ITER and future burning plasma facilities.

DIII-D is the best equipped – in terms of diagnostics, heating and control systems – and most flexible tokamak –in terms of shaping capability – in the world. It has an outstanding record of contributing to plasma science and technology, and to the development of advanced scenarios for tokamak operation that offer the possibility for ITER to achieve its goals at reduced plasma current. Its capabilities have been enhanced, and this excellent program is expected to continue producing centrally important information for the advancement of science and the magnetic fusion program.

The loss of DIII-D would:

- Eliminate a world-class program that contributes to understanding turbulent transport, fast ion instabilities, pedestal and divertor physics, and mode stabilization.

- Greatly reduce U.S. leadership in the world-wide development of a high β, high bootstrap current, "steady-state" plasma and of hybrid operating scenarios for ITER.
- Cede the U.S. position of strength in electron cyclotron current drive for current profile and MHD stability control.

NSTX has exceptional performance and diagnostic capability among spherical tori, with highly flexible shaping capability and the ability to access ultra-high plasma β values. The facility provides an operational regime for key science theories to be tested and confirmed, providing understanding for tokamak fusion devices in general.

The loss of NSTX would:

- Eliminate not only the U.S. leadership position in the world for device capability and research on spherical tokamaks, but also eliminates the U.S. ability to guide and contribute to spherical torus research at the "proof-of-principle" level.
- Eliminate numerous important experiments such as Bernstein wave current drive, plasma start-up in a proof-of-principle scale experiment, ultra-high β operation, and fast ion measurements.
- Eliminate U.S. leadership in high β spherical tokamak research, and the U.S. independence to pave the path to the future strategic option of constructing a Component Test Facility based on the spherical torus.

4.3. Recommendation

In a February 2005 report titled *'The Knowledge Economy: is the United States losing its competitive edge?,'* the Task Force on the Future of American Innovation "developed a set of benchmarks to assess the international standing of the U.S. in science and technology. These benchmarks in education, the science and engineering workforce, scientific knowledge, innovation, investment and high-tech economic output reveal troubling trends across the research and development spectrum. The U.S. still leads the world in research and discovery, but our advantage is rapidly eroding, and our global competitors may soon overtake us."[5]

In the particular area that is the subject of this report, that of the scientifically interesting and broadly important topic of magnetic fusion energy research, the Facility Panel finds that the U.S at present holds a position of international strength and leadership. The next major magnetic fusion research facility will be the offshore burning plasma experiment, ITER. Fostered by the recent decision over its ITER site, interest in ITER is rapidly growing in the international research community. The loss of any of the three major U.S. toroidal fusion facilities would fundamentally jeopardize the ability of U.S. researchers to perform relevant fusion research, and thus would undermine the current U.S. position of international excellence.

The three major U.S. magnetic fusion facilities – C-Mod, DIII-D, and NSTX – represent a massive investment of talent, intellect, and finances in tackling the key issues of toroidal confinement. Each of these facilities has made seminal contributions to the development of toroidal confinement and to the fundamental science and technology that undergird it. The wealth of discoveries and the generation of knowledge made possible by the three coordinated U.S. facilities has enabled the U.S. to be an effective presence among the larger foreign programs involved in ITER. Premature closure of one of these three major facilities would seriously compromise the effectiveness of the U.S. fusion program internationally and also the U.S. ability to advocate future proposals for advanced performance scenarios that could lead to a more economically competitive high-power-density fusion system.

The Panel's recommendation is that the three major United States toroidal magnetic fusion facilities continue operation to conduct important unique and complementary research in support of fusion energy sciences and ITER.

5. References

1. *Scientific Challenges, Opportunities, and Priorities for the U.S. Fusion Energy Sciences Program: A Report to the Fusion Energy Sciences Advisory Committee,* (Apr 2005); http://www.ofes.fusion.doe.gov/more_html/FESAC/PP_Rpt_Apr05R.pdf .

2. *Fusion Simulation Project: Integrated Simulation & Optimization of Fusion Systems,* (Dec 2002); http://www.ofes.fusion.doe.gov/News/FESAC/FSP_report_Dec9.pdf .

3. Presentations at the 20th IAEA Fusion Energy Conference, Vilamoura, Portugal, (Nov 1-6, 2004); http:// www.cfn.ist.utl.pt/20IAEAConf/Announcement.htm .

4. *Device schedules/Program elements in 2005*, compiled by O. Gruber. 3rd Joint Workshop of IEA Implementing Agreement on "Implementation of ITPA Coordinating Research Recommendations," Eynsham Hall, U.K. (Dec 8-10, 2004).

5. *The Knowledge Economy: is the United States Losing its Competitive Edge? Benchmarks of our Innovation Future*, Task Force on the Future of American Innovation (Feb 2005): http://www.futureofinnovation.org .

Appendix 1. Tables of Characteristics of the Three Major U.S. Toroidal Magnetic Fusion Facilities

In this Appendix are provided tabulated values of the characteristic plasma (Table A1-1) and machine (Table A1-2) parameters for the three major U.S. toroidal fusion facilities. In Table A1-3, parameters for a subset of international facilities are compared with those for ITER [http://www.iter.org].

Table A1-1. Plasma Parameters [present {future}].

Plasma Parameters	Alcator C-Mod	DIII-D	NSTX
$\langle n \rangle_{max}$ ($\times10^{20}$ m^{-3})	6.0	3.0	0.7
$T_i(o)_{max}$ (keV)	5.6	27	2.5
$T_e(o)_{max}$ (keV)	6.0	16	4.1
W_{tot}^{max} (MJ)	.25	4.2	0.43
n/n_{GW} (H89$\geq$ 1.6)	0.7	1.4	1.1
Max H89P for $5\tau_E$	2.1	3.2	2.5
T_i/T_e *[range]*	0.5 – 1.9	0.15-5	0.6 – 1.7
$\tau_E * \nu_{ei}$ *[range]*	0.5 - 100	0.2-200	4 - 14
$\nu*$ *[vol. av. value; pedestal top value]*	0.007-23; 0.5-20	0.001-3; 0.02-20	0.1 [ρ=0–0.6]; 0.5
$1/\rho*$ *[vol. av. value; pedestal top value]*	170-500; 250-1000	50-600; 120-650	LFS avg. 50 – 100; 50
$\langle\beta\rangle$ % max	1.4	12.5	40 {50}
β_N (max)	1.78	6.0	7.2 {8}
$\beta_N H_{89}$ (for 1 τ_R)	3.1	9 {12}	12
$\varepsilon\beta_{pol}^{max}$	.77	1.8	1.6
f_{BS}^{max}	.22 {.7}	0.85	0.6
β_{fast}/β_{tot} *[range]* (volume averaged β)	0.07-0.73	0 – 0.6	0 – 0.75 (β_{fast} 0 – 25%)
V_{fast}/V_{Alfven}	0.2-0.8	2.5	1-5
<Pressure> MPa	0.18	0.125	0.022
f_{NI} *%;* (τ_{pulse}) / (τ_R)]	{100 %, 10}	100 %; 9	70 %; 4 {100%, 10}
SOL: density; collisionality range	1×10^{18} to 1×10^{20} m^{-3}; 10-600	0.4-3.4 $\times10^{20}$ m^{-3}; 0.1-60	0.1 – 1 x 10^{19} cm^{-3}; 2 - 60
$Prad_{div}/P_{SOL}$ (H89$\geq$ 1.6)	Up to 80%	70%	~ 30%

Table A1-2. Machine Parameters [present {future}].

