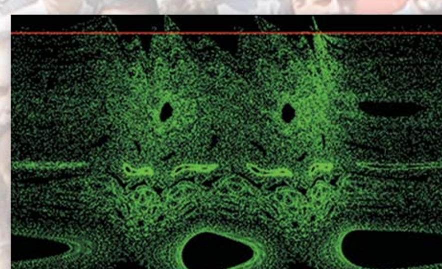
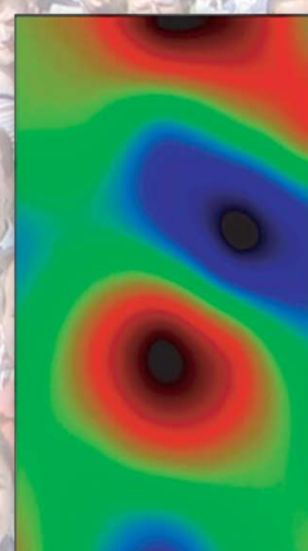
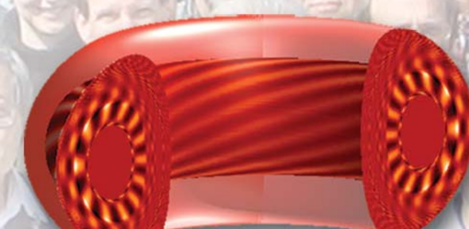
GA-A25889

THE DIII–D NATIONAL FUSION PROGRAM FIVE-YEAR PLAN 2009–2013







April 2008

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GENERAL ATOMICS

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by
PROJECT STAFF

Work prepared under
Department of Energy Contract
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GENERAL ATOMICS PROJECT 30200
APRIL 2008

DISCLAIMER

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1. THE DIII-D FIVE-YEAR PROGRAM PLAN

This document presents the DIII-D Program Plan for the period 2009–2013. The Plan was developed by the DIII-D national collaborative team, responsible for fusion energy science research using the DIII-D facility.

1.1. THE DIII-D MISSION AND GOALS

The ultimate goal of fusion research is an attractive, long-term source of energy. The National Research Council (NRC) Burning Plasma Report states "Fusion energy holds the promise of providing a significant part of the world's long-term, environmentally acceptable energy supply." Fusion fuels, deuterium and tritium, may be obtained from sources as common and abundant as sea water and lithium from the earth's crust. There are no greenhouse gas emissions from fusion, and the radioactive wastes from fusion are short-lived, potentially only requiring shallow land burial and oversight for about 100 years. Successful development of fusion energy would greatly reduce concerns over imported oil, pollution, global warming, and would help to meet the energy needs of all mankind for centuries. Developing the scientific basis for fusion energy for future generations represents a grand challenge for science and offers the prospect of great benefits for all mankind.

Fusion Research is carried out in the United States under the sponsorship of the Department of Energy (DOE) Office of Fusion Energy Sciences (OFES). The mission of the Fusion Energy Sciences (FES) program is *to advance plasma science, fusion science and fusion technology – the knowledge base needed for an economically and environmentally attractive fusion energy source*. The DIII-D National Fusion Facility is the largest magnetic fusion facility in the U.S. and is the key participant in developing the knowledge base for fusion energy production. Strongly supporting the mission of the national fusion program, the mission of the DIII-D National Fusion Program is

To establish the scientific basis for the optimization of the tokamak approach to fusion energy production.

Working with the DIII-D international team, the DIII-D program as illustrated in Fig. 1-1 has defined three main research themes:

1. Enable the success of ITER by providing physics solutions to key issues.
2. Establish the physics basis for steady-state high performance operation for ITER and beyond.
3. Advance fundamental understanding of fusion plasmas along a broad front.

ITER Physics. ITER is the key breakthrough experiment for magnetic fusion energy. The purpose of the ITER Project is to "demonstrate the scientific and technological feasibility of fusion energy for peaceful purposes." ITER is a Presidential initiative and its success is critical for fusion energy to move forward in the U.S. and in the world. ITER is an international fusion research project designed to produce, control, and sustain a burning plasma, a plasma whose high temperature is maintained largely by self-heating with the alpha particles produced by the fusion reactions in the plasma. The success of ITER is the highest priority of the DIII-D program. DIII-D is approximately one-fourth the size of ITER and produces plasmas with ITER relevant temperatures, collisionality, and normalized pressure (or beta). In addition, DIII-D is equipped with several control tools that are envisioned for ITER operation. Because of these similarities in physics regimes and control capabilities, DIII-D has provided significant information for the design and operation of ITER.

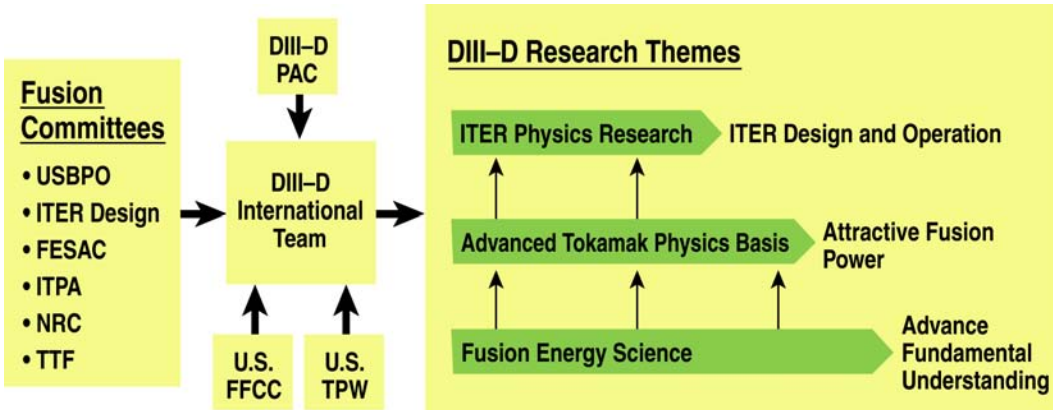

Fig. 1-1. The DIII-D Research Plan was formulated within the context of the several DOE committee recommendations and DIII-D's unique worldwide capabilities. The DIII-D Program Advisory Committee (PAC) provides advice on the long-term program plan and provides annual assessment of experimental plans. The Fusion Facilities Coordinating Committee assists in coordinating plans amongst the three U.S. facilities. The plan was presented and discussed in an open Tokamak Planning Workshop (TPW). DIII-D aims to assure ITER success, establish and optimize an attractive fusion concept and advance fundamental understanding of fusion plasmas.

In the next five years, the DIII-D program plan will contribute to the success of ITER in five important areas:

1. Provide the physics basis for key ITER design issues.
2. Develop and validate integrated operational scenarios that meet ITER physics objectives.
3. Develop a predictive understanding of issues key to ITER's performance and optimization.
4. Develop the physics basis for steady-state high performance scenarios that will enhance ITER's research capabilities.
5. Provide a state-of-the-art research platform that prepares next-generation U.S. fusion scientists for ITER operation and leadership roles in ITER research.

The DIII-D program will be an essential participant in the U.S. Burning Plasma Organization (USBPO) on ITER relevant research. In addition to this national coordination, the DIII-D program will coordinate its research activities with the international community through the International Tokamak Physics Activity (ITPA), consisting of scientists from the seven partners of ITER (European Union, Japan, United States, Russia, Korea, China, and India). Figure 1-2 illustrates the U.S. research agenda for ITER for the next 10 years, as developed by the USBPO in response to the Energy Policy Act of 2006. The planned DIII-D contributions are extensive.

Fig. 1-2. Areas where the DIII-D plan will make major contributions to the ITER Research are indicated by two checks (√√), and significant contributions by one check (√). This table is the first 10 years from the U.S. research agenda for ITER as developed by the USBPO in response to the Energy Policy Act of 2006.

Advanced Tokamak (AT). Realizing the ultimate potential of the tokamak as a magnetic confinement device through the demonstration of the feasibility of steady-state, high performance tokamak operation is a high level objective of the DIII-D program. The DIII-D program emphasizes AT research, both for ITER and for future magnetic fusion research. The aim of research in this area is to develop the physics basis for steady-state high performance plasmas and to realize the ultimate potential of the tokamak as a magnetic confinement device. DIII-D is the world leader in the development of the AT concept, which utilizes increases in confinement and stability with noninductive current drive to achieve steady-state operation (high duty cycle) near the ideal stability limit (optimal performance). The DIII-D program initiated AT research, which is now being pursued by all the world's tokamak research programs. AT concepts pioneered on DIII-D are now key elements in new tokamaks now coming on line or in the planning stages.

The DIII-D program remains a world leader in nearly every aspect of steady-state high performance operation. Research in this area will provide the basis for steady state operation and the basis for proceeding with a fusion development facility (or component test facility), and then to demonstration power plant (DEMO). This research includes integrating different elements of the plasma core (transport, magnetohydrodynamic (MHD) stability, current drive) and integrating a high performance core with a plasma boundary that adequately controls particles and heat. Developing, understanding, and controlling these plasmas are exciting scientific challenges and will provide a compelling approach for attractive fusion energy.

The DIII-D program is uniquely suited to make strong contributions to steady-state high performance research by virtue of its comprehensive set of plasma diagnostics provided by an international team of fusion scientists, its operational flexibility (shaping, heating, particle control, and current drive), and its highly capable plasma control system. With these capabilities, DIII-D scientists are learning to access and control steady state discharges to make detailed physics measurements in order to validate theory and develop models that can extrapolate this performance to ITER and DEMO. In particular, the DIII-D program leads the world research effort in the understanding and control of MHD instabilities. The DIII-D Program Plan will contribute strongly to developing advanced scenarios in support of ITER, Fusion Development Facility/Component Test Facility (FDF/CTF) and DEMO: Fig. 1-3 indicates the broad steps we will follow for each case. Steady-state scenarios suitable for ITER are near at hand. A FDF or CTF aims to enable development of fusion's energy applications in preparation for DEMO. With additional off axis current drive [off-axis neutral beam injection (NBI) and electron cyclotron current drive (ECCD)] ITER relevant steady state scenarios with low torque input, and $T_e \sim T_i$, can be evaluated, and suitable boundary solutions can be evaluated. For FDF and DEMO, we will be evaluating and optimizing control for high values of beta that the double null shape affords. A key part of the advanced scenario development will be the integration of the boundary with the high performance core.

Fusion Science. Strong emphasis on fusion science lies behind the success of the DIII-D program, as it seeks to "advance fundamental understanding of fusion plasmas along a broad front." The key strength of the DIII-D program is the capability to carry out outstanding scientific investigations of cutting edge theory, which in turn enable optimization of the tokamak concept through improved understanding of the key processes. DIII-D research in support of ITER and AT development are built on excellent scientific research, as evidenced by the most recent (2007) John Dawson Award for Excellence in Plasma Physics Research going to a multi-institutional team for research carried out on DIII-D. This award is the fourth such award for the DIII-D program. This Five Year Plan provides growing opportunities to conduct leading-edge scientific research. Over the next five years, we plan to invest significant effort and experimental time to validate theories and models aimed towards developing a predictive understanding of the basic processes in fusion plasmas.

The program plan includes specific investigations in five scientific areas; transport, MHD stability, energetic particles, heating and current drive, and boundary physics, as indicated in Fig. 1-4 There has been tremendous recent progress in recent years in turbulence simulations to predict transport in

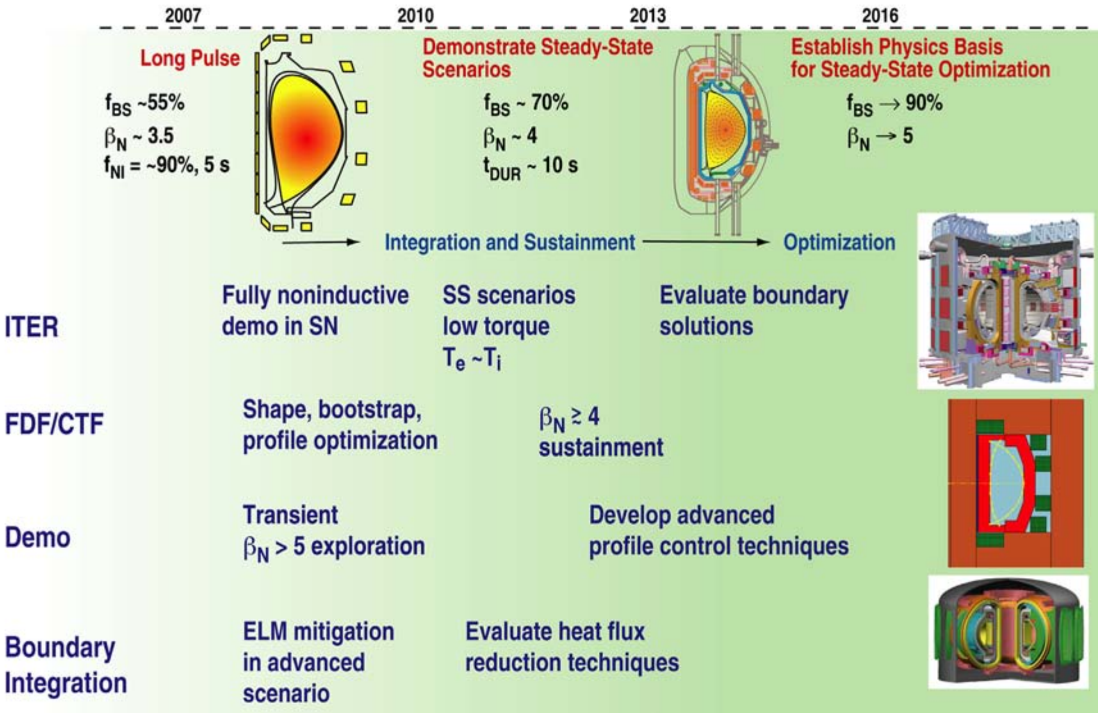

Fig. 1-3. DIII-D is in position to contribute strongly to steady-state scenario development for ITER, a FDF, and DEMO.

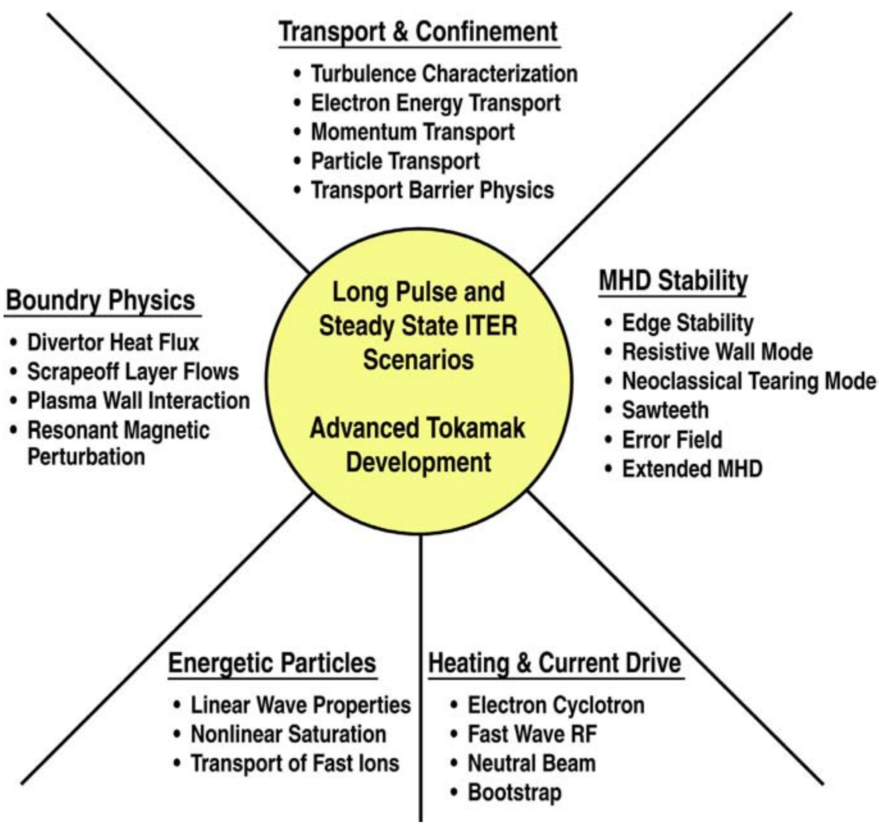

Fig. 1-4. DIII-D carries out comprehensive research on fundamental fusion science topics which also supports ITER research needs and forms the foundation for steady-state AT development.

tokamaks. Over the next five years, the DIII-D program will develop specific experimental tests to validate these transport models, using the capabilities of the DIII-D facility, including improved measurements. In the area of MHD stability, the research focus will move to the evaluation of nonlinear, nonideal MHD, and the exploitation of MHD control developed in recent years. Energetic particles and energetic particle driven instabilities offer a new challenge for burning plasmas such as ITER. Significant progress will be made on DIII-D on the theory and models needed for ITER plasmas using fast particles from NBI and ion cyclotron radio frequency (ICRF). Although much progress has been made in the validation of ECCD theory, detailed models of the neutral beam current drive (NBCD), especially off-axis current drive, remain to be developed. The boundary physics area will focus on accurate measurements in the pedestal and scrape-off region and comparison to two and three dimensional code models, including toroidal and poloidal flows. Particle control and power exhaust remain an important challenge for future high power density facilities. In addition to these five areas, the integration of the elements of steady-state high performance operation, provides a significant scientific challenge. The upgrades and research plan proposed here should enable DIII-D to provide the science base for access, sustainment, and control of these high performance scenarios which is a key strength of the DIII-D program.

The DIII-D program and facility have several key components required for scientific excellence and will work diligently to preserve and build on these:

1. Broad community participation with an excellent well established international team (universities, national laboratories, industry).
2. Access to fusion relevant parameters; temperature, collisionality, beta, time scales.
3. Tokamak access; hands on maintenance allows ready modification and diagnostic innovation; large number of ports provide access for diagnostics and heating and current drive systems.
4. Device flexibility: shaping, heating and current drive systems, nonaxisymmetric coils …
5. Capability to modify and add new systems.
6. Advanced and flexible digital plasma control.
7. Comprehensive state-of-the-art diagnostic set (world leading).
8. Strong coupling between theory/simulation and experiment.

Integration of Program Elements. The DIII-D research program is a strong science program aimed at the energy goal (Fig. 1-5). The program is built on sound scientific investigations of fundamental processes in fusion. A close interaction with theory and simulation leads to innovative and cutting edge research. This interaction is enabled by a comprehensive set of plasma control tools that allow the isolation of key parameters identified by the theory, enabling elucidation of the important physical processes. Modeling and simulation tools then provide the framework to fully test and validate these theories. The synergy of the basic scientific research and support for ITER R&D within the DIII-D program then provides a compelling basis for ensuring the success of this most important burning plasma experiment. Tokamak optimization toward integrated steady-state high performance prepares us to take best advantage of ITER and provides the basis for FDF/CTF and DEMO; leading to attractive fusion energy.