Machine Parameters	Alcator C-Mod	DIII-D	NSTX
R (meters); R/a *[range]*	0.61-0.74; 2.8-3.6	1.49-1.88; 2.5-4.5	0.8-1.0; 1.27-1.6
κ; δ *[range]*	0.9 – 1.85; 0 - .84	1.15-2.6; -0.1-1.0	1.5-2.7; 0.2-0.8
B (Tesla)*[range]*	2-8	0.5-2.2	0.3-0.45 {0.6}
I (MA) *[max]*	2.05	3.0	1.5
Plasma Vol. (max, m^3)	1.0	24	14
τ_{pulse} (sec) *[max at full field; at 75% nominal]*	1 @ 8 Tesla; 3 {5} @ 5.3 Tesla	6 {10} @ full field; 15 {25} @ 75% nominal	1.5 @ B_T = 0.45T; 3.5 @ B_T = 0.34T
Heating: - Type - MW	ICRF - 8 MW; LHRF – 3 {4} MW	+NBI 17.5 {20}; +NBI 12.25 {14} @ 5 s; +ECH 4.5 {6.75}; +FW 3 {6}	NBI 7.5MW; HHFW 6MW; {EBW 4MW}
Current Drive: - Type - MW	LHCD – 3 {4} MW; MCCD – 8 MW; FWCD – 8 MW; ICCD – 8 MW;	+NBCD 8.75 {10.5} co / 3.5 ctr @ 5 s; +ECCD 4.5 {6.75} co/ctr; +FWCD 3 {6} co/ctr	NBI 7.5MW; HHFW 6MW; CHI 0.4MA {0.5MA}; {EBW 4MW}
Nearby Conducting wall *[yes or no]*	Inner (high-field side): close; Outer (low-field side): far; Effects of outboard structures (limiters, antennas): {TBD}	Yes	Yes, w/a_{plasma} ~ 1.3 -1.5
Error Field Correction coils *[yes or no]*	Yes	Yes	Yes
RWM feedback coils *[internal, external, or none]*	No	Internal and External	External
NTM stabilization (e.g. EC, LH, EBW)	MCCD, ICCD, LH	ECCD: proven stabilization of 3/2 & 2/1 NTM	{EBW}
First Wall and Divertor Material	molydenum {tungsten}	Carbon, periodic boronization	CFC/Graphite, Li coating {Lithium target}
Divertor *[shape, pumped or not]*	LSN, USN, DN, inner-wall limited, lower nose limited {pumped, 2005}	LSN (pumped), USN (pumped), DN (pumped)	Double null; upper, lower single null {lithium}{Cryo-pump}
Power/R (MW/m)	9.0 {13.4}	14.7 {19.3}	16 {20}
Power/Surf.Area (MW/m^2)	0.86 {1.3}	0.46 {0.60}	0.4 {0.5}
Power/Vol. (MW/m^3)	6.7 {10.0}	1.3 {1.7}	1.1 {1.3}
Fueling *[list of options]*	Low pressure gas; high pressure gas; D_2 pellets; Li pellets	Pellets: outer, top, inner (radial), inner (45 deg), top (ELM pacemaking); Gas: divertor, inner wall, outer, through divertor baffles (top & bottom, toroidally symmetric)	Gas injectors: LFS, HFS, divertor Supersonic Gas Injector NBI Solid pellet injector (Li, B, C, {doped}) {deuterium pellet injector} {CT injector}
Facility Replacement Value, in current $ US	$180 M	$694 M	$ 464 M

+ Power launched into plasma

Table A1-3. Comparison with International Machines.

Parameters	ASDEX-U	FTU	JET	JT-60U	MAST	TCV	ITER
R (m); R/a	1.65; 3.2	0.935; 3.2	2.9; 3.1	3.2; 4.0	0.85; ≥ 1.3	0.89; 3.6	6.2; 3.1
κ; δ	Up to: 1.80; 0.55	1.05,; 0	1.9; 0.5	1.8; 0.4	2.5; 0.5	1-2.8; -0.6-0.6	1.7; 0.35
B (Tesla)	1.5-3	8	3.8	4.8	0.52T @ R	1.43 @ R	5.3
I (MA)	Up to: 1.4	1.6	4.5	5.0	1.4	0.8	15
W_{tot}	Up to: 1.5 MJ	0.14 MJ	17 MJ	11 MJ	~ 160 kJ	40 kJ	350 MJ
Heating Type	NBI, ICRH, ECRH {LH?}	LH, ECRH, IBW	NBI, LH, ICH	NBI, LH, ECH	NBI, {EBW}	X-2nd, X-3rd ECRH	NB, ICH, ECH, LH
Current Drive	NBCD, ECCD {LHCD?}	LH (8 GHz)	NBCD, LHCD, ICCD	NBCD, LHCD,ECCD	NBCD, {EBWCD}	ECCD	NB, ECH, LH
First Wall; Div. Mat.	W; C	Inconel + Mo limiters	C; Be	C, periodic boronization	C	C (92% coverage)	Be wall; CFC, W div
RWM control	{Planned}		No	No	No	No	No {Yes}
ν* range	0.04 – 3 (Vol. averaged)	0.2-10	0.001-2	0.001-3	0.02 - 0.2	30 – 0.08 (T_e= 0.5-5keV, n_e=4-1x10^{13}/cm^3)	0.02
1/ρ* typical	100	450	350	400	~ 60	≥80 (Ti ≤ 1keV)	800
<β> % max	Up to 5 in I_p-ramp down; ~3.5 SS	0.3 (volume av.)	4.3	2.7	16%	2.5	2.5{3.9}
$\beta_{N,}$ max	Up to 5 in I_p-ramp down; ~3.5 SS	0.9	3.9	4.8	5.3	3.2	1.8{2.8}
β_{fast}/β_{tot} (typical)	0.05-0.3; typ. impr. H-mode with 0.1-0.2	0.0	0.2	0.4	~ 15	≤ 25 (fast electrons)	0.1
V_{fast}/V_{Alfven}	0.5-1 for NBI particles	0.0	1.0	2.0	~ 0.8 – 2.5	N/A	1.8
Divertor *(shapes with/wo pumping)*	Pol. div with cryo-pumping in lwr SN ∀ shapes, upper SN without pumping	N/A	LSN pumped	LSN pumped	LSND, USND, DND, {pumping}	SNL, SNU, DND, No	LSN pumped
P_{SEP}/R (MW/m)	Up to: 15	2.6	9	12	~ 6	5.1	12{14}
P_{SEP}/S (MW/m^2)	Up to 0.6	0.22	0.2	0.3	~ 0.2	0.26	0.11{0.12}
P_{CORE}/V (MW/m^3)	Up to 2	1.55	0.3	0.4	~ 0.5	5.6 – 10	0.6{0.8}