The DIII-D National Fusion Program is carried out by the DIII-D International Research Team on the DIII-D National Fusion Facility with strong coupling and outreach to other fusion experiments and to the U.S. fusion theory efforts. The DIII-D National Fusion Facility is located at the General Atomics, La Jolla, California, and is the largest magnetic fusion facility in the United States. The DIII-D tokamak is a medium-sized tokamak that can access fusion relevant parameters, but maintains hands-on access and the flexibility of a university experiment. The capability to easily modify the device has led to innovative solutions to fusion challenges such as active divertors, current drive systems, and nonaxisymmetric coils for resistive wall mode stabilization and ELM suppression. The plasma measurement system, with enormous contributions from universities, is unsurpassed in the world. The facility is equipped with powerful and precise heating and current drive systems, particle control systems, and a world leading digital plasma control system.

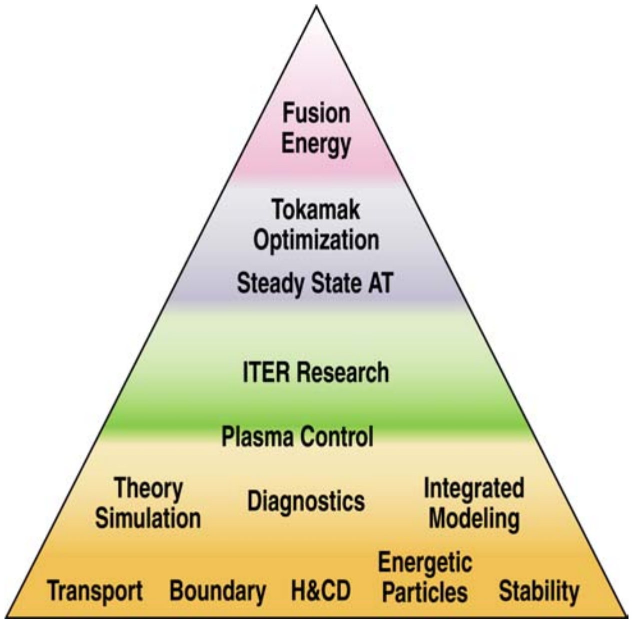

Fig. 1-5. DIII-D: A science program with an energy goal.

1.2. THE DIII-D TEAM

The DIII-D International Research Team is key to the scientific excellence of the DIII-D program. The DIII-D program is operated by General Atomics for the U.S. Department of Energy's Fusion Energy Sciences Program, but the management and program leadership is drawn from the broader DIII-D team. The DIII-D Deputy Program Director is an employee of a collaborating institution. The International team participates in program planning, proposes experiments, leads DIII-D experiments, and presents and publishes important results. DIII-D team members, including many collaborators are very heavily engaged in the USBPO, and in the International Tokamak Physics Activity. The DIII-D National Research Team includes professors, senior researchers, post-doctoral fellows and graduate students. There are more than 400 users of the DIII-D facility, 355 authors on DIII-D papers in 2007. These team members are from 92 institutions (Fig. 1-6).

- 41 Universities (26 domestic, 6 Europe, 2 Japan, 2 Russia, 2 Canada, 2 Korea, 1 Australia)
- 36 National Laboratories (7 domestic, 17 Europe, 5 Russia, 2 Japan, 2 China, 1 India, 1 Korea, 1 Kazakhstan)
- 15 Industries

The DIII-D staff has been recognized for its outstanding research accomplishments: 55 DIII-D staff are Fellows of the American Physical Society (APS), and 11 are recipients of the APS Excellence in Plasma Physics Research Award. The national character of research excellence is evidenced by the 2007 Excellence in Plasma Physics Research Award, where three of the four recipients were from collaborating institutions. Presently (2008) there are 17 graduate students pursuing their doctoral thesis work on DIII-D.

The DIII-D Program Plan includes an increase in the number of post-docs and graduate students on DIII-D. The excellent DIII-D team, the strong coupling to the GA theory group, excellent integration of all staff into the planning and execution of experiments, and a broad support structure provides an outstanding environment for graduate work. In addition, the DIII-D facility is accessible for students and faculty and can easily accommodate innovate ideas for diagnostics and hardware.

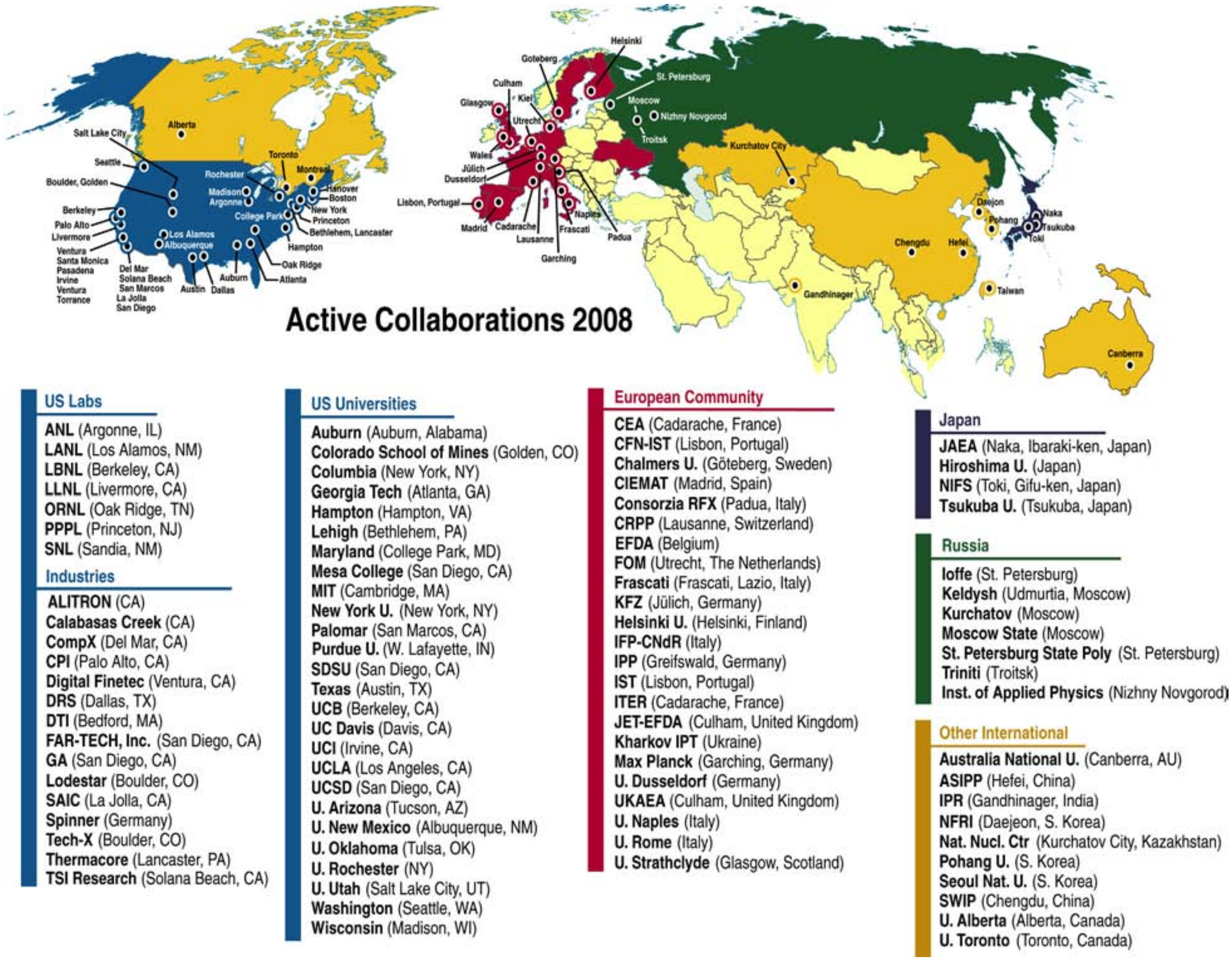

Fig. 1-6. The DIII-D National and International Team is the key to scientific excellence of the DIII-D Program.

The DIII-D research program is managed by an effective and inclusive system of governance. The primary governing body is the DIII-D Executive Committee (DEC), which advises the DIII-D Program Director and Deputy Director on matters of program planning, direction, budgets, and institutional issues. The DEC is comprised of the DIII-D division directors, and leaders from the major collaborating laboratories, and major collaborating universities. The DIII-D Program Advisory Committee, consisting of leaders and technical experts from other national and international fusion programs, provides advice annually on the program plans and other major programmatic issues. The Research Council, headed by the DIII-D Deputy Director (presently filled by an employee of Lawrence Livermore National Laboratory), provides specific advise on the annual experimental plan and priority of experimental efforts. After major research emphases are chosen, experimental proposals are solicited from the entire

International DIII-D Team at the Research Opportunities Forum. These proposals are discussed and further developed and prioritized in open meetings. Task Force Leaders and the standing physics area leaders present final research plans to the Research Council, and the Research Council provides advise on the program balance for the year.

Strong linkages between the DIII-D experimental program and theory/simulation (both at GA and at collaborating institutions) greatly enrich the DIII-D program and contribute to the scientific excellence of the program. The broad participation in the DIII-D program provides a broad science base and allows access to expertise not always available on-site, increasing the effectives and efficiency of carrying out the research program. The DIII-D program is highly interactive with other fusion programs both domestically and internationally. We assist in plasma control on domestic facilities and on Experimental Advanced Superconducting Tokamak (EAST) in China and Korean Superconducting Tokamak Advanced Research (KSTAR). A large number of international scientist participate in the DIII-D program yearly and DIII-D scientist participate in foreign experiments, many of these joint experiments are facilitated by the International Energy Agency (IEA), and ITPA.

1.3. DIII-D FACILITY CAPABILITIES AND PROPOSED IMPROVEMENTS

The DIII-D facility is well positioned to provide critical input for ITER's success, develop and advance steady-state high performance scenarios, and advance the fundamental understanding of fusion plasmas. To achieve our goals for the next five years, upgrades and modification to the heating and current drive systems, nonaxisymmetric coil systems, vessel heat removal capability, and diagnostics are planned. The evolution of the DIII-D facility capabilities to accomplish the planned research is shown in Fig. 1-7. The physics issues addressed by these facility improvements are listed in Table 1-1.

Neutral Beam Systems. DIII-D is equipped with four neutral beam lines with two ion sources each, for a total of eight ion sources. Presently in operation, there are five co-injected sources (12.5 MW) and two counter-injected sources (5 MW). Early in this program period, DIII-D will resume operation of all eight ion sources, which will increase the total power to 20 MW and the co-injected power to 15 MW.

It is planned to reorient two of the beam lines (four ion sources) to inject approximately 0.3 m below the magnetic axis (see Figs. 3-8 and 3-9). This injection geometry places the maximum neutral beam current drive at approximately the half-radius and provides access and control to high performance, steady-state discharges on DIII-D.

This program plan also includes increasing the pulse duration of the beams from a nominal 4 sec duration (full power) to 10 sec. This provides DIII-D with a 20 MW, 200 MJ NBI system for long pulse steady-state high performance discharges.

Electron Cyclotron Heating and Current Drive Systems. Our program plan is to increase the long pulse EC system from 6 (1 MW) gyrotrons to 8 (1.5 MW) gyrotrons. Electron Cyclotron Power is an extremely powerful scientific tool that supports each of the main elements of the DIII-D program. It not only provides off-axis current drive for stabilization of neoclassical tearing modes (an important consideration for ITER) and feedback control of the current profile for steady-state AT research but also enables a wide range of scientific investigations into transport, stability and stability control. The plan is to complete development of the 1.5 MW gyrotron with the assistance of the Virtual Laboratory for Technology, add two new transmission lines to bring the system to 9 MW, and then replace the 1 MW gyrotrons with 1.5 MW gyrotrons as appropriate. One additional launcher and power supply will be needed.

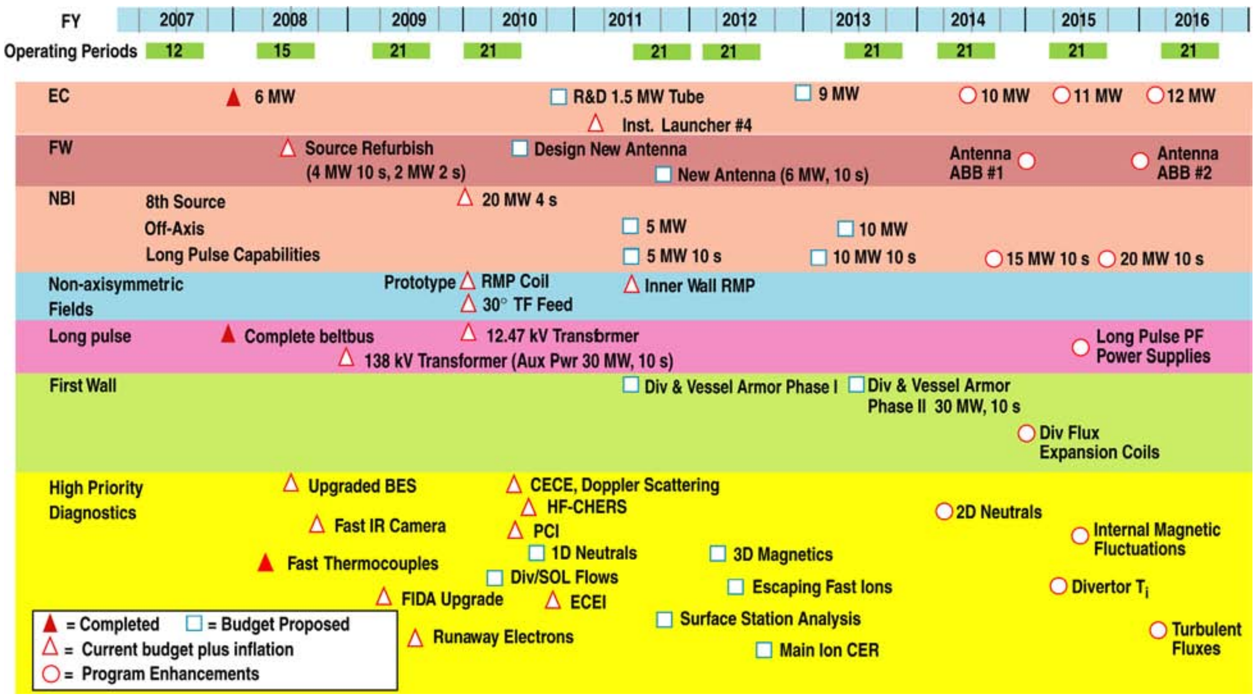

Fig. 1-7. The DIII-D National Facility Five-Year Plan (2009-2013). These capabilities will provide an excellent platform for fusion and plasma science, ITER Support, and AT development for the next decade. The program enhancements shown beyond 2013 will further advance the program and we would plan to implement them as rapidly as funding allows. Symbols show dates when the item is available for experiments.

Table 1-1
Physics Issues Addressed by the Major Facility Improvements

Hardware	Research Elements
NBI: 10 MW, off-axis 20 MW, 10 s	J(ρ), Energetic particles, toroidal/poloidal rotation
ECH (12 MW, 10 s)	Long pulse AT, heat flux control, NTM, J(ρ), $T_e \sim T_i$
FW (6 MW, 10 s)	J(ρ), $T_e \sim T_i$, energetic particles
Inner Wall RMP	ELM control, heat and particle control
Divertor flux expansion coils	Heat and particle control
Divertor and vessel armor upgrade	10 s high performance, physics of heat removal
Hot wall operation	Hydrogenic co-deposition and removal
Custom pellets, Ludwieg tube, liquid jet	Disruption mitigation
138 kV and 12.47 kV transformers	Long pulse AT
Improved and new diagnostics	Fusion science, control, optimization

Fast Wave Systems. DIII-D presently has three Fast Wave systems: two operated at nominally 100 MHz and one at 60 MHz. The transmitters for the 100 MHz systems are completing an upgrade of the final power amplifiers. A new long pulse antenna for the 180 deg, 60 MHz system is planned. The fast heating and current drive system provides very useful electron heating for transport studies with T_e/T_i approaching unity, and is extremely useful for modifying the fast ion content for energetic particle studies. The fast wave also improves electron cyclotron current drive efficiency and raises the bootstrap fraction by increasing T_e, and provides axial current drive for central current profile control in steady-state AT discharges.

ELM Suppression Coils. Recently, DIII-D has shown ELM suppression with the use of a nonaxisymmetric coil set (the I-coils). Our plan is to add additional coils on the inner wall to further develop the scientific basis of resonant magnetic perturbations (RMPs) for ELM suppression by providing more flexibility to vary and optimize the spectrum of applied magnetic perturbations. The inner wall (IW) coils will differentiate between local effects and resonant effects and allow scientists to efficiently drive a wider range of modes.

Ten Second Operation. Ten-second operation allows evaluation of both stationary inductive and steady state scenarios on relevant times scales. The toroidal field belt bus and primary power will be completed early in the program plan.

300 MJ Heat Removal. We plan to upgrade the heat removal capability of the divertor to accommodate 10 second, 30 MW heating pulses. The carbon tiles in the two divertor areas will be modified to carbon fiber composite (CFC), with minor modifications to the water cooling, and the inner wall tiles will be contoured. This provides the capability to evaluate our full performance steady-state scenarios, and the capability to evaluate core boundary coupling at relevant high heat flux values.

Diagnostics. DIII-D has a world leading and comprehensive diagnostic set which is essential for making progress in the understanding and control of fusion plasmas. We propose a significant number of new diagnostics and upgrades to existing diagnostics. The highest priority diagnostics are listed in Fig. 1-7, and a more complete list is included in Chapter 8. These diagnostics are driven by the physics requirements and objectives, and are developed, built and operated by staff from the broader DIII-Team: universities are responsible for a large number of the excellent diagnostic systems. These include key diagnostics in advancing our understanding of turbulence and transport [beam emission spectroscopy (BES), high-k fluctuations, high field charge exchange recombination spectroscopy (HF-CHERS)] energetic particle instabilities and their effect of fast particle transport (BES, 3D magnetics, escaping fast ions, fast ion d-alpha (FIDA)], heat flux to the divertor and projection to larger devices [fast thermocouples fast infrared (IR) camera], mass flow in the boundary [divertor scrape-off layer (Div/Sol) flows and surface station analysis], particle sources and their impact on the pedestal (1D neutrals and 2D neutrals), disruption processes (runaway electrons), and the impact of plasma rotation on transport and stability [main ion charge exchange recombination (CER)].