Appendix 2. FESAC Facilities Assessment Charge Letter

April 5, 2005

Professor Richard D. Hazeltine, Chair
Fusion Energy Sciences Advisory Committee
The University of Texas at Austin
Institute for Fusion Studies
1 University Station, C 1500
Austin, TX 78712-0262

Dear Professor Hazeltine:

Funding U.S. participation in the ITER project will place considerable pressure on the Office of Science budget in future years. I am committed to a strong base fusion research program during ITER construction but want to be sure that the program is cost-effective. In particular, I do not believe that we should continue running existing facilities just because we can do more research on them but because we can do unique and important research on them. Accordingly, I would like you to assess characteristics of the three major U.S. toroidal magnetic fusion facilities in the context of the international fusion programs and determine what contributions they can make to fusion science and the fundamental vitality of the U.S. Fusion Program during the next five years.

I would like you to consider the following issues in your assessment:

1. What are the unique and complementary characteristics of each of the major U.S. fusion facilities?

2. How do the characteristics of each of the three U.S. fusion facilities make the U.S. toroidal research program unique as a whole in the international program?

3. How well do we cooperate with the international community in coordinating research on our major facilities and how have we exploited the special features of U.S. facilities in contributing to international fusion research, in general, and to the ITER design specifically?

4. How do these three facilities contribute to fusion science and the vitality of the U.S. Fusion program? What research opportunities would be lost by shutting down one of the major facilities?

I would like to have your assessment by the end of August.

Sincerely,

/s/

Raymond L. Orbach
Director
Office of Science

Appendix 3. The Three Facilities Joint Document
[submitted to the FESAC Facilities Panel on June 2, 2005]

A3.1. Cover Letter

June 2, 2005

Dear Dr. Dahlburg:

The undersigned three Directors of the Alcator C-Mod, DIII-D, and NSTX Programs hereby submit to the FESAC Facilities Panel the attached unsolicited document to assist in the Panel's deliberations. This document seeks to describe for the Panel those unique research roles for the United States in the world fusion program enabled by the combined contributions of the three U.S. Facilities: Alcator C-Mod, DIII-D, and NSTX.

This document was jointly written by the three Program Directors undersigned, in consultation with our teams. This document reflects our mutual view that the United States fusion program and the world fusion program derive great benefit from the continued operation of all three facilities.

We hope this submission is of value to the Panel.

Sincerely yours,

Earl S. Marmar

Earl Marmar (Alcator C-Mod)

Ron Stambaugh

Ronald Stambaugh (DIII-D)

Martin Peng

Martin Peng (NSTX)

A3.2. Joint Document

Unique Research Roles for the United States in the World Fusion Program Enabled by the Combined Contributions of the Three U.S. Facilities: Alcator C-Mod, DIII-D, and NSTX.

This report identifies a set of research areas in which the combined resources of the three U.S facilities provide world leadership in answering key scientific questions. The United States is well positioned by having three facilities whose complementary capabilities enable unique research contributions. Examples of those research areas are:

1. **The role of non-axisymmetric fields in plasma performance.**
2. **The role of plasma facing components.**
3. **The physics of plasma rotation and momentum transport.**
4. **Advanced Tokamak / ST performance through off-axis non-inductive current drive.**
5. **The role of density/collisionality in edge plasma turbulence and divertor physics.**
6. **The physics of energetic particle and Alfvén eigenmodes.**
7. **The dependence of transport on beta**

1. The role of non-axisymmetric fields in plasma performance.

Locked Modes

Departures from axisymmetry in the tokamak magnetic field structure produced by the as-built coil set can destabilize non-rotating tearing modes (locked modes) which can significantly impact plasma operation. Such error-field induced modes can lead to degradation of confinement and even major disruptions. The impact of these modes on the operation of future burning plasma experiments such as ITER has been a matter of concern; key issues are the prediction of the error-field sensitivity (threshold perturbation) in such devices and the requirements for corrective measures.

The U.S. program is making key contributions in this area. All three U.S. facilities have undertaken studies of their intrinsic error fields resulting from imperfections in fabrication and assembly of their as-built coil sets. Each has installed non-axisymmetric coils capable of correcting these asymmetries. Joint experiments making use of the non-axisymmetric coils to apply known field perturbations are being undertaken among C-Mod, DIII-D, NSTX, and JET in order to provide a basis for extrapolative prediction (in field and size) of the symmetry requirements for ITER. These experiments are conducted with matching shape and normalized plasma parameters, spanning a range of size of nearly a factor of five and a range of field of a factor of 20. Preliminary results indicate that the required symmetry will be in the range of a part in 10^4, which is within the capability of the planned ITER correction coil system. Earlier projections had indicated a more stringent requirement, in the range of a part in 10^5.

Ongoing experiments are continuing to investigate the origin and impact of non-axisymmetric error fields, and the physics of locked modes. An important issue is the extent to which correction of the toroidally-coupled “sidebands” of the dominant mode (usually 2/1) is required. Experiments on DIII-D have indicated that these terms have a non-negligible impact, while C-Mod has succeeded in greatly expanding its available operating space by correcting only the 2/1 component of the intrinsic error field. It might be expected that sideband correction would be even more important in the low aspect ratio ST configuration, the investigation of which on NSTX is planned for 2005. Additional experiments have been proposed for C-Mod and DIII-D to study the effect of non-resonant 3D perturbations on magnetic surfaces near the separatrix, as well as possible effects on the H-mode pedestal.

<u>Resistive Wall Mode</u>

The pressure limit in the tokamak is set by the ideal kink mode, the lowest order plasma driven departure from axisymmetry. This limit is of primary importance since the fusion power is proportional to the square of the plasma pressure. In the presence of a conducting wall, the kink mode manifests itself as the Resistive Wall Mode (RWM) which rotates slowly and grows on the flux diffusion time through the wall. While transient operation above the no-wall limit was demonstrated in DIII-D in 1994, in a major breakthrough in 2001, DIII-D showed that it was possible to sustain operation of a tokamak above the no-wall beta limit with a plasma rotating inside a close fitting conducting shell. The key physics discovery was that above the no-wall beta limit, the RWM amplified intrinsic asymmetries in the magnetic field structure of the same helicity as the RWM. A set of coils external to the vacuum chamber counteracted this resonant field amplification, maintained the plasma rotation, and thus stabilized the RWM. The new wall-stabilized normalized beta limit is found in some cases to be twice the no-wall limit, implying possibly 4 times the fusion power and 2 times the bootstrap current fraction.

The RWM coils on DIII-D and NSTX afford the U.S a unique opportunity to study both the no-wall and with-wall beta limits over a range of aspect ratio and plasma rotation speed, normalized to the intrinsic sound and Alfvén wave speeds. ITER is seriously looking into whether such trim coils can be added to the ITER design. Evidence of rotational stabilization of the RWM has been seen since 2002 on NSTX, at very high toroidal beta values. NSTX has recently added a set of external coils for error field and resistive wall mode feedback control of the strongly rotating, very high beta plasmas. This coil set and the existing external and internal coil sets on DIII-D will enable both NSTX and DIII-D to operate plasmas above the no-wall limit by means of wall stabilization with rotation. DIII-D has a set of RWM feedback coils inside its vacuum vessel to study direct fast feedback stabilization of the RWM in plasmas with low rotation, as is expected in ITER. The rotation in DIII-D will be nulled by toroidally balanced neutral beam injection. Initial results of feedback experiments have been positive. The coils in NSTX, similar in configuration to what is being considered for ITER, will permit studies of the mode-plasma coupling physics and its dependence on sound vs. Alfvén wave physics. With the planned ability in 2005 of resonant error field correction using the newly commissioned control coils, NSTX plans to determine the effects of rotational stabilization of the RWM for durations longer than the plasma current relaxation time at beta values significantly exceeding the no-wall limit. No other tokamaks in the world have such trim coil sets to enable exploration of the tokamak operating space above the no-wall limit. While C-Mod has not yet extended its operating regime to beta values above the no-wall limit, the AT program on that device has as a long-term goal the study of optimized configurations sustained for times much longer than the current diffusion time at high plasma temperature. In this context, installation of RWM coils, based on the designs perfected on DIII-D and NSTX, may prove necessary.

2. The role of plasma facing component materials

Current projections to ITER based on carbon Plasma Facing Component (PFC) tokamaks indicate that ITER could be limited to tens to hundreds of discharges before being forced to stop and remove the co-deposited tritium from the PFC surfaces owing to a tritium inventory restriction. Carbon divertor materials are the ITER baseline design owing to their proven ability to handle both high steady-state and pulsed heat fluxes. Such a limitation on ITER operation is a very important problem.

The US facility complement has a very valuable diversity of first wall materials that is being brought to bear on the choice of ITER plasma facing component materials. The most obvious expectation is that use of high-Z PFCs (e.g. tungsten) would reduce the tritium retention in ITER substantially as the theoretical limit is 0.4 T for each carbon and <0.01T for each tungsten in the lattice. The ITER project, while considering such a change, is primarily concerned about the minimal existing operational capability (e.g. compatibility with a high performance core plasma) of such materials in the presence of high heat loads and possible melting. Alcator C-Mod is addressing these concerns by utilizing molybdenum as the PFC material for all in-vessel armor. To address the need for more tungsten-specific experience C-Mod is currently testing tungsten "brush" divertor tiles and will convert the entire high heat flux region to tungsten. These studies will also shed light on the concentration of high-Z impurities that can be tolerated in the plasma core, and the approaches to maintain H-mode plasmas in the presence of such impurities.

NSTX and DIII-D operate with carbon walls, concentrating their efforts on reducing the T retention in C and developing T removal methods that can be effective on the time scales needed. Possible solutions to the carbon problem are being pursued by research on boundary plasma flows to understand where in the divertor tokamak the codeposited layers will form, by investigating methods of periodic removal of the codeposits (principally baking in an oxygen environment) from those limited areas, and by pursuit of methods to desorb the vacuum chamber wall generally (e.g. by use of the radiation flash from mitigated disruptions). Lastly NSTX is exploring the more speculative possibility of switching to a liquid lithium divertor target, which would be resistant to disruption damage, and could in principle be cleansed of tritium outside the device. The peak divertor heat flux on NSTX can equal that predicted for normal ITER operation, providing an excellent test-bed for such techniques. Key questions to be answered in these tests include the heat flux capability and the MHD stability of the liquid during ELM's and disruptions.

There are also questions of what role PFCs and wall conditioning (e.g. boronization) play in affecting core and divertor performance. Boronization was introduced at least a dozen years ago and is now applied to essentially all magnetic fusion research devices. Its application is generally found to improve plasma operation although the reasons are not well understood. However, it is not yet clear if boronization can be applied in a meaningful way in ITER, casting doubt on achieving the best confinement in next-step devices. DIII-D and NSTX are exploring this for boron applied over carbon plasma facing components while C-Mod is providing the equivalent experience with high-Z. NSTX plans also to investigate the effects of lithium-coating on carbon. All three facilities have programs in place to characterize dust formation and transport.

Lastly, the choice of PFC material affects divertor and core performance. When C is present in the plasma in sufficient quantities from the PFCs less extrinsic impurities are needed to increase divertor radiation so as to dissipate the core power efflux before it reaches the divertor plates. C-Mod is required to add low-Z impurities to increase the divertor radiation. In all cases the question is how to control the level of such impurities that are needed for power dissipation without adversely diluting the core reactivity and increasing radiation there. The three experiments are exploring this question and the effects of their different divertor geometries and materials, thus providing complementary approaches to solving this problem in ways applicable to ITER.

3. The physics of plasma rotation and momentum transport

The rotational stabilization of the RWM sparked increased interest in the physics of why plasmas spin toroidally even without apparent applied torque. In the conventional picture, the toroidal rotation is the result of toroidal torque from tangential neutral beams balancing a momentum loss through diffusion. Because ITER has high energy negative ion beams with low momentum content, this conventional picture predicts rotation speeds in ITER below but possibly near the threshold value for RWM stabilization. However the underlying physics has not been fully clarified. Hence it is important to study RWM stabilization with and without rotation and to understand the connection of rotation damping to the plasma sound and Alfvén speeds.