1.4. FACILITY OPERATIONS

We propose 21 weeks of operation annually for the years 2009–2013 (Fig. 1-7). Twenty-one weeks of operation would allow for more appropriate utilization of the facility, and yet provide sufficient time for diagnostic calibration, maintenance, facility upgrades, and time for experimental planning. Some of the facility upgrades included in the plan require an extended maintenance period, and a schedule is envisioned similar to the 2005 and 2006 schedule, during which experiments were executed in early FY05, a long maintenance period was held crossing 2005/2006, and experiments were executed during the last half of 2006. Sufficient resources to allow work to proceed on two shifts can shorten the duration of the upgrade period. The DIII-D facility continually operates with a very large research back-log. As an example, there were 600 proposals made at the Research Opportunities Forum for 2006/2007. Of these 484 were unique experiments, of which approximately 20% were executed in the 24 weeks of total operation in 2006 and 2007. Most recently, over 450 proposals were received at the 2008 Research Opportunities Forum. The planned increase in operation will allow many more of these excellent experiments to be completed.

1.5. NATIONAL LEADERSHIP

This DIII-D Five-Year Program plan will provide national program leadership in support of the mission of the Fusion Energy Sciences Program:

to advance plasma science, fusion science and fusion technology – the knowledge base needed for an economically and environmentally attractive fusion energy source.

In addition to the National Research Program implemented on DIII-D, the DIII-D program will play an additional role of outreach and leadership to other community groups and processes. The DIII-D program will

- Play lead role in promoting and stimulating *theory and model development* and theory/model validation with the broader theory community, and foster strong alliances with such groups as the U.S. Transport Task Force (TTF), the edge coordinating committee (ECC), the Scientific Discovery through Advanced Computing (SciDAC) theory efforts, and theory groups across the U.S.
- Continue its active program of collaboration and coordination with national in *international experiments* toward a better scientific understanding of fusion plasmas and the international development of fusion energy, including strong participation in the ITPA, and IEA bilateral agreements.
- Continue active *participation and leadership in domestic programs*, coordinating with the USBPO, the Fusion Facilities Coordinating Committee (FFCC), and other U.S. organizations.
- Seek to broaden *collaborations with universities*, both in research involvement on DIII-D, and assisting in the success of university experiments.
- Will continue to work for the *success of ITER*, providing physics input toward key decisions, and physics basis for operational scenarios on ITER, working with the USBPO, the U.S. ITER Project Office (IPO), and the international ITER Organization (IO).
- Continue active participation and leadership in evaluating *new initiatives for the U.S. program.*

- Participate in developing *enabling technologies* critical to the success of the ITER, the AT, and fusion energy science.
- Will support the *outreach* of excellent fusion research to the broader science community, communicating the excitement and progress of fusion energy science, making available data from well diagnosed DIII-D high temperature plasmas, and making the DIII-D facility available for nonfusion research as appropriate.

1.6. INTERNATIONAL LEADERSHIP

The DIII-D program is well respected in the international community, and provides international leadership toward scientific understanding, the success of ITER, and the development of fusion energy. The DIII-D program will continue active involvement in the international community. The DIII-D program will:

- Continue its strong participation and leadership in the ITPA. Presently ~ 40 DIII-D team members participate.
- Continue collaborative experiments on other facilities throughout the world.
- Will provide assistance and leadership in plasma control and plasma start-up world-wide, especially superconducting tokamaks,
- Will seek to play a larger role in research on superconducting tokamaks, aiming to contribute our scientific knowledge and experience, as well as gain knowledge and experience.
- Will continue to provide support and assistance to preparation for operation of ITER, to ensure its success and gain the greatest scientific benefit from.

1.7. BENEFITS OF DIII-D RESEARCH

Research on DIII-D will provide significant benefits to the world fusion community in three main areas:

ITER Physics. DIII-D research will contribute significantly to the success of ITER. The DIII-D facility is uniquely positioned to contribute to resolving issues for the ITER design, especially with respect to ELM suppression, disruption avoidance and mitigation systems, and the adequacy of equilibrium and stability control systems. DIII-D will validate the operational scenarios planned for ITER and provide important data for validating models that extrapolate to ITER. The AT program on DIII-D will be the major contributor over the next decade to developing the physics basis for steady-state high performance scenarios for ITER and beyond. In addition to demonstrating sustained high-performance steady-state scenarios, DIII-D will provide excellent detailed data and modeling to validate such scenarios for ITER, FDF and DEMO. DIII-D operation is providing excellent training for young scientists on ITER-relevant plasmas and ITER-relevant research, to provide the scientific staff needed for success on ITER and needed for the U.S. to maximally benefit from ITER operation.

The benefit to ITER from research on DIII-D will include:

- Detailed physics information for decisions on critical ITER design issues including ELM control coils, poloidal coil requirements, and first wall materials, etc.

- Solutions to critical ITER operational issues including ELM control, MHD stability control, and disruption mitigation, etc.
- Demonstrated, validated long-pulse scenarios that satisfy ITER's high gain mission,
- Physics basis for steady-state operational scenarios on ITER,
- Validation of key physics models for projecting ITER performance, assisting in preparation of ITER experiments, and interpreting of ITER results,
- Demonstrated real-time control capabilities relevant for full range of ITER actuators and scenarios,
- Critical information for the design of ITER diagnostics,
- Training of scientific staff for the success of ITER, and for the U.S. to benefit from operation on ITER.

FDF/CTF, DEMO. Because of the comprehensive diagnostic systems, the flexible heating and current systems, and the high beta stability control systems, DIII-D will be the leading contributor to AT research over the next decade. In addition to demonstrating access to advanced scenarios, and their controlled sustainment, a key contribution of DIII-D will be to provide sufficient detailed knowledge of the discharge characteristics and evolution that will enable validation of complete integrated models and the development of model based control of these discharges. This line of research can provide the basis for steady state scenarios on ITER and moving aggressively forward with a Fusion Development Facility, FDF, in the U.S. in the near term. The AT research on DIII-D and successful exploitation on ITER and other steady state tokamaks, will provide the basis for an attractive DEMO based on this advanced physics. Advanced Reactor Innovation Evaluation Study – Advanced Tokamak (ARIES-AT) provides a vision for cost effective fusion energy production using a high confinement steady-state high beta, high bootstrap fraction tokamak.

The benefit to FDF/CTF and DEMO from research on DIII-D will include:

- Proof-of-principle demonstration of high beta, steady-state operation at performance levels suitable for attractive fusion power,
- Detailed physics basis of steady-state operation including determination of key physics and control requirements,
- Demonstrated methodology for active profile and instability control at optimum performance levels,
- Improved understanding of viable boundary solutions for heat and particle control for high performance steady-state plasmas,

Fusion Energy Science. The DIII-D Program will advance fusion energy science on a broad front. Access to fusion relevant parameters, a comprehensive diagnostic set, flexible heating and current drive systems, nonaxisymmetric coils, and advanced digital plasma control make the DIII-D facility the most advanced scientific instrument in the world fusion program. A broad international team, with excellent ties to the international theory program, provides a unique opportunity to contribute to model validation and predictive understanding of key plasma science and fusion science issues. Detailed scientific investigation and evaluation is a strength of the DIII-D program, and the program is committed to continued excellence in this area, providing appropriate attention to physics measurements and continued broad involvement of the fusion community as part of the objective. Increasing participation of universities is one goal of the program.

The benefit to fusion energy science from research on DIII-D will include:

- Cutting-edge advances in understanding of key physical processes impacting fusion plasma performance
- Validated models that adequately describe complex plasma behavior
- Development of state-of-the-art diagnostics for plasma and fusion science research
- Excellent data for the U.S. and international community for the development and testing of theories and models
- Education of young scientists, including students and post-docs
- A world-leading fusion sciences facility, committed to participation and use by the broad U.S. fusion community

Many remaining important physics questions require outstanding diagnostic access and control of the plasma to address. The ability to independently control important plasma parameters, such as plasma rotation, T_e/T_i, collisionality, current profile, details of the shape, etc. allow unique opportunities to test the models. These independent control capabilities are available on DIII-D, but become very difficult or impossible on future larger experiments. Measurement capabilities, readily available on DIII-D, are very difficult or not possible on larger or nuclear devices. The measurement and control capabilities on DIII-D make a world leading scientific instrument and better suited to address many outstanding scientific questions than would be possible on larger nuclear facilities.

2. RESEARCH IN SUPPORT OF ITER

2.1. GOALS FOR ITER-FOCUSED RESEARCH

ITER is the single most important element of the U.S. fusion program, and the DIII-D program is well positioned to make key research contributions that will help ensure the success of ITER. DIII-D can reproduce ITER's conventional and advanced operating configurations in most key dimensionless plasma parameters, including discharge shape, collisionality, and beta. Furthermore, DIII-D's operating flexibility makes it possible to explore the parameter space around these nominal operating points, and to vary other parameters such as torque and nonaxisymmetric fields. DIII-D's extensive set of diagnostics provides detailed measurements of the plasma behavior in these ITER-relevant regimes. These capabilities allow DIII-D to contribute to almost all of the key areas of research for ITER, addressing specific issues for ITER plasmas as well as the underlying plasma science.

In order for DIII-D's results to realize their potential impact on ITER design and operation, good communication with the rest of the ITER community is essential. Such communication exists now through the International Tokamak Physics Activity (ITPA) and U.S. Burning Plasma Organization (USBPO) where many DIII-D scientists participate actively and hold positions of leadership, as well as the usual large and small international conferences. Direct contact with the ITER organization is also effective and visits by DIII-D scientists to the ITER site have begun.

The emphasis of DIII-D's research in support of ITER will evolve with time, in response to ITER's needs. In the near term, DIII-D will provide solutions to key scientific and technical issues for the ITER design, thus helping to ensure that the design is both cost-effective and adequate for ITER's mission. In the longer term, DIII-D will develop solutions for specific ITER operational issues, integrated scenarios for ITER operation, and advanced algorithms for control of the plasma shape, profiles, and magnetohydrodynamic (MHD) stability, thus helping to ensure that ITER meets its primary goals and improving the quality of ITER's scientific output. The evolution of DIII-D research for ITER over the next 10 years is summarized in Fig. 2-1 and outlined in the rest of this section.

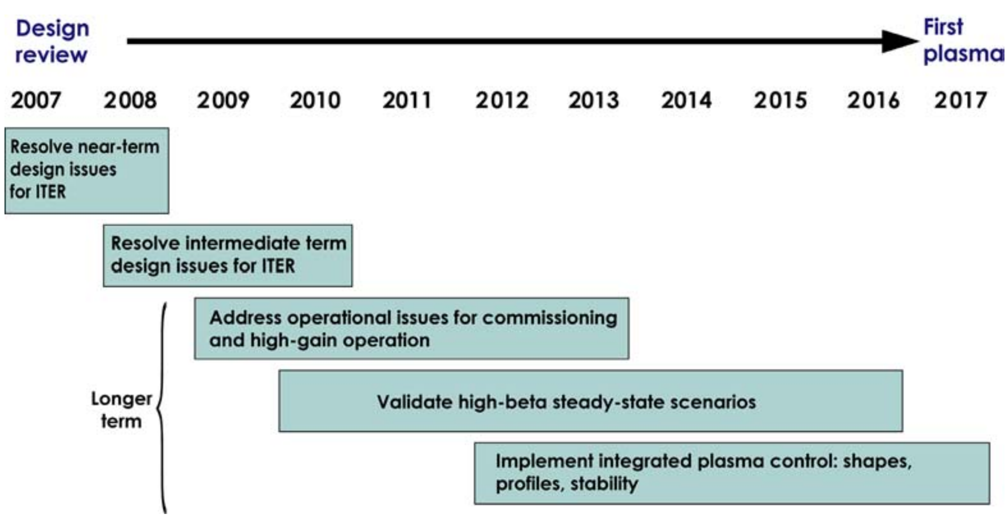

Fig. 2-1. The changing focus of DIII-D research in support of ITER, between now and the start of ITER operation.

2.1.1. Resolve Near-Term Design Issues for ITER (2008–2009)

The choice of near-term research topics in support of ITER is guided by needs of the ITER design and the unique capabilities of DIII-D. The most immediate issues are those with a potential impact on the design of major components, where decisions are likely to be made within the next 1–2 years, such as the design of proposed nonaxisymmetric coils for MHD stability control, and specifications of the poloidal field coils and power supplies. Design-related research should be mostly completed in 2008–2009, but the operation of these systems will continue to be the subject of physics research after the design decisions are made.

Edge localized mode (ELM) control (Section 2.2) is a key issue for ITER's baseline scenario. The design of an ELM suppression system using nonaxisymmetric perturbation coils is an immediate issue for ITER, and DIII-D research will have a major impact in the design decision and validation of the design.

Resistive wall mode (RWM) control (Sections 2.4 and 3.6) is a key issue for ITER's steady-state scenarios. Design issues related to RWM control include possible modifications of ITER's neutral beam (NB) systems for greater torque, validation of the design for EFCCs, design of resistive wall mode control coils, and validation of the selected approach for RWM feedback control.

Flexibility of plasma startup (Section 2.6) is both a near-term and a medium-term issue. In the near term, assessment of axisymmetric stability during plasma startup is a key issue for the design of ITER's poloidal field coils and their power supplies.

Disruption mitigation and characterization (Section 2.7) is again both a near-term and a medium-term issue. A preliminary assessment of the requirements for impurity injection systems to mitigate the effects of disruptions will have immediate relevance to the specification of ITER's pumping systems. Characterization of the electromagnetic and thermal effects of "natural" and impurity-mitigated disruptions is another immediate need that may impact the design of plasma facing components.

2.1.2. Resolve Intermediate-Term Design Issues for ITER (2010–2011)

Medium-term research for ITER is motivated by its potential impact for systems where there will still be some flexibility to change the design during the next few years. These include the design of electron cyclotron heating (ECH) systems for control of neoclassical tearing modes, plasma fueling by solid pellet injection, systems for disruption mitigation, and the choice of first wall materials. Again, design-related research should be mostly completed by 2010–2011, but research related to the operation of these systems will continue after that time.

Neoclassical tearing mode (NTM) stabilization (Section 2.3) is likely to be required in ITER's baseline scenario. DIII-D research will provide validation of localized electron cyclotron current drive (ECCD), the planned approach for NTM control in ITER, and test a possible new method for control of low-beta tearing modes as well as high-beta NTMs.

Fueling by pellet injection (Section 2.5) will be required on ITER. Here the primary issues are the optimization of the pellet injection system for efficacy of fueling and the avoidance of instabilities as the plasma profiles suddenly change.

Disruption mitigation and characterization (Section 2.7) is a critical issue for the long-term survival of the first wall in ITER. DIII-D research in the physics of impurity delivery and assimilation into the plasma, and investigation of other means of impurity injection in addition to gas jets (e.g., pellet injection) may impact the design of ITER's impurity injection system.

Control of tritium inventory (Section 2.8) is an issue closely related to the choice of first-wall materials. The primary focus of the DIII-D program in this area will continue to be on evaluating graphite as a viable first-wall material. This research has the potential to resolve a significant issue for ITER and to contribute to a determination of the first wall material for ITER startup operation.

Fast wave (FW) antenna coupling (Section 2.11) is a challenge in H-mode discharges due to the steep gradients, large plasma-wall gap, and ELM instabilities. DIII-D research will test a prototype of the ITER antenna design and investigate means of improving the coupling.

Divertor heat flux (Sections 3.7 and 5.6) is a potential issue for high-performance discharges in ITER. DIII-D research will validate models for heat transport in the divertor, and explore new methods for heat flux reduction using axisymmetric and nonaxisymmetric coils, with a possible impact on the design of ITER's control coils.

2.1.3. Address Longer-Term Issues for Commissioning and High-Gain Operation (2012–2013 and Beyond)

Longer-term research for ITER will focus initially on operational issues for the commissioning and high-gain phases of ITER operation, including scenario development and other more specific issues. This research will begin in the five-year period described in this plan but will continue until the start of ITER operation. Although it is important that this information be available prior to the beginning of ITER operations, DIII-D's ability to reproduce many of ITER's key operating features will make information from DIII-D very valuable well into ITER operations.

Hydrogen plasma operation (Section 2.10) may yield values of parameters such as the energy confinement time and the power threshold for L to H mode transition that are somewhat different from deuterium or deuterium and tritium (DT) plasmas. In collaboration with tokamaks such as Joint European Torus (JET), Axisymmetric Divertor Experiment Upgrade (ASDEX-U), and C-Mod, DIII-D will expand and refine the database for hydrogen operation in ITER's initial operation, and will design improvements to the operating scenarios in hydrogen plasmas.

Core transport and edge pedestal properties (Section 5.2) determine the fusion power and fusion gain. DIII-D research will continue a strong focus on the physics of thermal transport, leading to validated models that can be used to predict the fusion performance of ITER.

Hybrid scenario plasmas (Section 2.9) represent a potential improvement for ITER's baseline scenario operation. DIII-D research in the next five years will continue to build the physics basis for understanding this mode of operation in ITER-like plasmas, in order that the results can be extrapolated with confidence to ITER.

Energetic ion instabilities (Section 5.4) have potential consequences for ITER's alpha heating. DIII-D research will validate predictive models, develop tools for controlling fast ion-driven instabilities, and explore the possibility that these instabilities can be exploited to improve plasma performance.

2.1.4. Validate High-Beta Steady-State Scenarios

Longer-term research for ITER will continue with operational issues for the subsequent steady-state phase of ITER operation. Here the scenario development is crucial to attain the steady-state goal, and every element of the discharge evolution must be carefully designed including the startup. Again, DIII-D is very well positioned to make significant contributions to development of these advanced scenarios as ITER is in its initial operating phases. To have maximum impact on the choice of ITER's heating and current drive systems and feature options for the first wall divertor, this work should largely be completed prior to ITER operations. But, input from DIII-D on steady-state scenarios would be valuable well into ITER operations.

High-beta steady-state scenarios (Section 3.3) are the basis for the advanced operation planned later in ITER's program. DIII-D research in the next five years will continue its focus on the development of experimental scenarios integrated numerical models that will allow DIII-D results to be extrapolated to ITER.

Flexibility of plasma startup (Section 2.6) is crucial to achieve the range of scenarios that are planned for ITER. DIII-D research will use both modeling and experiments to validate the planned ITER startup scenarios for advanced tokamak (AT) plasmas and to develop improved scenarios.

2.1.5. Implement Integrated Plasma Control: Shape, Profiles, Stability

Comprehensive, robust control of the plasma, its internal profiles, and its stability are common themes of the work described here. The research is aimed at providing control solutions in a form that can be transferred to ITER and other future tokamaks. This work should be largely completed before ITER begins operation, but can continue during the early phases of ITER operation.