Other motivations for studying momentum transport and rotation include the role of rotation in producing sheared ExB flows that stabilize turbulence, the role of plasma flows in edge physics, and the role of rotation in stabilizing internal plasma modes. The discovery of strongly rotating plasmas (up to 130 km/s, Mach number 0.3) at high densities on Alcator C-Mod with no momentum input (purely ohmic and ICRF heated) shows that the conventional picture misses some important element of the physics. DIII-D has also found toroidal rotation with EC heating only. NSTX has observed strong edge flows associated directly with RF heating. If spontaneous rotation could be understood and controlled, this might allow for rotational RWM stabilization in ITER.

DIII-D has studied both toroidal and poloidal rotation because of the role rotation plays in forming the radial electric field whose shear forms both internal transport barriers and the H-mode transport barrier (pedestal) at the edge. Such flows and transport barriers are also studied in Alcator C-Mod and NSTX. From studies of the edge plasma flows, Alcator C-Mod has advanced the hypothesis that the physics on the open field lines in the SOL determines the edge rotation which sets the boundary condition for the core plasma rotation. The role of intrinsic plasma rotation in the H-mode power threshold in different magnetic configurations (with the X point towards and away from the gradB drift direction) has been demonstrated in Alcator C-Mod plasmas with no external momentum input. Edge plasma flows from drifts and ExB flows are studied for all three machines using the BOUT edge turbulence code and the UEDGE fluid code.

NSTX has strong co-rotation from its neutral beam with comparable values of Mach and Alfvén Mach numbers (due to very high beta). This capability is being utilized to study the stabilizing effect of rotation on internal MHD instabilities. The ability to measure the plasma rotation speed with spatial resolution equal to the ion gyroradius on NSTX is permitting accurate studies of the relationship of internal structures to rotation damping. This will help untangle possible electrostatic and electromagnetic mechanisms that drive core momentum diffusion. Soon DIII-D will be equipped with both co- and counter-beam injection capability, which in cooperation with the near perpendicular injection capability and ripple loss on JT-60U, should allow the physics of rotation/electric field formation from momentum input and orbit loss to be disentangled. Plasmas with no applied torque can be compared using balanced beams in DIII-D versus RF only in Alcator C-Mod, DIII-D, and NSTX. Momentum drag can be altered with the RWM coils in DIII-D and NSTX and the A-coil set in Alcator C-Mod.

The physics of rotation is important and contains many unsolved physics problems. The study of rotation has become a U.S focus area in which all three facilities combined will make important contributions.

4. Advanced Tokamak performance through off-axis non-inductive current drive.

High fusion gain plasmas with high bootstrap current fraction for steady-state operation at and above the no-wall beta limit, require broad, if not hollow, current profiles. High bootstrap operation tends naturally toward hollow current profiles. Hence the use of non-inductive current drive off-axis at large radii is crucial to achieving such plasmas. DIII-D has pioneered off-axis current drive using electron cyclotron waves. The magnetic field strength in DIII-D is well suited to the use of gyrotrons for EC power. The basic physics of the off-axis ECCD process has been verified. Initial experiments have shown the ability to control the current profile in plasma operating above the no-wall beta limit. Upgrades to the long pulse EC power are underway to control the current profile on current diffusion time scales in high beta, high bootstrap fraction plasmas. Stationary hybrid mode plasmas for ITER have been operated for nine current diffusion times.

At the high magnetic field in Alcator C-Mod, the optimal off-axis current drive technique is lower hybrid current drive (LHCD). Also, because LH waves damp efficiently at relatively low electron temperature, they are particularly well suited for driving current in the outer plasma. The LHCD system on Alcator C-Mod has recently become operational, and low power coupling experiments have been completed to assess launcher performance. The LHCD experiments on C-Mod will be performed at magnetic field strength (5 Tesla) and plasma density (1-1.5x10^{20} m^{-3}) identical to those required for current profile control in ITER. The C-Mod facility will also provide critical data for divertor performance at SOL densities and power flows required for AT operation in ITER. Finally, the C-Mod experiments will be carried out with fully relaxed current density profiles ($\tau_{pulse} \geq \tau_{L/R}$) near the no-wall beta limit ($\beta_N \cong 3$), where AT operation is planned for ITER.

For over-dense plasmas, NSTX is developing Electron Bernstein Wave (EBW) current profile control. A unique aspect of these experiments will be use of the Ohkawa effect to drive off-axis current at low aspect ratio, where toroidal trapping effects are significant. Radiometer measurements and modeling have verified efficient emission of EBW at low multiple harmonics from the over-dense plasma core, and conversion to the X and then the O mode ECW near the upper hybrid layer usually located in the SOL. This has established a reference approach for EBW heating and current drive at the MW power level, which is planned to be implemented in the next 3 years. The theoretical efficiency of this new current drive technique is very high, and techniques of this type will be required in future high-beta, over-dense devices.

It is very important to study all of these off-axis current drive techniques because they are the key to the Advanced Tokamak and ST approaches to high gain steady-state plasmas for future fusion power systems, including advanced performance modes in ITER and a possible Component Test Facility. ITER currently has a high power ECCD system planned, and the option to add comparable amounts of LH power.

5. The role of density/collisionality in edge plasma turbulence and divertor physics.

The three U.S facilities combined afford a unique opportunity to address another crucial element of a successful ITER program, the ability of the SOL and divertor plasma to exhaust impurities and the high power produced in the ITER plasma. Central to this issue is the development of reliable physics-based prescriptions for heat and particle transport in the edge and divertor regions, which can extrapolate to ITER conditions. ITER's edge and divertor will operate with a unique combination of parameters that are not simultaneously accessible to any one device except ITER: high absolute plasma densities and temperatures at the separatrix, high parallel heat fluxes into the divertor region, and high density, low temperature (partially detached plasmas) near the divertor target plates. Therefore, the understanding of SOL/divertor physics must be built in terms of the dimensionless parameters that control it, using present experiments to collectively access a sufficiently wide parameter range. Two dimensionless parameters that are thought to influence SOL transport physics are: plasma collisionality and the transparency of the edge to neutral penetration, (i.e., neutral mean free path divided by machine size), that scales with poloidal field. Additional parameters that are believed important for divertor physics include the opacity of the divertor to Lyman-alpha radiation and neutral-neutral mean-free path compared to divertor size.

The U.S is fortunate to have facilities that cover this entire range in these parameters anticipated for ITER. The machine parameter with most leverage in this regard is plasma density, which scales roughly as plasma current divided by machine size squared. In terms of absolute density, ITER is projected to operate from the middle of the Alcator C-Mod density range down to the middle of the DIII-D and NSTX density range. Alcator C-Mod's high field and compact size allows it to explore the high density end of the spectrum, producing SOL and divertor plasmas with high collisionality and low neutral transparencies. Correspondingly, divertor regimes with high Lyman-alpha opacity and fluid-like neutrals are readily accessible in C-Mod. DIII-D and NSTX provide essential access to lower density regimes: edge collisionalities that are comparable to ITER, a SOL more transparent to neutrals, and divertor regimes with kinetic neutrals and low levels of Lyman-alpha trapping.

Some important differences in edge plasma transport behavior are detected across the US facilities, with important implications for ITER. At medium to high densities in Alcator C-Mod with a SOL that is opaque to neutral penetration, a high level of cross-field transport is evident: clumps of plasma are transported deep into the scrape-off layer; SOL profiles develop a shoulder; plasma dominantly refuels from main chamber surfaces instead of the divertor. At the high end of its density range, DIII-D sees a similar edge turbulence behavior. But at the low density range, corresponding to ITER's collisionality, the density shoulder in DIII-D is much less pronounced, the edge turbulence is much less clump-like, and the plasma refuels from the divertor. NSTX also sees very strong clump-like behavior in L-mode, but much quieter edge activity in the H-mode. The range of physics behaviors that are possibilities in ITER may be contained within the broad range of edge parameters afforded by the three facilities.

The edge plasma turbulence has a strong in-out asymmetry. In this regard, NSTX adds a unique capability since the in-out asymmetry of turbulence is maximized at low aspect ratio. All three facilities have observed the bursty, clumped plasma edge turbulence and NSTX, compared to DIII-D and Alcator C-Mod provides a test of the effects of large gyroradius (low field) as well as large in-out asymmetry. The range in B encompassed by the three devices provides a unique opportunity to understand the underlying physics of this behavior. The turbulence behavior of the plasma edge is being studied for all three machines with the BOUT code. Particular challenges to the physics in this code are posed by ExB drifts (DIII-D), edge plasma flows connected to the grad-B drift direction (Alcator C-Mod), and the large in-out asymmetry at low aspect ratio (NSTX). The uniquely extended parameter range afforded by these three facilities, integrated through modeling, creates a unique opportunity for the United States to contribute to projecting edge physics to ITER.

6. The physics of energetic particles and Alfvén eigenmodes.

The physics of energetic particle-wave interactions is an area where the three facilities have complementary capabilities. Between them, theoretical models of Alfvén eigenmode damping and drive mechanisms can be tested under a wide range of conditions and with a wide range of diagnostics, to provide a better understanding of energetic particle-wave interactions to extrapolate to a burning plasma.

Gaps in the Alfvén continuum, and hence the structure, spectrum and damping of Alfvén eigenmodes, are predicted to depend strongly on the plasma shape, including the aspect ratio. DIII-D and C-Mod have aspect ratios similar to ITER while NSTX, with a/R almost twice as large, provides a test of the predicted aspect ratio dependence, and thence the underlying physics. The tighter aspect ratio in NSTX also creates a larger population of trapped particles, facilitating the study of energetic particle modes driven by a resonance with the trapped particle bounce frequency. Through similarity experiments on the three devices with the same plasma shape, except aspect ratio, measured differences in the drive and damping could isolate the underlying physics.

Models for the fast ion driving term can be tested with the varying fast ion velocity distributions and ratios of fast ion velocity to Alfvén velocity that are found in the three machines. Due to the low magnetic field, the neutral beam ions in NSTX are super-Alfvénic, with ratios of beam ion velocity to Alfvén velocity between 1 and 4. Strong ICRF heating in C-Mod, provides super-Alfvénic ions with a velocity distribution quite different from the neutral beam slowing-down spectrum and with equilibrated thermal ion and electron temperatures. In DIII-D, variation of the toroidal field can make the neutral beam ions sub- or super-Alfvénic: V_f/V_A can vary from about 0.3 to 2, the latter at reduced magnetic field. The new capability of mixed co and counter beam injection in DIII-D provides variation in the velocity distribution. RF heating capabilities in DIII-D and NSTX also provide additional flexibility in modifying the fast ion distribution, including acceleration of beam-injected ions. The different heating schemes also allow tests of the dependence of the Alfvén mode spectrum and stability on plasma rotation and rotational shear. Beam-heated plasmas in NSTX rotate strongly, up to a significant fraction of the Alfvén velocity, while rf-heated plasmas in C-Mod have lower rotation, as expected in ITER. Beam-driven rotation in DIII-D provides an intermediate case or moderate rotation, and mixed co and counter beam injection in DIII-D will allow variation down to near-zero rotation.

The three machines also have complementary diagnostics. C-Mod has the key capability to study Alfvén eigenmodes with active MHD spectroscopy; this allows detailed measurements of the frequency and damping rate of stable modes, and thus separate measurements of the driving and damping terms. A wide range of core fluctuation diagnostics including phase contrast imaging (C-Mod), reflectometry (NSTX, C-Mod and DIII-D), beam emission spectroscopy (C-Mod and DIII-D), laser interferometry and far infrared scattering (DIII-D), and far infrared tangential interferometry and polarimetry as well as imaging microwave reflectometry (NSTX) can be used to investigate the newly discovered core-localized Alfvén modes. DIII-D and NSTX's capabilities for measurement of the q-profile (motional Stark effect) and the fast ion distribution (D-alpha spectroscopy and scanning neutral particle analyzer, respectively) allow study of the q-profile dependence of mode stability and the effects of instabilities on fast ion transport. The MSE system on NSTX enables detailed measurements of the q-profile in regimes when the beam ions are super-Alfvénic.

The complementary characteristics of the three machines are already being exploited: comparisons of toroidal Alfvén eigenmodes and Alfvén cascades in all three devices have indicated similarities in the drive mechanisms but with differences in the detailed time evolutions of the mode structures. Such comparisons between machines, under similar and widely varying conditions and with a broad range of diagnostics, provide a much more powerful confirmation of theories than would be possible in one machine alone.