Control of plasma profiles and MHD stability (Sections 3.4 and 3.5) are fundamental to both high-gain and high-beta steady-state scenarios in ITER. The DIII-D five-year plan envisions strong efforts in integrated control of current density profiles, plasma shape, and plasma stability, making use of new and upgraded actuators and diagnostics. The development of advanced, model-based control algorithms will provide more robust control in DIII-D and the physics basis for reproducing the control in ITER.

Disruption avoidance by plasma control (Sections 2.7, 3.6, and 7.3) is critical to a broad-based physics program in ITER, in order that high-performance plasmas can be operated with confidence in regimes that may be near stability limits. An integrated system for disruption avoidance will include multiple levels of protection: accurate control of the plasma shape and profiles, real-time detection of stability limits, active suppression of instabilities when they occur, soft shutdown of the plasma when an off-normal condition occurs, and fast shutdown ("disruption mitigation") only as a last resort. DIII-D research will develop these elements, first individually and then as an integrated system.

Emulation of ITER plasma control (Sections 7.3 and 7.4) will take advantage of DIII-D's extensive poloidal field coil set and flexible plasma control system (PCS) to explore the control issues of ITER and other superconducting tokamaks. These include the limits on discharge shaping and positional stability associated with the location and dynamic response of ITER's coils. This emulation capability will allow accurate development and testing of ITER operational scenarios, and will also create an ideal training facility for ITER operators before ITER begins operation.

2.1.6. DIII-D Research in Support of ITER

Key ITER issues and the planned DIII-D research in those areas during the next five-year period are summarized in Table 2-1. This table is focused on the topics discussed in the present chapter, and serves as a summary of the chapter.

2.2. PEDESTAL AND ELM CONTROL

The development of techniques to control the parameters of the edge plasma pedestal is an exciting element of the DIII-D program in support of ITER, and extrapolation of the tokamak to power producing reactors beyond ITER. Techniques for pedestal control range from plans for independent particle fueling and edge current drive, to perturbations of the edge magnetic fields using internal and external coils.

This area of research in support of ITER also involves exciting basic fusion science. For example studies of ablation physics and cross-field penetration of particles from injected pellets will be done in addition to plasma physics research in fully 3D, even stochastic, magnetic field configurations. In the latter case there will be connections between the DIII-D work in the tokamak community and the 3D magnetic field physics from the stellarator community.

Section 2.2.1 below gives an overview of the broad range of physics issues and control techniques associated with the pedestal, including the present status, the DIII-D research elements planned for the 2009–2013 period, and the hardware improvements required for this plan. The various subtopics are then treated in more detail in Sections 2.2.2 to 2.2.6.

2.2.1. Goals of Pedestal and ELM Control

2.2.1.1. Motivation and Status. Performance projections for future tokamak devices, including ITER and future power producing reactors, indicate that optimum core performance is obtained when the plasma is operated with an H-mode edge pedestal. In particular, the improved plasma energy and density content of high-confinement (H-mode) operating regimes will be needed in ITER to meet scenario 2, 3 and 4 goals [Progress 2007]. However, the steep gradients of plasma pressure associated with the H-mode edge transport barrier (ETB) produce large currents localized to the steep gradient region through the bootstrap effect. The combination of high pressure gradient (drive for ballooning modes) and high local current density (drive for kink-peeling modes) makes the pedestal region subject to coupled peeling-ballooning modes that can grow nonlinearly into large edge localized modes (ELMs). The onset of ELMs is in good agreement with theoretical stability limits predicted by the ideal MHD peeling-ballooning model, as shown below in sections 2.2.3 and 2.2.4. Each Type-I ELM, the most common type, delivers

Table 2-1
Contributions of DIII-D research to Key ITER Issues. Significant Hardware Elements Planned in 2009–2013 are Highlighted in Yellow

ITER ISSUE	RECENT PROGRESS	CHALLENGES AND PLANS (2009–2013)
2.2. Pedestal and ELM Control		
• Design of nonaxisymmetric coils for ELM control in ITER • Physics basis for extrapolation of ELM control to ITER • Optimization of ELM control methods for ITER	• ELMs suppressed by resonant magnetic perturbations in a narrow resonant range • ELM-free QH-mode with counter injection • Separation of p′ and J in ETB at high collisionality • Fueling pellets observed to induce ELMs • Small ELMs observed in limited operating regimes	• Inner wall RMP coils for greater range of spectra • Understand RMP effects on particle transport • Extend QH-mode to near-zero or co-rotation • Well-controlled variation of p′ and J with ECCD • Edge pellet dropper for ELM pacing • Effect of RMP and rotation on small ELM regimes • Improve diagnostics and modeling for all regimes
2.3. Neoclassical Tearing Mode Control		
• Validation of predicted power requirements • Optimization of control methods for ITER • NTM stability at low plasma rotation	• NTM pre-emptively suppressed with continuous ECCD and active tracking of q surface	• Validate predicted dependence on current drive width, alignment, and modulation • Oblique ECE diagnostic: local island detection • Simultaneous suppression of multiple modes with increased ECCD power • NTM control at low plasma rotation
2.4. Dynamic Error Field Correction and Resistive Wall Mode Control		
• Design of nonaxisymmetric coils for RWM control in ITER • Critical rotation frequency for passive stability • Optimization of error field correction at low rotation	• RWM stabilization observed at low plasma rotation — consistent with kinetic effects • Resonant RWM response requires minimization of error fields at high beta and low rotation • Nonlinear response to transients (e.g., ELMs) may require feedback in addition to rotation	• Evaluate physics and limits of RWM stability at low plasma rotation • Develop direct feedback control at low rotation • Upgrade I-coil amplifiers: feedback and error correction with expanded harmonic spectra • Upgrade TF coil feed to reduce error field • Improve measurement of plasma perturbation with upgraded SXR, ECE, magnetic diagnostics
2.5. Pellet Fueling		
• Efficacy of pellet fueling if pellets do not penetrate to the plasma center • Compatibility with avoidance of ELMs, NTMs, etc.	• High field side pellet fueling more effective than low field side	• Upgrade high field side pellet injectors for higher throughput • Demonstrate core fueling with moderate penetration • Demonstrate compatibility with plasma stability • Proof of principle test of microwave acceleration
2.6. ITER Startup Scenarios		
• Specification of poloidal field coils and power supply requirements, consistent with required shape evolution and vertical stability	• Constant q_{95} startup discharges are near the vertical stability limit • Early x-point with ℓ_i feedback maintains stability	• Simulate ITER scenarios, validate transport modeling • Low ℓ_i startup and sawtooth free operation for AT and hybrid scenarios • Additional F-coil choppers for full shape control

Table 2-1 (Cont.)

2.7. Disruption Mitigation and Characterization		
• Prediction of disruption effects in ITER • Minimization of heat flux, electromagnetic forces, and runaway electrons during disruptions • Reliable methods of avoiding disruptions	• Massive gas injection is effective in reducing wall heat loads and halo currents • Gas injection in quantity needed for suppression of runaway electron avalanche is difficult to achieve • Delivery rate is limited by long delivery tube • Impurity mixing during the thermal quench depends on MHD activity	• Validate physics of runaway electron multiplication: runaway electron diagnostics • Validate nonlinear, 3D models for impurity mixing and disruption • High density impurity delivery for runaway suppression (inverse jet, custom pellets, …) • Alternate means of runaway suppression (RMP) • Characterize thermal and electromagnetic loads • Develop reliable control for disruption avoidance
2.8. Plasma Facing Materials		
• Choice of first wall material for initial operation of ITER • Tritium retention in co-deposited layers of fuel and carbon ions • Dust a potential safety issue	• Divertor material evaluation station (DiMES) shows reduced codeposition at increased surface temperature • Thermal oxidation rate of carbon co-deposits increases with temperature, oxygen pressure, and deposit thickness • Rayleigh scattering and fast-framing video measure dust during plasma operation	• Advanced main chamber materials station • Coated tiles to study materials migration • DIII-D operation with heated walls/divertor • Removal of hydrocarbon deposits by oxygen bake • Characterize dust formation mechanisms, transport, and accumulation rates
2.9. Hybrid Scenario Development		
• Potential improved performance for ITER's baseline scenario • Physics basis for extrapolation to ITER is needed	• Low rotation hybrid plasmas sustained near the threshold for mode locking • Observed increase in low-k turbulence when T_e/T_i increases	• Physics understanding of q(0) modification and improved confinement • Validate hybrid performance at ITER-like parameters ($T_e \sim T_i$, low rotation)
2.10. Hydrogen Plasma Operation		
• Need confidence that ITER will achieve good H-mode performance during the initial hydrogen operation phase	• Empirical scalings suggest ion mass scaling of L-H power threshold and energy confinement is unfavorable for ITER hydrogen phase operation	• Determine ion mass dependence in consistent, ITER-like plasmas: P(L → H), τ_E, ELMs and ELM control, wall equilibration • Ion species mix diagnostic
2.11. Fast-Wave Coupling Tests for ITER		
• Validation of ITER low-voltage antenna design (parallel-fed triplet straps) • Reliable coupling to H-mode plasma edge	• Coupling to H-mode plasmas is challenging due to large plasma-antenna gaps and ELM-induced transient changes in coupling	• Test gas injection to raise SOL density near antenna • Install and test ITER-style antenna in DIII-D

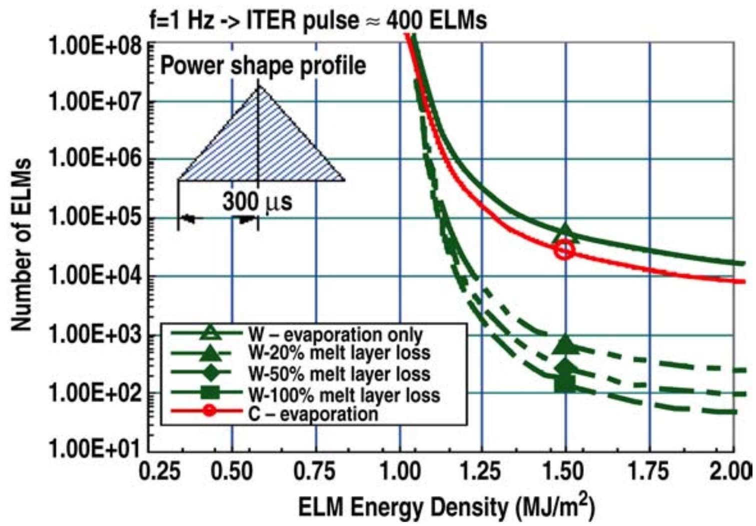

Fig. 2-2. Prediction of the number of ELMs that can be handled by the ITER divertor targets before the induced erosion requires target PFC replacement vs. ELM energy density. Curves are shown for pure carbon and tungsten assuming no melt layer loss, and for tungsten assuming three different melt layer loss models [Federici 2003, Reprinted from J. Nucl. Mater., Vol. 313-316, G. Fererici et al., "Key ITER plasma edge and plasma–material interaction issues," p. 12, Copyright (2003), with permission from Elsevier.]

an intense, short, pulse of energy and particles that flash-heats divertor plasma facing components (PFCs). Projections of the material erosion in the divertors of ITER and future tokamak reactors indicate that there is a threshold in ELM size such that the transient erosion during ELMs will significantly limit the operation of these devices (Fig. 2-2 from [Federici 2003]).

Recent material tests have revised downward the tolerance of candidate materials to large heat pulses. The new data on changes in the surface properties and thickness of tungsten and carbon fiber composite (CFC) materials, when exposed to impulsive energy bursts similar to Type-I ELMs, indicates that ELM energy impulses of greater than 3 MJ at the divertor target plates in ITER will cause a significant erosion of the divertor components with each ELM. In these studies, erosion was negligible at 0.5 MJ/m^2 but increased dramatically at 1.5 MJ/m^2, leading to an average CFC erosion of 0.3 μm per pulse, and intense droplet ejection from tungsten surfaces [Zhitlukhin 2007]. A 3 MJ ELM in ITER represents only 1% of the total 300 MJ stored plasma energy and approximately 3–4% of the pedestal energy, but projections from current experiments suggest that in ITER the energy lost from the pedestal will range between 15% to 20% of the pedestal energy, or approximately 15 MJ. Clearly, 15 MJ ELMs must be avoided in ITER.

Resonant magnetic perturbations (RMP) produced by nonaxisymmetric coils have completely eliminated ELMs in DIII-D, (as shown in Fig. 2-3 from [Evans 2006]) when operating inside a well defined resonant window in q_{95} specified by the RMP coil geometry. In addition, when operating outside the resonance window, a reduction in the amplitude of the ELMs is found [Evans 2008]. Such mitigation has reduced the ELM energy to ~2% of the total stored energy in JET [Liang 2007] and DIII-D [Evans 2004] but this is still a factor of 2 above the ablation limit when scaled to ITER. In addition to complete ELM suppression during operation in the q_{95} resonant window with RMPs, discharges without ELMs have also been achieved under specific conditions that define the quiescent H-mode (QH-mode) operating regime. Several other approaches have also been identified for mitigating or eliminating ELMs. Each will be discussed in more detail in later subsections:

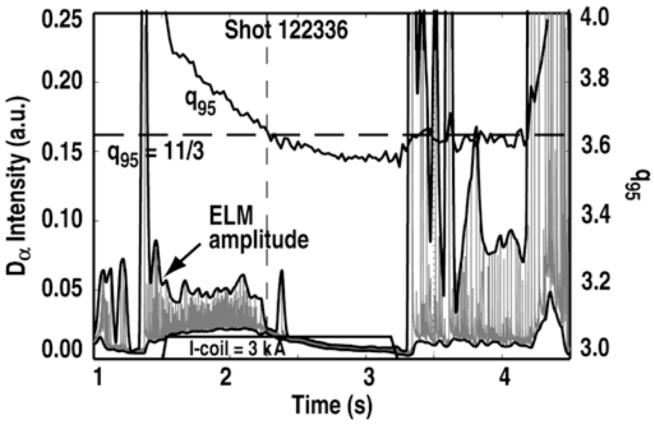

Fig. 2-3. Lower divertor D_α intensity, q_{95} safety factor at the 95% normalized poloidal flux surface evolution, and applied I-coil current in DIII-D discharge #122336 [Evans 2006].

- Application of RMPs to the edge plasma using magnetic coils to control the edge plasma pressure (Section 2.2.2)
- QH-mode operation, in which the edge plasma self-generates an edge harmonic oscillation (EHO) that prevents ELMs (Section 2.2.3)
- Independent control of the edge pressure and edge current density profiles, using fueling, heating and current drive tools, to change the edge stability (Section 2.2.4)
- Use of high frequency small pellet injection to increase the frequency of the ELMs and simultaneously reduce their amplitude (Section 2.2.5)
- Exploration of several “small-ELM regimes” that have shown promising attributes in DIII-D and other tokamak experiments (Section 2.2.6)

2.2.1.2. Research Elements for 2009–2013. The goal of the Pedestal and ELM Control program on DIII-D is to qualify techniques that control the edge plasma pedestal parameters and the ELM particle and energy losses without core confinement degradation, and to develop the physics understanding needed to reliably extrapolate the techniques to ITER and tokamaks beyond ITER.

The best way to achieve the physics understanding of the mechanisms controlling the pedestal, and to insure that the most promising combination of pedestal and ELM control techniques are developed, is to pursue multiple methods simultaneously and to explore synergistic effects of the combined techniques. The DIII-D program is uniquely positioned to develop five of the most promising approaches for pedestal and ELM control. The program will search for commonalities and synergies in the physics producing pedestal and ELM control across these techniques to strengthen the confidence in the extrapolability of the most promising techniques to ITER.

The major research elements in this area include

- Understanding the physics of ELM suppression by RMPs, including edge particle transport in the presence of the RMP and the interaction of plasma rotation with the applied magnetic perturbation. (Section 2.2.2)
- Understanding the physics of the QH-mode regime, including the saturation mechanism for the Edge Harmonic Oscillation, edge particle transport in the presence of the EHO, and extension of the QH-mode regime to plasmas with low rotation. (Section 2.2.3)

- Use of gas puffing and ECCD for separate variation of the edge pressure gradient and current density, toward the goal of ELM pacing, mitigation, or elimination. (Section 2.2.4)
- Understanding the physics of ELM pacing by shallow pellet injection – the primary ELM mitigation technique planned for ITER at present. (Section 2.2.5)
- Development of operating regimes with H-mode confinement where ELMs are naturally much smaller and more rapid than the typical Type-I ELMs. (Section 2.2.6)

Because of DIII-D's unique ability to study each of these approaches individually or in combination, research in this area will emphasize the development of a reliable, highly optimized, ITER ELM control technique. Our extensive experimental database will help to establish a more comprehensive theoretical understanding of the structure of the pedestal region and its relationship to the nature of the ELM instability. Experimentally validated numerical codes will be used to predict how the properties of the pedestal, ELMs and ELM control techniques will scale to ITER. Strategic hardware and software upgrades to our control, diagnostic and analysis tools will enable us to acquire basic data needed to develop a more fundamental physics understanding of pedestal transport and stability. These tools will allows us to synthesize more efficient and predictable ELM control scenarios based on individual or combined approaches that are unique to the DIII-D program.

2.2.1.3. Hardware and Diagnostic Improvements. The planned research in this area is aimed at two broad goals: a deeper **scientific understanding** of transport, stability, rotation, and nonaxisymmetric magnetic fields in the H-mode pedestal, and the application of this understanding to the **control** of ELMs and other edge pedestal properties. Both of these goals rely strongly on improved diagnostic measurements and improved control hardware.

The research described in the following subsections calls for several new control actuators:

- One or more sets of new, internal nonaxisymmetric RMP coils (Section 2.2.2)
- Supersonic gas jet for edge profile modification (Section 2.2.4)
- 10 m/s pellet dropper (Section 2.2.5)
- 300 m/s high-throughput pellet injector (Section 2.2.5)

In addition, the research described in the following subsections requires significant improvements in diagnostics:

- Improved time and space resolution for the Thomson scattering system
- Improved signal to noise ratio, number of digitizer samples, and mode structure analysis software for magnetics and soft x-ray (SXR) diagnostics
- Improved edge current density diagnostics
  - — Multi-view motional Stark effect (MSE)
  - — Improved lithium beam Zeeman effect measurements
- Measurement of main ion temperature and rotation velocity
- Fast ion loss detector
- Gas puff imaging system to observe ELM dynamics

- Fast infrared (IR) cameras with improved viewing optics and higher spatial resolution at several toroidal locations
- Toroidally resolved divertor calorimetry system
- Upgrade of the divertor Langmuir probes and thermocouples

2.2.2. New Nonaxisymmetric Coils

2.2.2.1 Nonaxisymmetric Coils: Status. Experiments using the low field side (LFS) DIII-D internal I-coil, shown in Fig. 2-4 2, configured for n=3 RMPs have proven successful for completely eliminating ELMs in low collisionality plasmas with ITER similar shapes. These coils, designed primarily to interact with resistive wall modes in the core plasma when configured for n=1 perturbations, produce weak pedestal resonance with poloidal mode numbers between 9 and 15 that reduce the pedestal pressure and stabilize ELMs when configured for n=3 operations. Since these coils are not particularly well suited for producing large pedestal resonances with small core resonances as needed for optimal ELM control, the operating space accessible for eliminating ELMs in DIII-D is limited to a rather small range of shapes, collisionalities and toroidal momentum values.

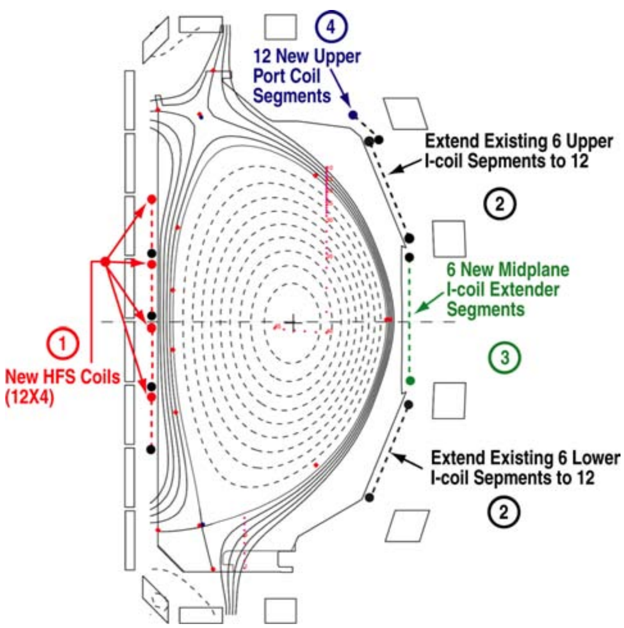

Fig. 2-4. Existing and proposed RMP ELM control coil set for the DIII-D facility. Shown here are: 1 the proposed high field side coils with 2 rows having 12 toroidal segments above the midplane and 2 rows having 12 toroidal segments above the midplane, and 2 the existing upper and lower I-coil with 6 toroidal segments. In addition, the figure shows several possible future upgrades: 2 an extension of the upper and lower I-coil to 12 toroidal segments, 3 a set of low field side midplane coils with 6 toroidal segments and 4 a set of low field side upper port coils with 12 toroidal segments.

2.2.2.2. Nonaxisymmetric Coils: Research Elements. Based on the unique DIII-D successes of RMPs to suppress ELMs in previous experiments, a phased approach of adding new multi-loop internal coil sets is proposed in the next five years. This proposal will assure that DIII-D remains in the forefront worldwide in RMP ELM Control research. The proposed strategy is to start with a new coil set that our present understanding indicates gives the maximum flexibility to modify the toroidal and poloidal mode spectrum of the perturbation. The initial coil set upgrade will be evaluated extensively in experiments and the resulting physics understanding will be used to plan future upgrades. Details of possible further coil set upgrades are given below.

With the new coils described below, we propose to study the detailed physics of how RMPs suppress ELMs by combining experimental results from application of new RMP mode spectra possible using the new coils with edge pedestal stability and transport modeling. Stochastization of the edge magnetic surfaces using the DIII-D I-coil represents an edge pressure control technique for avoidance of the peeling-ballooning instability. Understanding of how this pressure reduction is achieved will be a fundamental focus of the physics studies with the new coils. This work will be coupled with ELITE modeling of peeling-ballooning stability and simulations of edge transport in the presence of RMPs to optimize the pedestal pressure consistent with ELM suppression using this technique. In particular, the modeling will focus on several topics: (1) understanding the physics of edge particle transport with RMPs, (2) determining whether the ELM suppression has an upper bound in edge collisionality, (3) investigating the influence of screening of the RMP fields by the plasma response in a rotating plasma, and (4) determining if a criterion for predicting ELM suppression in future devices can be developed based on a minimum required width of a region in the edge with good overlap of the perturbed magnetic islands when all the screening factors are included. Finally, the theory and modeling effort will also investigate the effect of the low n-number resonant braking and nonresonant braking of the plasma by the components of the new mode spectra that are not resonant in the pedestal. This work will require close coupling with the core theory communities where much of the rotation braking theory (both resonant and nonresonant) is being developed.

2.2.2.3. Nonaxisymmetric Coils: Hardware Improvements. Additional coils, specifically designed to eliminate ELMs in a wide variety of DIII-D shapes and operating parameters, are needed to advance our understanding of the basic physics mechanisms involved in pedestal and ELM control. Several new coils designs were considered, as shown in Fig. 2-4, for addressing various aspects of ELM control physics in DIII-D. The option selected as the highest priority for the next five years is a high field side (centerpost) coil set composed of 4 rows (2 upper and 2 lower) consisting 12 window frame segments toroidally each (Fig. 2-4 1). This coil set will produce large n=3 and n=4 pedestal resonances with much smaller core resonances than the existing n=3 I-coil. RMPs from this coil set can be used with or without those from the I-coil to give us independent control over the mix of core and pedestal resonances. The combination of this coil set and the I-coils provides the required experimental flexibility of both the toroidal and poloidal mode spectrum that will be needed to validate physics models of RMP ELM suppression. Figure 2-5 shows just one unoptimized example of the kind of increases in the edge perturbation that can be achieved with the combination of the present I-coils and this new set of coils on the centerpost. We will also be able to independently rotate either the n=3 or n=4 high field side resonances from this coil set while keeping the LFS I-coil resonances fixed. Since this coil set is located on the centerpost, it will not produce large near field perturbations on peeling-ballooning eigenmodes located in the bad curvature region of the discharge and can be used to isolate resonant effects from nonresonant effects. To produce sufficient RMP for the proposed physics studies, these coils should be capable of approximately 4 kA-turns (kAt) each for pulses of several seconds (3–4 seconds is desirable for DIII-D discharges). Frequency capability should be in the 0 – 50 Hz range. Parameters are summarized in Table 2-2. Modest current, multi turn, inertially cooled coils with a relatively simple design compared with the present high current, single turn, actively cooled I-coils, should be able to fulfill this combination of requirements (Section 10.7.1).

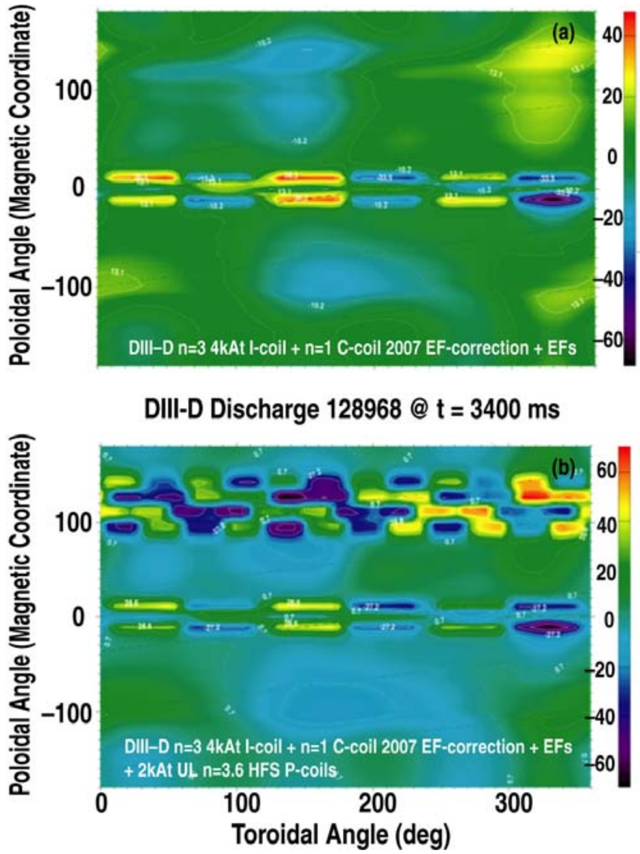

Fig. 2-5. Comparison of the toroidal and poloidal perturbation structure using (a) the I-coils alone in n=3 at 4 kAt, and (b) and unoptimized combination of the I-coils plus the new inner wall 48 coil set (P-coils) in n=3 and n=6 at 2 kAt. For the selected q95 = 3.6 both the I-coils and the P-coils are pitch resonant at the edge so that the perturbation to pedestal particles is substantially enhanced with the addition of the P-coils.

Table 2-2
Design Parameters Of Prioritized Coil Options Shown in Fig.2-4 Indicating the Number of Poloidal and Toroidal Coil Segments, the Total kA-turns in the Coils, the Number of Turns in the Coil Winding, the Required Pulse Duration and the Operating Frequency Range Required

Status	Segments (pol X tor)	Current (kAt)	Number of Turns	Δt (s)	f (Hz)
① Proposed	4 X 12	4	4	3-4	0-50
② Possible Upgrade	2 X 12	8-10	1	10	0-1000
③ Possible Upgrade	1 X 6	8-10	1	10	0-1000
④ Possible Upgrade	1 X 12	8-10	1	10	0-1000

Several possible later upgrades to follow the high field side coils are also shown in Fig. 2-4. An upgrade of the existing upper and lower rows of the existing I-coil (as shown in Fig. 2-4 2) from 6 toroidal segments each to 12 toroidal segments each will allow us to operate the I-coil with either n=3 or n=4 RMPs and to rotate the perturbations toroidally as well as increasing the relative magnitude of the pedestal resonances with respect to the core resonances. A set of LFS midplane coils (as shown in Fig. 2-4 3) composed of 6 toroidal window frame segments will allow us to supplement the n=3 pedestal spectrum produce by the upgraded I-coil and the centerpost coil thus increasing the relative weight of the pedestal resonances to those in the core. Finally, an upper port coil configuration located near the R+2 ports in DIII-D (as shown in Fig. 2-4 4) will mimic the ITER port plug RMP ELM control coil design when combined with the LFS midplane coil (shown in Fig. 2-4 3). This coil set will allow us to compare the efficacy of the ITER port plug coil design compared to the upgraded I-coil and centerpost coil (P-coil) designs. Parameters are summarized in Table 2-2.

2.2.3. Axisymmetric Control (QH-mode)

2.2.3.1. QH-Mode: Status. A QH-mode edge would be the ideal edge for ITER, since the absence of ELMs means there are no pulsed divertor particle and heat loads. The QH-mode was originally discovered in DIII-D [Burrell 2001, Burrell 1999] and has since been seen in ASDEX-U [Suttrop 2003], JT-60U [Sakamoto 2004] and JET [Suttrop 2005]. QH-mode plasmas have the standard H-mode edge pedestal and exhibit H-mode levels of confinement; typical ITER H_{89P} confinement enhancement factors [ITER 1999a] are about 2. However, as illustrated in Fig. 2-6, QH-modes operate without edge localized modes (ELMs) and with constant density and radiated power.

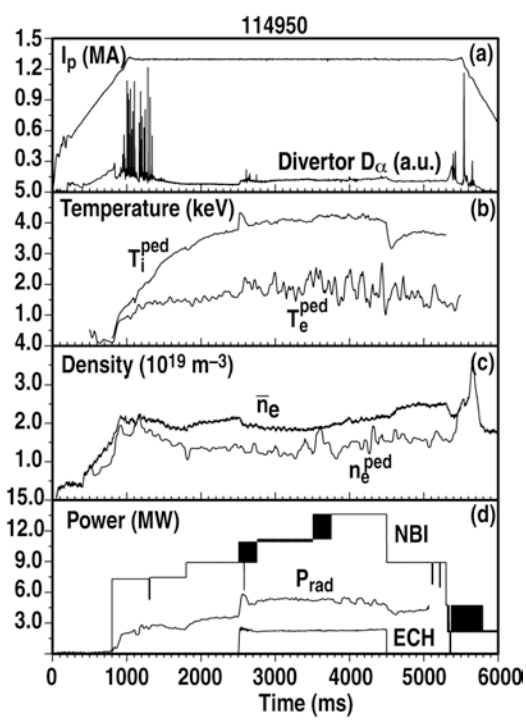

Fig. 2-6. Time history of QH-mode shot 114950 showing (a) absolute value of plasma current and divertor D_α emission, (b) pedestal electron and ion temperature, (c) line-averaged and pedestal electron density, (d) neutral beam and ECH power and total radiated power. Toroidal field is 2.0 T.

A key feature of the QH-mode is an edge electromagnetic mode, the edge harmonic oscillation (EHO). Experiments have shown that this mode enhances edge particle transport. Coupled with divertor cryopumping, this extra particle loss from the plasma provides a means of controlling the pedestal density and, hence, pedestal pressure. Peeling-ballooning mode stability calculations using the ELITE code [Snyder 2002, Snyder 2004] indicate that it is the edge pressure reduction which allows the plasma to operate in the parameter regime which is stable to the peeling-ballooning modes, as shown in Fig. 2-7.

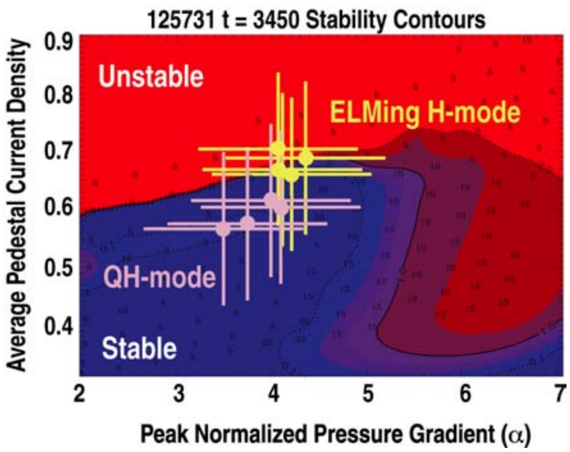

Fig. 2-7. Results of ELITE stability calculation a lower single null plasma. Stability is shown as a function of normalized edge pedestal pressure and normalized edge pedestal current density. Blue regions are stable and red regions are unstable to peeling-ballooning modes. The data points show the operating point for QH-mode.

A major goal of research in this area in the next five years is to create QH-mode in low toroidal rotation plasmas, such as those anticipated in ITER. The initial QH-modes in DIII-D were run using neutral beam injection (NBI) opposite to the plasma current (counter-injection), which produces plasma rotation in the counter direction. A recent series of experiments investigated the effects of combined co plus counter NBI. The most surprising and significant result of these experiments was the discovery that the edge particle transport in QH-mode plasmas can be controlled by using different beam combination to change the edge rotation. One of the continuing themes in fusion research is the search for techniques to allow direct control of the plasma transport. Using combined co plus counter NBI in QH-mode plasmas, we have now demonstrated this control for the edge particle transport, but in order to apply this control in future devices, we need a detailed physics understanding of how this works. We do know that the character of the EHO changes significantly when we change the edge rotation, but the mechanism by which this change in EHO character affects the particle transport is not understood.

2.2.3.2. QH-Mode: Research Elements. In order to make progress in understanding the QH-mode edge plasma, we propose both further experiments and broader theory-experiment comparison. One area where we need improved control tools is the area of intrinsic error fields caused by minor imperfections in the DIII-D coil set, since error-field induced locked modes become a greater problem as the plasma rotation decreases. In the future we will employ the error field feedback control techniques first pioneered by the group studying resistive wall modes in order to optimize the error field correction, especially for high beta conditions.

We propose a significant upgrade in the poloidal magnetics analysis, which couples the EFIT-determined shape of the plasma with models of the toroidal currents, in order to obtain information on the structure of the EHO. By doing a least squares adjustment of those currents to match the magnetic measurements in the presence of the EHO, it should be possible to obtain information on the radial location of the helical currents in the EHO.

We propose to develop the diagnostic capability to measure the deuterium rotation velocity because it is a critical parameter needed to validate nonlinear MHD code models of QH-mode plasmas. Since the EHO is an electromagnetic plasma oscillation, MHD codes such as NIMROD should be able to describe it in detail. However, at present the charge exchange spectroscopy system typically measures the rotation speed of fully stripped carbon ions in the plasma. Especially at low rotation speeds, theory tells us that the main ion and carbon ion toroidal rotation speeds can be quite different.

Another critical measurement that we propose to begin making on a regular basis in the next five years is the current density profile in the edge pedestal, which is required for accurate peeling-ballooning mode stability analysis. To date, most analysis has been done using the calculated bootstrap currents. We have a few shots where this has been compared with edge current density measurement using the lithium-beam polarimetry system; the agreement with the calculations was reasonable. However, this measurement is far from routine. The lithium-beam polarimetry diagnostic and improved MSE using co plus counter beams are both possibilities for edge current density measurements.

In the next five years we will have as a goal the development of a full model for the saturation mechanism responsible for generating the EHO in QH-mode plasmas instead of ELMs. The ELM free quiescent H-mode may represent a regime in which access to the low n branch of the peeling-ballooning mode coupled with strong rotational shear in the edge leads to saturation of the instability. Additional modeling work with ELITE is needed to understand how to optimize this effect. This would involve coupling of a model for the transport effects of the mode with a nonlinear edge stability codes which includes the effect of rotational shear and wall drag.

Although we have MHD codes like ELITE and NIMROD to compare with the EHO measurements, there is still a major hole in the theory of the QH-mode. We need a theory which tells us how the EHO can move particles across the flux surfaces. This is an area where we must work with the theoretical community to develop an improved understanding.

2.2.3.3. QH-Mode: Hardware Improvements. In order to carry out the work described above, we need improved control tools, improved diagnostics and improved analysis of existing diagnostic data. A list of the diagnostic and analysis improvements is given in Table 2-3.

Table 2-3
Diagnostic and Analysis Improvements for QH-mode Studies

1.	Improved poloidal magnetics analysis to localize EHO
2.	Main ion T_i and v_ϕ measurement
3.	Improved Thomson scattering edge spatial resolution
4.	Better edge current density measurement (lithium beam and/or MSE)
5.	Edge T_e measurement with better time resolution
6.	Longer data record for fast magnetics
7.	Fast ion loss detector, similar to TFTR and ASDEX-U

2.2.4. Separate Control of Edge Pressure and Current Density

2.2.4.1. Edge Control: Status. The pedestal research goal that would provide the maximum flexibility for generating solutions to pedestal and ELM problems in future devices is to understand how to independently control the two components that determine the pedestal stability, namely the edge pressure and the edge current density. The DIII-D program has unique capabilities to explore this independent control and these will be exercised in the next five years as described below.