7. The dependence of transport on beta

There is substantial theoretical and programmatic interest in the dependence of turbulent transport on β. The standard ITER confinement scalings imply confinement degrades like $1/\beta$, a scaling that severely punishes the thrust toward higher beta plasmas. On the other hand, carefully constructed dimensionless parameter scaling experiments between DIII-D and JET showed confinement was independent of beta. Recent re-analysis of the ITER confinement database has increased the uncertainty in the β-exponent of the confinement time scaling expression, due to the relatively larger errors in the dependence on the heating power inferred from data analysis compared to other plasma and machine parameters. More recently, NSTX confinement time data across a wide range of β_T values (10-30%) have been accumulated and analyzed. Preliminary results suggest a positive β-exponent in the confinement time scaling. Zero or positive β-dependence in the confinement time scaling may allow significant performance improvement in the higher beta-N operation scenarios estimated for the ITER hybrid and steady-state modes.

Nonlinear turbulence modeling has not advanced sufficiently to resolve these questions, as the electromagnetic effects are expected to emerge as β is increased. The combined contributions of Alcator C-Mod, DIII-D, and NSTX can significantly expand the range of the ITER confinement database in field (0.3-8T), beta (1-40%), and aspect ratio (1.3-3.0). This indicates a unique capability of the combined U.S. facilities to reduce this uncertainty in the beta-exponent, through a systematic database analysis, that accounts for the errors in the variables. These three machines are also well positioned to perform dimensionless parameter scaling experiments varying magnetic field and size at the same shape and aspect ratio (Alcator C-Mod and DIII-D), varying aspect ratio and field at the same shape and size (DIII-D and NSTX), and varying aspect ratio and field at the same shape but different size (Alcator C-Mod and NSTX). The results could shed additional light on the effects of the increasing electromagnetic effects, due to the increasing beta, on the underlying turbulence and transport mechanisms, an important scientific issue for all high beta confinement configurations.

Appendix 4. FESAC Facilities Panel June Meeting at the U.S. Naval Research Laboratory (June 13-17, 2005)

All members of the FESAC Facilities Panel (see page 2 of this report) met for five days to hear testimony from representatives of each of the three major U.S. toroidal magnetic fusion facilities, and to deliberate about panel findings and responses. The meeting agenda is reproduced in A4.1 below. The list of meeting attendees is provided in A4.2.

A4.1. FESAC Facilities Panel Meeting Presentations

Monday, June 13:

- Executive Session [08:45 – 09:45]
- Alcator C-Mod Presentations, and Q&A [10:15 – 18:00]
  - Introduction, *E. Marmar*
  - ITER/ Burning Plasma Supporting Research, *M. Greenwald*
  - Wave-Plasma and Advanced Tokamak, *A. Hubbard*
  - Transport, *M. Greenwald*
  - Plasma Boundary, *B. Lipschultz*
  - Macroscopic Stability, *E. Marmar*
  - Facility Capabilities, *E. Marmar*
  - Summary, *E. Marmar*

Tuesday, June 14:

- Executive Session [08:45 – 09:45]
- DIII-D Presentations, and Q&A [10:15 – 18:30]
  - DIII-D Program, Staff and Facility, *R. Stambaugh*
  - University Research Using DIII-D, *E. Doyle*
  - Advanced Tokamak and ITER Research, *M. Wade*
  - Transport Research, *K. Burrell*
  - Stability Research, *E. Strait*
  - Alfven Mode Research, *R. Nazikian*
  - Pedestal and Divertor Research, *M. Fenstermacher*
  - Answers to the [Charge] Questions – Summary, *T. Taylor*

Wednesday, June 15:

- Executive Session [08:45 – 09:45]
- NSTX Presentations, and Q&A [10:00 – 15:15]
  - Macroscopic Plasma Physics and Solenoid-free Start-up + HHFW and EBW, *J. Menard*
  - Multi-scale Transport Physics, *S. Kaye*
  - Energetic Particles, *D. Gates / N. Gorelenkov*
  - Plasma Boundary Interfaces, *J. Boedo*
  - Long-term Vision of ST as an Option for CTF, *M. Peng*
- General Discussion [15:15 – 15:45]
- NSTX Presentations, and Q&A, continued [15:45 – 18:30]

Science Overview, *E. Synakowski*
Facility Diagnostic Capabilities, *M. Ono*

Thursday, June 16: Executive Session [08:45 – 18:00]
Friday, June 17: Executive Session [08:45 – 18:00]

A4.2. FESAC Facilities Panel Meeting Attendees

U.S. Department of Energy Attendees:

Erol Oktay, DOE (June 13-15)
Stephen Eckstrand, DOE (June 13-15)
Adam Rosenberg, DOE (June 13-15)

Alcator C-Mod Attendees:

Earl Marmar, MIT (June 13-15)
Martin Greenwald, MIT (June 13-15)
Miklos Porkolab, MIT (June 13-15)
Ronald Parker, MIT (June 13)
Amanda Hubbard, MIT (June 13)
Steven Scott, PPPL (June 13)
Robert Granetz, MIT (June 13)
Paul Bonoli, MIT (June 13)
Steven Wukitch, MIT (June 13)
Bruce Lipschultz, MIT (June 13)

DIII-D Attendees:

David Baldwin, GA (June 13-15)
Ronald Stambaugh, GA (June 13-15)
Tony Taylor, GA (June 13-15)
Vincent Chan, GA (June 14)
Mickey Wade, GA (June 14)
Keith Burrell, GA (June 14)
Edward Strait, GA (June 14)
Max Fenstermacher, GA (June 14)
Raffi Nazikian, PPPL (June 14)
Edward Doyle, UCLA (June 14)

NSTX Attendees:

Rob Goldston, PPPL (June 13-15)
Richard Hawryluk, PPPL (June 15)
Edmund Synakowski, PPPL (June 13-15)
Martin Peng, ORNL (June 13-15)
Masayuki Ono, PPPL (June 15)
Stan Kaye, PPPL (June 15)
Jon Menard, PPPL (June 15)
Jose Boedo, UCSD (June 15)
David Gates, PPPL (June 15)
Stewart Zweben, PPPL (June 15)

Appendix 5. List of Terms and Acronyms

Advanced Tokamak (AT) Mode: An integrated-system MHD-stable operating scenario for a tokamak that has high self-driven (bootstrap) current, good particle and energy confinement, and a plasma edge that enables particle and power handling.

Alcator C-Mod: A compact high-field tokamak experiment, also denoted C-Mod.

Alfvén wave: A plasma wave, which involves bending or compression of the magnetic field. The Alfvén time scale is the interval for an Alfven wave to traverse the plasma.

Alpha heating: In a fusion power plant, energetic alpha particles and neutrons are created by the fusing of deuterium and tritium nuclei. As a charged particle, the alpha particle is unable to cross the confining magnetic field and gives up its energy as heat to the plasma.