The pressure gradient and current density profiles in the H-mode ETB along the with the shape of the flux surfaces and q value in this region set the instability threshold and spatial structure of the peeling ballooning mode which is responsible for ELMs. The critical pressure gradient for the instability affects the maximum pressure in the ETB which strongly impacts the overall energy confinement, while the structure of the instability affects the ELM size and may play an important role in saturation of the instability as QH-mode.

In the strongly shaped plasmas required for high performance operation, current density is stabilizing while pressure gradient is destabilizing along the high toroidal mode number ballooning branch of the instability, while the opposite is true along the low n peeling mode branch. In the region of strong coupling between the peeling and ballooning branches, which generally is the point of maximum obtainable pressure gradient, relatively small changes in the current density or pressure gradient can have a strong effect on the structure of the peeling-ballooning mode (Fig. 2-8). Therefore independent control of the pressure gradient and current density can provide control on structure of the instability.

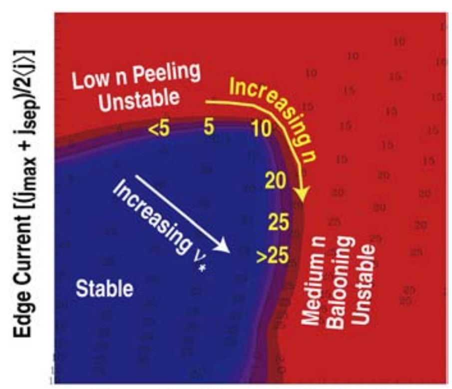

Fig. 2-8. Typical peeling-ballooning mode stability boundary in the space of H-mode pedestal current density and pressure gradient. High J destabilizes low toroidal mode number, n, peeling modes while high pressure gradient destabilizes high n ballooning modes. J is largely bootstrap current which is proportional to p´, but decreases with increasing collisionality, ν_*.

2.2.4.2. Edge Control: Research Elements. In experiments to date, the current density in the ETB is largely bootstrap current and therefore it is difficult to vary current density relative to the pressure gradient profile. The relationship between current density and pressure gradient can be broken at high collisionality where the bootstrap current is suppressed (Fig. 2-9). This case has been studied in detail on DIII-D as an insight into the instability but is not interesting as an ELM mitigation technique since future tokamaks will be collisionless. Pellet ELM pacing, as described in Section 2.2.5, represents a technique for separating p´ and j by raising the pressure gradient on a time scale faster than the edge current relaxation time. A supersonic gas puff system is proposed to provide a similar and more flexible facility. However, it would be difficult to obtain fine scale independent control of p´ and j with these techniques.

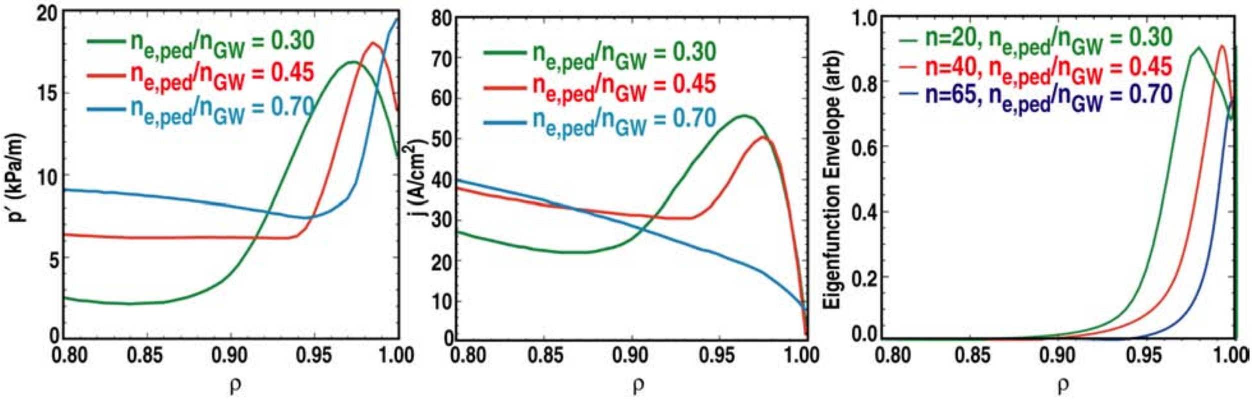

Fig. 2-9. Region of high pressure gradient, p´, shifts towards separatrix and edge bootstrap current, j, is suppressed as density is raised with gas puffing. Peeling ballooning eigenmode width decreases at high density and toroidal mode number, n, increases.

In future experiments we propose to use ECCD, to separate ETB p´ and j with good spatial and temporal control. Although ECCD is inefficient at the lower temperatures in the ETB, relatively small changes in the edge current are predicted to have significant effects (Fig. 2-10). Some possible applications of this technique are:

1. ECCD ELM pacing. Here we propose to use counter ECCD to drive across the ballooning mode boundary giving higher n modes and perhaps smaller ELMs,
2. Extension of QH-mode to higher density. The proposal here is to use co-ECCD to access the low n peeling-mode branch at higher density. If this was successful it would be an important result for ITER since current calculations indicate that the low n peeling branch is accessible in ITER only at densities well below those which are required for the desired fusion power output.
3. Reduced spatial extent of the instability region. By changing the spatial relationship of the current density and pressure gradient we propose to destabilize a smaller spatial extent of the ETB than would occur naturally and thereby reduce the ELM size.

2.2.4.3. Edge Control: Hardware Improvements. Key hardware elements for this research include

- Increase of electron cyclotron (EC) power to 12 MW, allowing simultaneous control of edge and core current density profiles
- Supersonic gas puff system for modification of the edge collisionality

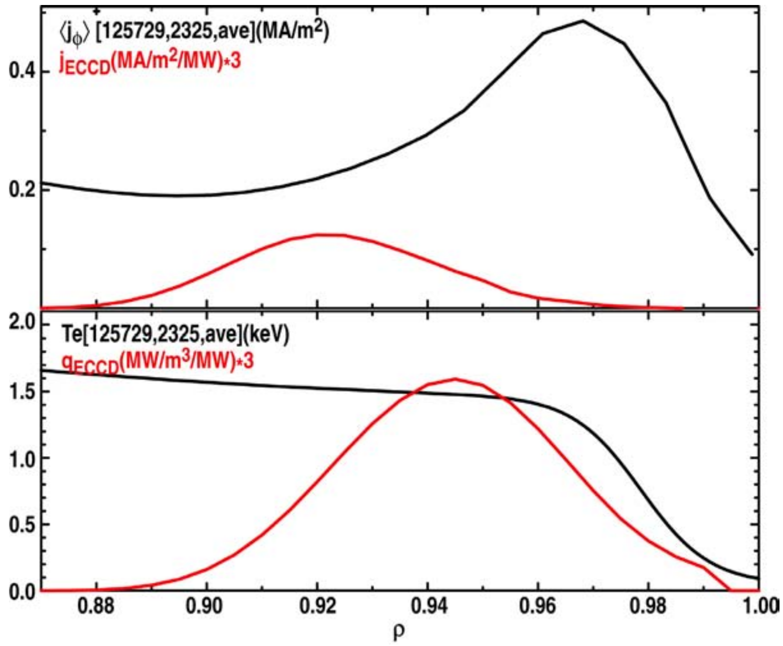

Fig. 2-10. Predicted electron cyclotron current drive, j_{ECCD}, and power density, q_{ECCD}, for 3 MW of ECH power computed based on DIII-D QH-mode discharge 125729 but with toroidal field = 1.5T rather than 2.0 T. Second harmonic resonance is inboard of major axis near upper X-point. Flux surface average plasma current density, $<j_\phi>$, is scaled down with the B_T ratio from actual experiment. There is also large electron heating in the pedestal region and increasing T_e by 50% would increase j_{ECCD} by a factor of 2.

2.2.5. Pellet ELM Pacing

2.2.5.1. Pellet Pacing: Status. Pacing of ELMs with small shallow penetrating pellets is an attractive option for mitigating the Type I ELM problem expected in ITER. ELM pacing with pellets has been pioneered on ASDEX-U by the injection of 0.8 mm equivalent size pellets at up to 80 Hz for 1 s from the inner wall. This was at up to 2.5 times the intrinsic ELM rate and led to smaller higher frequency ELMs with a modest 10%–15% reduction in plasma stored energy. A density increase of 5% was observed in these experiments likely due to the fairly deep penetration from the 1000 m/s pellet speeds.

2.2.5.2. Pellet Pacing: Research Elements. Improvements in ELM control using smaller slower pellets injected from the LFS are planned for ASDEX-U and JET as well as the DIII-D pellet dropper. It is anticipated in these experiments that less fueling efficiency will result in a more modest confinement degradation while maintaining the smaller high frequency ELMs for longer durations. Experiments are underway on DIII-D to develop this technique and better understand the physics of pellet triggered ELMs using a high-frequency, low-velocity pellet dropper, as described below.

It is important to note that although pellet pacing is identified as a primary ELM mitigation technique for ITER, there is little understanding of the effect. We will couple experimental data from the new high frequency pellet dropper on DIII-D, with a modeling effort that combines a pellet ablation model with transport and stability codes to illuminate how the pellets change the edge profiles to trigger the ELM. Nonlinear modeling of the ELM collapse phase may give understanding of how the ELM energy loss is reduced with pellets.

2.2.5.3. Pellet Pacing: Hardware Improvements. We propose to complete the installation, operation and full experimental testing of the pellet dropper system (Fig. 2-11) for ELM pacing on DIII-D in the next 1–2 years. The pellet dropper is capable of 50 Hz 1 mm pellets dropped from a vertical port on DIII-D and achieves a pellet speed of ~10 m/s. Experiments in the next two years will demonstrate whether this is a sufficient technique to trigger frequent small ELMs. The dropper has the capability to adjust the pellet sizes to about 50% smaller than 1 mm. If the range of pellet sizes is insufficient the diameter of the dropper ice extrusion can be modified to make larger pellets. Higher pellet speeds can be obtained only by adding a gas gun mechanism to the dropper, which can be done with some development effort.

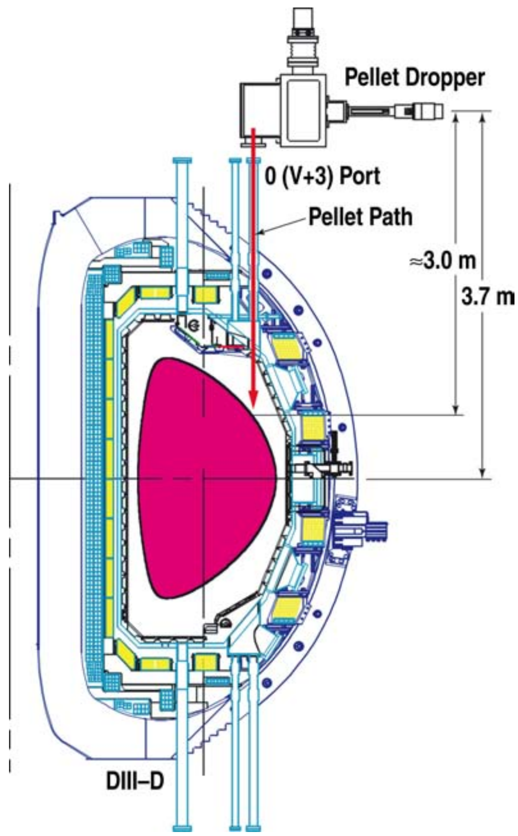

Fig. 2-11. Pellet dropper for ELM pacing. This system injects low-velocity pellets into the low-field side edge plasma.

If the pellet dropper proves inadequate for ELM pacing in DIII-D then a gas gun injector for ELM pace making will be developed on DIII-D in the next five years. Prototype components are being developed for the ITER pellet injector at Oak Ridge National Laboratory (ORNL) in the next three years. Once these are proven they should become available to construct a steady-state high throughput gas gun that could produce 300 m/s pellets of variable size and high rep rate. We propose to install such an injector on DIII-D to complement the existing fueling injector and make use of much of its infrastructure. Low field side injection near the horizontal midplane would be the preferred option for mounting such an injector that would be dedicated for ELM pace making optimization on DIII-D. Such a system would also be useful to study ELM pacemaking in an ITER like configuration.

2.2.6. Small ELM Regimes

2.2.6.1. Small ELMs: Status. In the next five years the DIII-D program will continue to explore the exciting possibility that the problem of large Type-I ELMs can be solved simply by learning to operate the plasma in good confinement regimes that generate much smaller, more rapid releases from the pedestal. There are a number of compelling reasons to develop intrinsic small ELM regimes that extrapolate to ITER including: (1) no specific hardware upgrades are needed to access these regimes, (2) active control actuators are not required, and (3) the plasmas are typically more stable than when using some of the other ELM control techniques. However, continued research is needed because there is presently little theoretical understanding of the small ELM regimes including whether or not they would be applicable to ITER.

In the Type II ELM and Grassy ELM regimes, ELM energy loss is reduced to an acceptable level for reactor scale tokamaks. Type II ELMs often appear as precursors to much larger Type I ELMs on DIII-D and other tokamaks. On ASDEX-U pure Type II discharges were obtained in near double null conditions at somewhat elevated collisionalities. From DIII-D results it is clear that the Type II regime is affected by nonaxisymmetric error fields (Fig. 2-12). In particular pure Type II discharges were obtained with the RMP I-coil in odd parity at higher collisionality. In contrast to fully RMP ELM suppressed

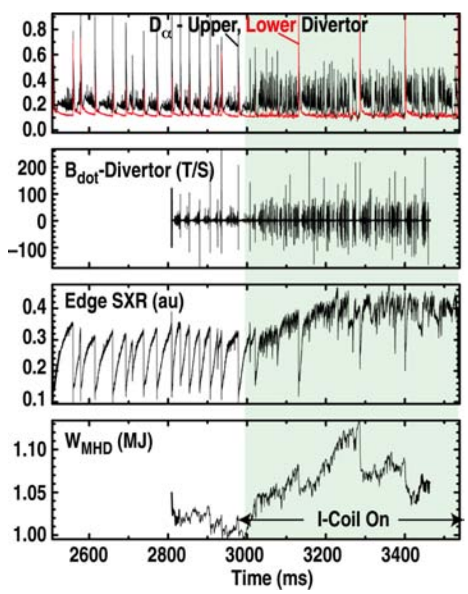

Fig. 2-12. Type II ELMs are enhanced while frequency of larger Type I ELMs decreases with application of RMP (I-coil) in higher collisionality discharge (ν_{*e}~1) DIII-D discharge.In other discharges there are no Type I ELMs for the duration of the RMP.

discharges at lower collisionality these discharges showed no degradation of the pedestal pressure relative to the Type I regime. With odd I-coil parity there is very little resonant magnetic perturbation, perhaps accounting for the lack of effect on the profiles, but large nonresonant magnetic field perturbation. A interaction between this nonresonant field and the peeling-ballooning mode might explain the enhancement of the ELMs. The dependence on error fields may connect the DIII-D and ASDEX-U results since edges of near double null shapes can more easily be made stochastic by nonaxisymmetric magnetic perturbations. Grassy ELMs are obtained on JT-60U at higher poloidal beta and reduced co-current rotation with low collisionality. A strong dependence of ELM size on plasma rotation was observed in this regime.

2.2.6.2. Small ELMs: Research Elements. In future experiments on DIII-D the I-coil will be used to explore the effect of nonaxisymmetric fields on access to the type II ELM regime. The ELITE, BOUT, and NIMROD codes, along with DIII-D's complement of edge profile diagnostics will then be used to build a theoretical understanding of this regime.

DIII-D is also well positioned to study the effect of rotation on ELM size due to the recent realignment of one of the NB lines to the counter-current direction. Rotational effects on peeling-ballooning instabilities have been included in the ELITE and NIMROD codes, and comparison with experimental results will be done to try to provide and understanding of this regime.

A study of the structure of the ELMs in these regimes coupled with a modeling effort to understand the instability mechanism and the reason for the small energy loss will be done in order to determine the applicability of these regimes to future tokamaks.

2.2.6.3. Small ELMs: Hardware Improvements. The research described here can be done largely with the existing hardware. The proposed set of inner-wall resonant magnetic perturbation coils will provide additional flexibility in determining the effect of nonaxisymmetric fields on Type II ELMs and other small-ELM regimes.

2.3. NEOCLASSICAL TEARING MODE STABILIZATION

2.3.1. Present Status

Neoclassical tearing modes (NTMs) will be the principal stability limit on performance in the baseline Scenario 2 operation of ITER [Hender 2007]. The m/n=3/2 and 2/1 islands, for example, destabilized and sustained by the helically perturbed bootstrap current, can reduce confinement and thus beta; worse, the drag on plasma rotation by eddy currents induced in the resistive vacuum vessel wall can bring the plasma rotation to a stop with concomitant loss of the high confinement H-mode, and, yet worse, disruption [La Haye 2006].

Either the 3/2 or the 2/1 NTM can be pre-emptively avoided from ever occurring in DIII-D with sufficient precisely aligned continuous wave (cw) co-ECCD [La Haye 2005, Prater 2007]. A peak current density j_{eccd} of about 3/5 of the local bootstrap current j_{boot}, with full width half maximum δ_{eccd} of about 6/5 of the "marginal" NTM island width (twice the local ion banana width) is adequate; this continuous pre-emptive ECCD is comparable to what is needed and available in ITER [La Haye 2007a] provided that alignment is sufficiently good. The m/n=3/2 stable operation in DIII-D at global parameters similar to ITER is shown in Fig. 2-13. The state-of-the art real time magneto-hydro-dynamic (MHD) equilibrium reconstruction , using the MSE measurements of the profile of the local magnetic field pitch angle, can locate rational surfaces to an accuracy of $\Delta R \approx 1$ cm which is good enough in the major radius R=170 cm DIII-D. This is shown in Fig. 2-14 for one of the discharges in Fig. 2-13 in which beta is stably increased in time by about 30%.

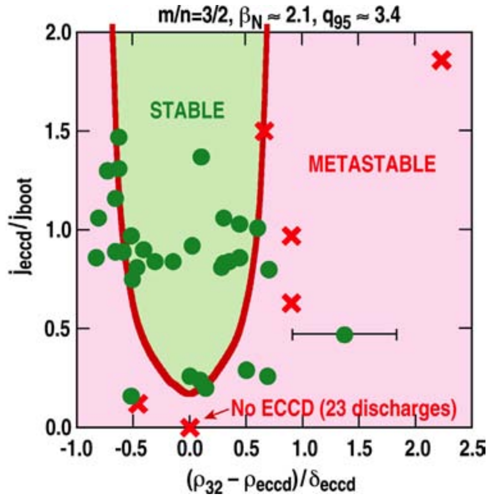

Fig. 2-13. NTM stability versus relative amplitude and radial alignment of ECCD. Active tracking in DIII-D with real-time MSE EFIT to locate q = 3/2 accurately and R_{surf} adjusted to keep aligned with ECCD. All pre-emptive ECCD discharges are evaluated at every potentially destabilizing ("seeding") sawtooth crash. ● (green circles) are stable and ✕ (red x's) are 3/2 mode destabilized.

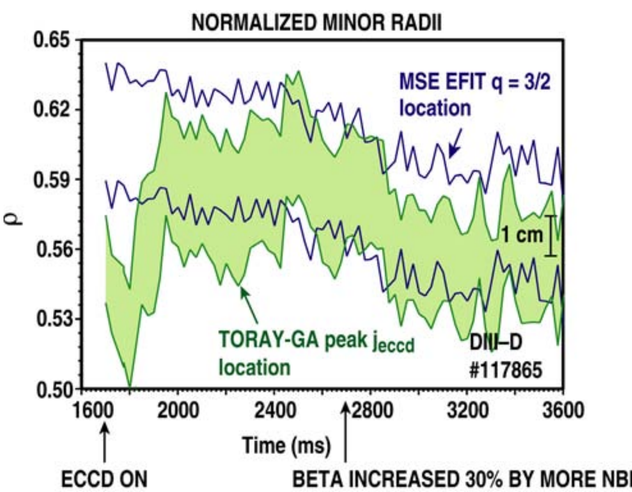

Fig. 2-14. DIII-D uses pre-emptive ECCD with real-time equilibrium reconstruction for alignment on q = 3/2 as beta is stably increased from β_N = 2.14 (t = 2700) to 2.80 (t = 3550). The blue (q = 3/2) and green (j_{eccd}) bands are the one sigma uncertainties.

For broader ECCD and existing islands of full width w ($w/\delta_{eccd} \gtrsim 5/3$) the effectiveness of continuous ECCD can be lower, and modulation of the current drive has been shown to improve stabilization of the 3/2 NTM in ASDEX-U [Maraschek 2007]. Very preliminary recent experiments in DIII-D for the 2/1 mode indicate that there is no advantage with modulation, but this needs further investigation.

When plasma rotation is reduced using a mix of co and counter beams, tearing stability (without ECCD) becomes worse in several regimes, i.e., sawteething H-mode, the hybrid scenario, and the advanced tokamak [La Haye 2007b]. An example is shown in Fig. 2-15 from a hybrid scenario discharge in which the amplitude of the preexisting m/n=3/2 NTM increases as the rotation is reduced with beta kept constant (not shown) by beam feedback; the rotation becomes low enough and the mode amplitude high enough that locking to the resistive vacuum vessel wall occurs with concomitant loss of H-mode. DIII-D has shown that growing m/n=2/1 modes that are going to lock or do indeed lock can be detected and used to trigger a new phase of control; a slowly rotating applied n=1 field from the "I-coil" inside the vessel has been used to grab and entrain such a proto-locked mode and spin it up, at least to a frequency of the order of the inverse n=1 vessel wall time [Volpe 2007].

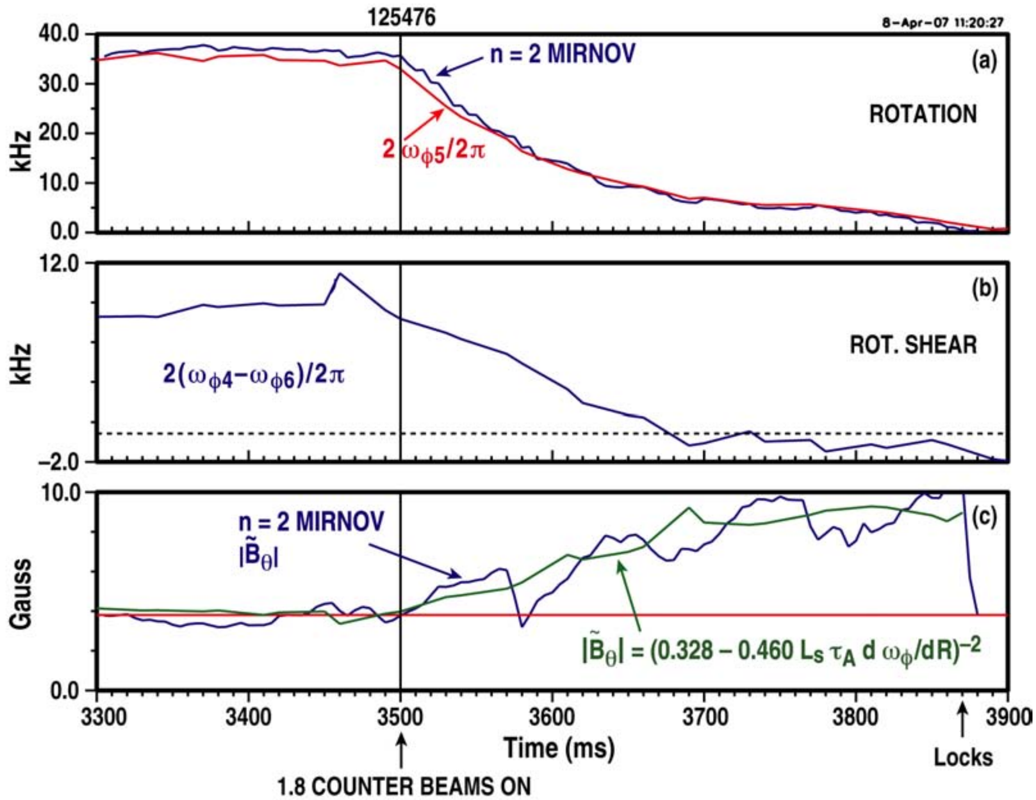

Fig. 2-15. m/n = 3/2 Mirnov amplitude increases as rotation (a) and rotation (flow) shear (b) are decreased by mixing in counter beams. A model for the effect of flow shear on tearing agrees well (correlation = 0.93) up to the time where the 3/2 island locks to the resistive vacuum vessel wall (c).

2.3.2. Planned Research

ITER presents several new challenges for NTM stabilization. ITER will of course be much larger, and with less applied torque will have slower rotation than most, if not all, existing tokamaks. ITER will rely on ECCD stabilization or mitigation of NTMs to achieve the baseline Scenario 2 parameters. An alignment acccuracy of $\Delta R \approx 1$ cm will be needed in the R = 620 cm ITER [La Haye 2007a]; this is about a factor of three smaller <u>relative</u> accuracy than in DIII-D which is arguably the best in the world. Modulation of the ECCD on the island O-point, if necessary in ITER, makes another control issue, that of phase detection and synchronization, in addition to that of radial alignment. The slower plasma rotation and smaller rotation shear at rational surfaces may remove the stabilizing effect of flow shear that occurs in existing tokamaks; this would make NTMs less classically stable [Coelho 2007], thus increasing the demands on the ECCD system for stabilization. So far, no existing tokamak, including DIII-D, has simultaneously stabilized both the 3/2 and 2/1 modes. If this can be done, with sufficient ECCD power sources and dual alignment on both rational surfaces, will other modes appear? As typically one dominant NTM occurs at a time [La Haye 2002], the next unstable NTM could be the benign m/n=4/3 mode near the axis. However, if ITER at q_{95} = 3.1, has the 3/1 or 5/2 modes destabilized, these would also have to controlled by ECCD putting a further strain on dividing limited ECCD resources. Tearing modes that lock to the resistive wall and/or resonant error fields cause loss of H-mode and can lead to disruption. Therefore, a means to mitigate them, keep them rotating if only slowly for a "soft landing", or spin the plasma back up to H-mode is highly desirable for ITER; internal coils in ITER are under study for other purposes but may be available for this if implemented.

The planned research on NTM control in DIII-D for ITER includes: (1) definitive comparison between modulated and continuous ECCD stabilization of the m/n = 2/1 mode with relative current drive widths and beta comparable to ITER, (2) simultaneous control of both the 3/2 and 2/1 modes, (3) control of the m/n = 2/1 mode in plasmas with both low rotation and low q_{95} ($\gtrsim$ 3) as in ITER, and (4) detection and active mitigation of an incipient locked mode so as to prevent it locking and then to remove it, thus avoiding disruption and/or recovering the high performance plasma.

Issues to investigate are: (1) real-time NTM phase determination for gyrotron modulation in an environment with other MHD activity as noise; (2) real-time ECCD mirror steering for alignment of multiple launchers on two different rational surfaces; (3) control of m/n = 2/1 modes in plasmas with low central rotation and low q_{95}, where the 2/1 mode occurs at larger ρ and lower frequency; and (4) entrainment of an n=1 mode by an applied rotating n=1 field so that it does not lock to the wall, and is instead removed by modulated ECCD with appropriate synchronization. In general, DIII-D will evaluate the methodology of NTM control by ECCD for ITER including the limits of real-time alignment, means of phase detection, and means of lock mode control/mitigation for disruption avoidance.

2.3.3. Hardware and Diagnostics Needed

DIII-D has, or will have, the resources and capabilities to address these issues for ITER. The DIII-D operation in 2009–2013 will demonstrate the integrated ECCD control of multiple NTMs in low rotation plasmas, as in ITER, with improved real-time accuracy of q surface location for alignment. The state of the art PCS should be able to do multiple mode alignment and phase synchronization for ECCD control; this will develop the control solutions that can be transferred to ITER. The following paragraphs briefly describe what will be done with existing and new resources.

The new counter-beam MSE view added to the three co-beam views allows greater accuracy in determining the radial electric field Er correction to the MSE data. With this and additional radial locations, more accuracy will be possible in locating rational surfaces. The ultimate accuracy of real-time location of rational surfaces will be pushed to the limit.

The new, as of 2006, combination of 5 co and 2 counter beams, or vice versa with reversed IP, allows reducing the plasma rotation to near zero to study the effects of reduced flow shear on NTM stability and ECCD stabilization. The existing observations can now be focused into an experimental plan for elucidating the relative effects of flow and flow shear, direction of flow, etc. Input for rotation requirements in ITER will be obtained.

With more gyrotrons, 6 at the minimum for $\gtrsim$ 2 MW injected at q = 2/1 and $\gtrsim$ 1 MW injected at q = 3/2, and real-time steerable launchers for dynamic alignment at two different rational surfaces, simultaneous control of both the 3/2 and 2/1 NTMs will be performed to: (1) demonstrate that it can be done, (2) show that the moving of multiple mirrors for alignment, as will be done in ITER, is feasible, and (3) investigate what other NTMs arise and what their consequences are, particularly at low rotation and low q_{95} as in ITER. The DIII-D PCS will be developed to best identify multiple modes at the same time including the separate phase and frequency, with rejection of "cross-talk" from each other and from other MHD events such as sawteeth, fishbones, and/or ELMs; this will allow study of simultaneous modulated ECCD control.

The unique to DIII-D I-coil will be used to study locked mode control or avoidance of locked modes by applying rotating n=1 fields. In conjunction with modulated ECCD this will be developed as a possible means of recovery from such events, if they occur, to re-establish the high confinement H-mode.

2.4. DYNAMIC ERROR FIELD CORRECTION AND RWM CONTROL FOR ITER

2.4.1. Present Status, Motivation

In order to meet the goal of demonstrating steady-state operation with a fusion gain of Q≥5 [Shimada 2007], ITER will likely have to operate at β values above the ideal MHD no-wall stability limit set by the n=1 kink mode (Fig. 2-16) [Hender 2007]. At these values of beta the finite conductivity of the wall slows the growth of the ideal kink mode and gives rise to the slowly growing resistive wall mode. The stabilization of the n=1 RWM is, therefore, a prerequisite for ITER operation above the no-wall limit. Suggested stabilization strategies are passive stabilization by sufficiently high toroidal plasma rotation and active feedback control using nonaxisymmetric coils. Recent experiments on DIII-D have also pointed out the increased sensitivity of high beta plasmas to nonaxisymmetric fields and highlighted the importance of error field correction [Garofalo 2007].

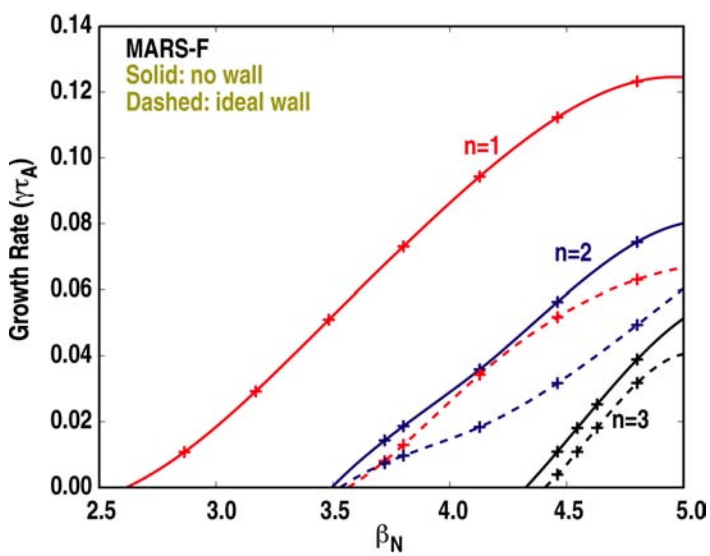

Fig. 2-16. Ideal MHD growth rates of the n=1, 2 and 3 kink modes without a wall (solid) and in the presence of the (ideal conducting) ITER double wall (dashed) for ITER scenario 4 [Liu 2007].

The most desirable way to stabilize the RWM is through sufficient toroidal plasma rotation. Experiments in DIII-D and JT-60U have shown that the RWM can be stabilized by relatively modest values of plasma rotation corresponding to less than one percent of the Alfvén velocity [Reimerdes 2007a, Takechi 2007]. While it is thought that kinetic effects [Bondeson 1996, Hu 2004] are responsible for the observed stability at low rotation, present kinetic models fall short of a quantitative description of the experiment [Reimerdes 2007b] and cannot be used for a reliable prediction of the rotation required in ITER.

If rotational stabilization fails, an active RWM feedback system is required for any operation above the no-wall β-limit. Owing to the significant uncertainties in both the expected rotation and the rotation required for RWM stabilization, ITER presently relies for RWM feedback stabilization on an externally mounted nonaxisymmetric coil set that was originally designed for error field correction. RWM feedback modeling using the VALEN code indicates that the proposed EFCCs are in principle capable to stabilize slowly growing RWMs up to β_N~3.2 (ITER scenario 4 taking into account only a double vessel wall) [Bialek 2005]. However, there is a concern of excessive power requirements and AC heating of the superconducting coils caused by the finite amplitude noise. The present modeling is also optimistic, since it neglects various conducting structures, such as blanket modules, toroidal field coil casings, etc., which separate the coils from the plasma. Modeling of improved RWM observers and advanced RWM feedback algorithm promises to improve the feedback performance and decrease power requirements [Katsuro-Hopkins 2007].

Accurate correction of nonaxisymmetric error fields is crucial for reliable tokamak operation. It has long been known that error fields can excite locked (nonrotating) tearing modes in low-beta plasmas or contribute to the locking and growth of a rotating tearing mode [ITER 1999b]. More recent experiments have shown the increased importance of error fields to the stability of high beta plasmas, where the presence of a resonant component, such as the externally applied n=1 field in discharge 118715 in Fig. 2-17, leads to an increase of the plasma rotation required for stability over discharge 127941 with good error field correction [Garofalo 2007]. The weakly damped RWM amplifies the field error through a process known as “error field amplification”, which results in a braking of the rotation and can lead to the onset of an instability [Strait 2007]. Open questions are how this braking process translates into a high-beta error field tolerance for ITER and if better error field correction could further reduce the low RWM stability rotation thresholds obtained in DIII-D and JT-60U even further.

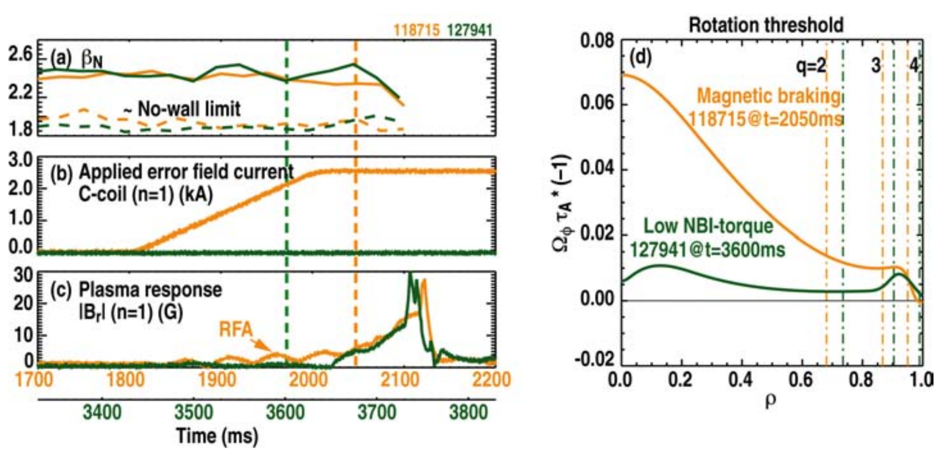

Fig. 2-17. Comparison of two wall-stabilized high beta discharges (a) with an externally applied n=1 error field 118715 (orange) and good error field correction and low NBI torque 127941 (green) (b). In the discharge with the externally applied error field the n=1 mode becomes unstable (c) at significantly higher rotation across the entire profile than in the discharge with error field correction (d).

The plasma's enhanced sensitivity to error fields at high beta can be turned to advantage through "dynamic error field correction," a technique, which has become well established in DIII-D experimental work [Garofalo 2002], Fig. 2-18. Here the large (but stable) mode amplitude is detected and used as input to a feedback system that applies an opposing nonaxisymmetric field. While the mode detection and the feedback control scheme are the same as for active stabilization of the RWM, dynamic error field correction only requires a feedback system that responds on the slow time scale of the discharge evolution rather than the faster time scale of an unstable, growing RWM.