Aspect ratio: In a toroidal device, the ratio of the major radius to the minor radius.

Ballooning mode: A pressure-driven MHD instability that "balloons" out with larger amplitude in the larger major radius (lower toroidal field) portion of a toroidal plasma, and has smaller amplitude at smaller major radius.

Beta: The dimensionless ratio of plasma pressure to magnetic pressure. Often denoted by the symbol β.

Bootstrap current: Currents driven by collisional transport effects in toroidal plasmas.

Confinement: Property of magnetic fields in preventing loss of energy and particles; degree or measure of this property, e.g., as in 'confinement time.'

Detached plasma: Cold dense plasma in the divertor chamber, separated from the walls by neutral gas. This is characteristic of a particular mode of divertor operation.

DIII-D: A medium scale, strongly shaped tokamak experiment.

Disruption: The abrupt termination of a tokamak plasma through the growth of large amplitude MHD instabilities. Control of disruptions is a critical problem for this type of confinement device.

Divertor: Region outside the plasma core with open field lines leading to a chamber some distance from the plasma.

DOE: U.S. Department of Energy

ECRF: Electron Cyclotron Range of Frequencies – refers to radio frequency heating and current drive using waves close to the electron cyclotron frequency.

Edge: Plasma region between the hot core plasma and the walls.

ELM: Edge Localized Mode – an MHD mode that relaxes the pressure gradient of a plasma device operating in H or High confinement mode.

Equilibrium: Usually refers to MHD equilibrium – a steady state solution of the MHD equations.

ETG: Electron Thermal Gradient (modes) – a 'micro' plasma instability that may be responsible for turbulent transport by very short wavelength (electron cyclotron radius) fluctuations.

FESAC: Fusion Energy Sciences Advisory Committee – Federal panel that provides independent advice to the Director of the DOE Office of Science on the scientific and technological issues that arise in the planning, implementation, and management of the OFES program.

Fusion gain (Q): The ratio of fusion power produced in a system to the amount of externally applied plasma heating power.

GA: General Atomics – a privately held research and development corporation in San Diego, CA; the site of DIII-D.

Gyroradius: The radius at which a particle gyrates about the magnetic field.

H-Mode: High (confinement) Mode – an experimental regime with a transport barrier or pedestal at the edge of the plasma.

IBW: Ion Bernstein Wave – short wavelength electrostatic waves in the ion cyclotron range of frequencies. Generated in the plasma by mode conversion.

ICRF: Ion Cyclotron Range of Frequencies – refers to radio frequency heating, current and flow drive using waves close to the ion cyclotron frequency.

ITB: Internal Transport Barrier – a regime of strongly reduced transport over some part of the core plasma.

ITER: A proposed international experimental fusion reactor based on the tokamak concept.

ITPA: International Tokamak Physics Activity – an activity that fosters corporative research that aims to advance the understanding of the physics of burning plasmas.

JET: Joint European Torus – a large tokamak experiment run jointly by a European Community consortium.

Kink Instability: A macroscopic (large scale) instability driven by plasma currents.

LH: Lower Hybrid – RF waves of frequency intermediate between the electron and ion gryofrequencies and used for heating and current drive.

L-Mode: Low (confinement Mode) – the baseline for confinement in magnetic fusion devices, dominated by strong turbulent transport.

Magnetic flux: The integral over a surface area of the magnetic field perpendicular to the surface. Examples are poloidal flux and toroidal flux.

Major radius: Distance from the center-line of the torus to the magnetic axis or center of the plasma cross section.

MHD: MagnetoHydroDynamics – a fluid description of magnetized plasmas.

MIT: Massachusetts Institute of Technology – a major university in Cambridge, MA; the site of C-Mod.

Minor radius: Distance from the center of the plasma cross section to the plasma edge.

Mode: (1) a plasma wave that has, at least approximately, a constant wavenumber and frequency or can otherwise be characterized as a normal mode of some wave equation; (2) a mode of operation of a device, such as AT-mode or H-mode.

ν^*e: Pronounced "nu-star-E" – electron collisionality parameter.

NBI: Neutral Beam Injection – a common method of plasma heating which employs intense beams of neutral atoms.

Neoclassical: The theory of collisional transport in toroidal geometry.

Normalized beta: A characterization of plasma beta (multiplied by toroidal field and minor radius, divided by plasma current) typically used to express the stabilization quality of a system. Initial high beta tokamak experiments reached normalized beta values of approximately 3.

NSTX: National Spherical Torus Experiment – an experimental fusion device based on the spherical tokamak concept.

NTM: Neoclassical Tearing Modes – resistive MHD modes driven unstable by bootstrap currents.

OFES: Office of Fusion Energy Sciences – the DOE Office of Science program office that sponsors research at C-Mod, DIII-D, and NSTX.

Ohmic: Characteristic of dissipation due to plasma resistivity; Ohmic heating, in which a current is driven inductively in the plasma.

Pedestal: Region of steep gradients at the plasma edge.

Poloidal: The short way around in a torus.

PPPL: Princeton Plasma Physics Laboratory – a U.S. DOE fusion energy and plasma physics research laboratory in Princeton, NJ; the site of NSTX.

RF: Radio Frequency – refers to methods of heating, flow or current drive depending on the interaction of plasmas with externally launched waves.

ρ*: Pronounced 'rho-star', the ratio of the ion gyroradius to the minor radius.

RWM: Resistive Wall Mode – an MHD mode driven unstable by finite wall resistivity.

Safety factor (q): The ratio of the number of times a magnetic field line wraps toroidally (the long way around a toroidal device) to the number of times it wraps poloidally (the short way around). Values less than one typically suffer internal instability, and larger values are typically more stable.

Sawtooth: a cyclical minor disruption in which core field lines reconnect flattening the temperature profile, followed by a slow reheat of the central plasma.

Separatrix: The boundary surface between regions of a plasma on closed and open field lines.

SOL: Scrape-Off Layer – region of plasma existing on open field lines outside of a separatrix.

Tearing modes: Resistive MHD modes driven by plasma currents.

Tokamak: A magnetic confinement device relying on a large toroidal current carried by the plasma itself. (The acronym is from the Russian phrase Toroidalnaya Kamera I Magnitaya Katushka, meaning toroidal chamber and magnetic coil.)

Transport Barrier: Region of strongly reduced transport – typically a bifurcated state.

Transport flux: The total energy or particle flow per unit area passing through a surface element.

Zonal flows: Flows that are constant on flux surfaces but with strong radial variation. These flows are self-generated by plasma turbulence for which it is also an important non-linear saturation mechanism.