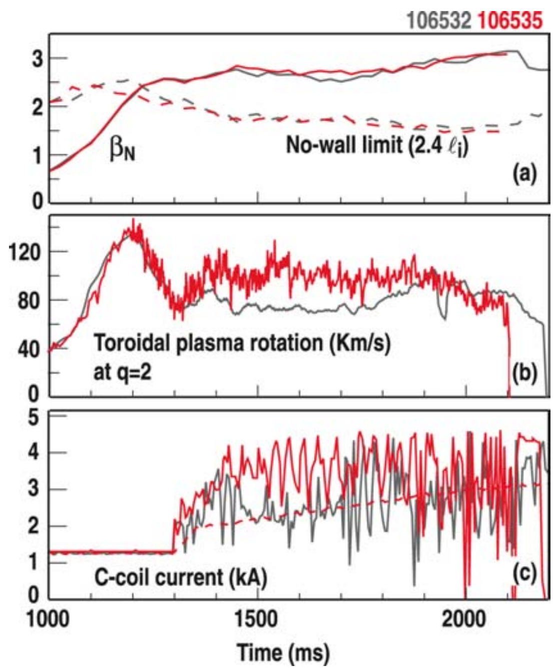

Fig. 2-18. Active feedback in two discharges that exceed the no-wall stability limit (a) using the RWM sensor starting at 1300ms dynamically determines error field correction currents in the C-coils (c). The resulting current of discharge 106532 is used as an offset in discharge 106535. The improved error field correction manifests itself in higher toroidal plasma rotation (b) [Garofalo 2002].

2.4.2. Research Elements for 2009–2013

DIII-D research in the next five years will investigate the physics of error fields and RWM stability at high beta. Scientific progress in this area is crucial to predicting instability thresholds in ITER and their dependence on plasma rotation and on the amplitude and harmonic spectrum of the error field. DIII-D research will also address a number of practical issues of importance to ITER, including error field and RWM detection and control algorithms for dynamic error field correction and RWM feedback.

DIII-D will continue to investigate the physics of RWM stabilization by rotation, in comparison with several recent theories that are currently being implemented in stability codes. Dynamic correction of error fields to very low amplitudes and the new capability to control the rotation to very low values by mixed co- and counter-beam injection form the key to these studies. The plasma stability will be probed by active MHD spectroscopy, allowing detailed comparison of damping rates and mode rotation frequencies to predictions [Reimerdes 2005]. This research will clarify the relevant time scale for the control of nonaxisymmetric fields in the ITER high beta scenario.

Another key issue is the physics of the beta collapse caused by an error field-driven RWM. DIII-D research will investigate the possible role of driven islands in the rotation decrease, subsequent beta collapse, and its implications for active stabilization. This work will include the effects of various poloidal and toroidal spectra of the error field on the rotation decay and instability onset. The onset of the RWM may also be influenced by coupling to edge-localized modes (ELMs) and other transient instabilities. An understanding of the interaction of externally applied and plasma generated magnetic perturbations with plasma rotation and RWM stability will permit to quantify the high β error field tolerance and rotation requirements in ITER.

In the same way that DIII-D's flexible coil system is well suited to study the effects of error fields, it will also be used to develop dynamic error field correction strategies for ITER. The capabilities and limitations of ITER's EFCC will be simulated with DIII-D's external C-coils, while the possible advantages of a more closely coupled set of control coils and a variable harmonic spectrum will be investigated using the I-coils. The capability to correct the crucial components of the error field to very low amplitude will relax ITER's requirement for rotation.

Direct RWM feedback control is likely to remain an element of the ITER design, until rotational stabilization can be predicted with greater confidence. Therefore DIII-D research for the next five years will also support the development of RWM feedback control for ITER. As for dynamic error field correction DIII-D's C-coils allow to test an RWM control system similar to the system presently planned on ITER in ITER like plasmas. Further bandwidth limitations can be simulated in the control software. In addition DIII-D's I-coils can be used to evaluate proposed designs for internal coils and weigh in on the decision whether to install such a coil set. Once the coil design is fixed the research effort will focus on validating control models and optimizing the control algorithm for the chosen configuration.

Dynamic error field correction as well as direct RWM feedback control can benefit from an optimized detection of plasma deformations, which measures the nonaxisymmetric perturbations with high sensitivity. This research will include improvements on the magnetic detection systems now in use and the exploration of nonmagnetic detection methods using SXR detector arrays or electron cyclotron emission (ECE) radiometers, which are potentially more sensitive to the internal response of the plasma. Included here is also the development of advanced algorithms such as Kalman filters to discriminate against unwanted signals in the mode detection system, including other MHD modes as well as true noise in the instrumentation. Minimization of noise input to ITER's control system is particularly important for RWM feedback with external control coils in order to meet the power requirements and minimize ac heating of structures near the control coils.

2.4.3. Diagnostics and Hardware Improvements

In order to address the outlined research elements, several upgrades of diagnostics and power systems are necessary. The research would also greatly benefit from proposed additional midplane I-coil segments, the reduction of the magnetic field error from the 30 deg B-coil feed and an increase of the FW power.

Proposed diagnostic upgrades will improve physics studies, error field correction and direct RWM feedback control. The proposed upgrade of the magnetic diagnostic system for 3D equilibrium reconstruction (described elsewhere) will also provide extensive data for improved mode discrimination and detection of higher-n modes. Alternatively, the addition of four poloidal field probes to the existing midplane array can accomplish many of the same goals. Proposed new multiple-input integrators with adjustable balancing of the inputs will improve the signal to noise ratio and rejection of the axisymmetric equilibrium fields. In addition to magnetic detection internal sensors, which directly measure the nonaxisymmetric perturbation of flux surfaces, will be developed. A proposed upgrade of the toroidal array of SXR detectors (new dust shields, radiation hard photodiodes, improved amplifiers and flexible filter wheels), described in Section 8, will permit the development of an RWM sensor for feedback control, which promises a reduction of RWM feedback currents. Further insight into the mode structure including the role of island formation is obtained from one or more additional ECE arrays at different toroidal locations.

A further reduction of the error field will be beneficial for the study of error fields and crucial for RWM feedback control in very slowly rotating plasmas. A significant improvement can be achieved by reducing the error caused by the 30 deg B-coil feed (either through a redesign of the feed or through a local correction coil) as proposed in Section 10.3.1. Residual error fields can be corrected through improved C- and I-coil correction schemes and will likely require the control of the current in individual I-coils. The possibility of controlling individual I-coil segments will also gain in importance with further integration of control techniques, such as additional edge transport control for the suppression of ELMs.

While a maximum current of 1.5 kA in individual I-coil segments, achievable with the existing switching power amplifiers (SPAs) (Table 2-4), is expected to be adequate for error field correction, there will be too little margin for direct feedback control. Since the presently used Techron 7782 audio-amplifier (AA) can only be configured in up to eight parallel amplifiers, their maximum current is limited to 1.5 kA. A solution based on the existing amplifiers will, therefore, require a crossover network:

Option 1. Crossover network with sub-SPAs driving 1.5 kA at low frequencies and 48 (24 exisiting and 24 new) Techron 7782 AAs arranged in quartets driving 750A at high frequencies.

Alternatively, solutions based on new amplifier types can be persued in order to increase the current capability for 12 independent circuits:

Option 2. Crossover network with sub-SPAs driving 1.5 kA at low frequency and 24 new, high current AA systems (single amplifiers or multiple paralled amplifiers), which due to the crossover network can have a lower cutoff frequency, capable of driving up to 750A at high frequencies.

Option 3 A set of 12 new high current amplifier systems driving up to 2.5 kA from DC to high frequencies.

Table 2-4
I-coil Connections and Corresponding Current Limitation for the Present Set of 24 Audio-Amplifiers (AAs) and Four Switching Power Amplifiers (SPAs)

I-coil Connections	Number of Circuits	Toroidal Mode Numbers (n)	Poloidal Mode Spectrum	Maximum Current (A) – AA (Fast)	Maximum Current (A) – SPA (Slow)
Helical quartets	3	1 and 3	Fixed	1500	4500
Odd-n pairs	6	1 and 3	Variable	750	1500
Helical pair	6	1, 2 and 3	Fixed	750	1500
Single coil	12	1, 2 and 3	Variable	375	1500

The DIII-D facility is already equipped with flexible sets of nonaxisymmetric coils. The adequacy of ITER's external error correction coils for RWM feedback control can be evaluated using DIII-D's six external C-coils, which are similar in number and location. While the limited spatial coverage of ITER's proposed midplane port-plug coils can be approximated using a subset of I-coils, this work would benefit from the installation of six midplane I-coil extender segments (see option 3 in Sections 2.2.2. and 10.7.1). Such a configuration results in a better coupling of the applied field to the RWM, Fig. 2-19. It also increases the flexibility in simulating proposed ITER coil designs.

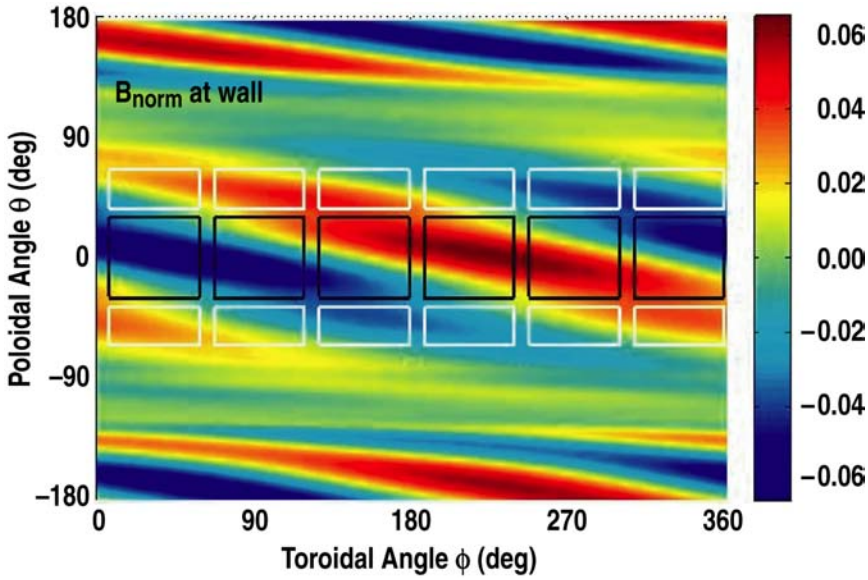

Fig. 2-19. Existing I-coils (white) and proposd midplane extender segments (black) can provide a good coupling to a typical perturbed magnetic field pattern of an n=1 RWM calculated with the MARS-F code for DIII-D discharge 127838.

The research described above would also benefit from an increase of torque-neutral heating power, such as an increase of the ECH power to >12 MW, outlined in Section 10.4, and of the FW power to 6 MW, outlined in Section 10.6. Presently low rotation plasmas, which significantly exceed the no-wall stability limit, can only be obtained at reduced toroidal magnetic field with repercussions for current profile control and ECE measurements. An increase of the torque-neutral heating power would greatly enhance our ability to access and sustain operating regimes, where the RWM is unstable.

2.5. PELLET FUELING

2.5.1. Present Status

The pellet fueling system on DIII-D is a repeating pneumatic injector with three independent barrels. Two of the barrels produce slow 1.8 mm pellets at <= 7 Hz for transport through curved guide tubes to reach the plasma from the inner wall or top of the vessel. The third barrel is a higher speed gun that is primarily useful for outside midplane injection.

Experiments have been carried out in the past five years to study pellet fueling deposition physics, pellet ELM interaction, pedestal modification, and H-mode threshold modification. It has been found that inner wall pellet fueling is much more efficient than conventional outside midplane injection and produces smaller ELMs. For these reasons inner wall injection is planned for fueling of ITER. The present fueling rate of inner wall pellets on DIII-D is equivalent to 45 Torr-L/s of gas. This fueling rate has been found to be insufficient to produce a significant change in H-mode pedestal density and pressure conditions despite the deep efficient fueling. The large ELMs that are triggered by LFS injected pellets have motivated the development of the pellet dropper for ELM pace making studies (Section 2.2.5). This system, which will be fully operational in 2008, will only provide ~30 Torr-L/s of very shallow particle fueling and is therefore not useful for fueling studies.

The U.S. has been chosen to supply the pellet injector system for fueling and ELM mitigation in ITER. This effort provides an exciting opportunity with both many rewards and challenges in both technology and physics to ensure successful completion.

2.5.2. Research for the Next Five Years

Experiments in the next five year plan with enhanced fueling capability are centered around high repetition rate pellet injection from the inner wall for density buildup in ELM mitigated scenarios. In addition we will explore the operation of AT plasmas at high density using this capability. An enhanced fueling capability with higher repetition rate pellets combined with the pellet dropper operation will provide a valuable test of the combination of fueling and ELM pace making that is planned for ITER, with an injection geometry very similar to ITER (Fig. 2-20).

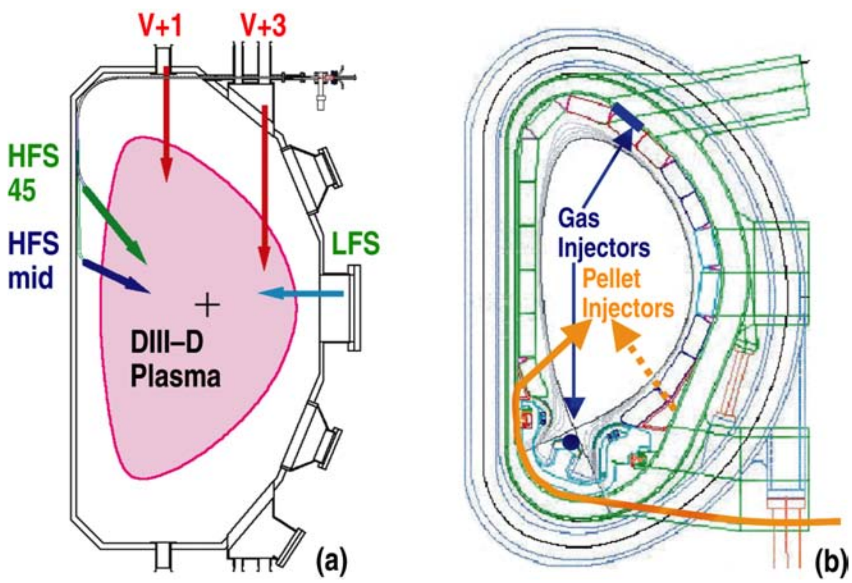

Fig. 2-20. The inner wall injection geometry in (a) DIII-D and (b) ITER are very similar.

A significant portion of the work planned in the next five years involves a physics demonstration that pellets can provide central fueling and ELM control for an ITER-like plasma without inducing unwanted instabilities. Challenges include showing that: (1) central fueling is obtained with normalized pellet penetration comparable to that expected in ITER, (2) instabilities such as the 2/1 tearing mode are not induced in plasmas with q_{95} near 3.0 as planned for ITER, and (3) pellet injection nearly tangential to the separatrix can provide suitable ELM control.

In order to control the normalized pellet penetration in DIII-D to be similar to ITER, smaller pellets and/or a hotter edge are required. The hotter edge is the most desirable solution, but with the available edge electron heating capability, we may be limited to a single pellet where it may be more difficult to track the particle flow. With smaller pellets it will be possible to put in a string of pellets without cooling the edge too much.

The question of pellet induced instabilities, particularly on the q=2 surface, can be done with the present hardware, since plasmas with $q_{95} \approx 3$ can be run routinely. It is simply a matter of ensuring the appropriate diagnostics are available, and a suitable experimental plan has been developed to adequately explore the possibilities.

For tangential pellet injection to induce ELMs, the pellet dropper being installed now is suitable. For higher speed, larger size pellets, the dropper can be removed, and the former injection line to the pellet injector can be re-installed. The biggest challenge will be developing a plasma target whose separatrix is close to tangent to the V+3 port injection line at R = 2.1 m.

2.5.3. Hardware Improvements for the Next Five Years

A higher fueling rate from the inner wall is desirable for pedestal modification physics studies and for simulating operational scenarios for ITER. Larger pellet size is one method to increase the fueling rate, but it would be likely to be too perturbative. A more desirable way to increase the inner wall fueling rate is to increase the repetition rate of the existing size pellets by improving the repetition rate of all the guns.

The ITER gas gun pellet injection system must be capable of 10 Hz operation with low propellant gas flows. The technology to do this for steady state is currently under development. We propose to use this development to update the DIII-D pellet injector with higher repetition rate capability and a third slow gun for inner wall fueling. This will enable increasing the inner wall fueling rate to approximately 80 Torr- L/s. Also steady state capability can be achieved on the DIII-D pellet injector with the addition of a screw extruder. A screw extruder prototype is presently under development for ITER and this will be used on DIII-D for one of the guns to provide steady-state 10 Hz operation.

In addition, to demonstrate acceptable central fueling in ITER, it may be necessary to reduce the pellet size from the existing guns to achieve normalized pellet penetration similar to ITER. The basic question is to see how small we can make the pellets without compromising the reliability of the injector.

Pellet acceleration by absorption of microwaves has been proposed as a means to achieve high-velocity injection from the inner wall of ITER [Parks 2006]. This approach may allow an increase in pellet velocity without the risk of fragmentation in the guide tube. A proof of principle test is planned in DIII-D during the next five years.

2.6. ITER STARTUP SCENARIOS

The ITER superconducting coil geometry and double layered vacuum vessel place unique constraints on the plasma initiation and rampup phases, particularly for AT and hybrid scenarios. Rampup scenarios to enter the burn phase at plasma current flattop with the desired target configuration must be developed and will require modeling and model validation using present day tokamaks. For example AT discharges in present day tokamaks operate with $q_{min} > 1$ and have no sawteeth, and the ITER AT startup scenario must be designed to achieve this condition at the beginning of the burn phase.

The relative distance of ITER's poloidal field coils from the plasma is greater than in most existing tokamaks, creating greater challenges in maintaining vertical stability during rampup and rampdown.

Experiments were started in 2007 to demonstrate some aspects of the ITER baseline startup scenario. In particular, plasma current rampup at constant q_{95}, with discharges limited on the LFS DIII-D poloidal limiters, was successfully demonstrated. During this rampup, the limited discharge was successfully diverted to the ITER shape at a time scaled from the ITER baseline scenario. In these experiments, ℓ_i was significantly higher than values assumed for ITER, leading to a vertical instability (Fig. 2-21). This indicates the importance for continued modeling to determine acceptable trajectories for ITER. Results from these experiments have already led the ITER team to examine alternative ITER startup scenarios, and one scenario has been tested on DIII-D.

This new startup scenario, compared with the baseline scenario in Fig. 2-21, uses a larger bore (higher plasma volume) just after initiation, diverts earlier to reduce LFS limiter heating, and then applies feedback for a phase of constant ℓ_i during the rampup. In addition sawteeth, while not eliminated, are delayed. DIII-D has developed ℓ_i feedback control using dIp/dt as the actuator in order to evaluate the effects of ℓ_i using this new startup.

ITER's operating requirements, including lower ℓ_i during startup, control of heat flux to the limiters, and vertical stability, approach the limits of the poloidal field system and thus may require improvement of the capabilities of the poloidal field coils and their power supplies.

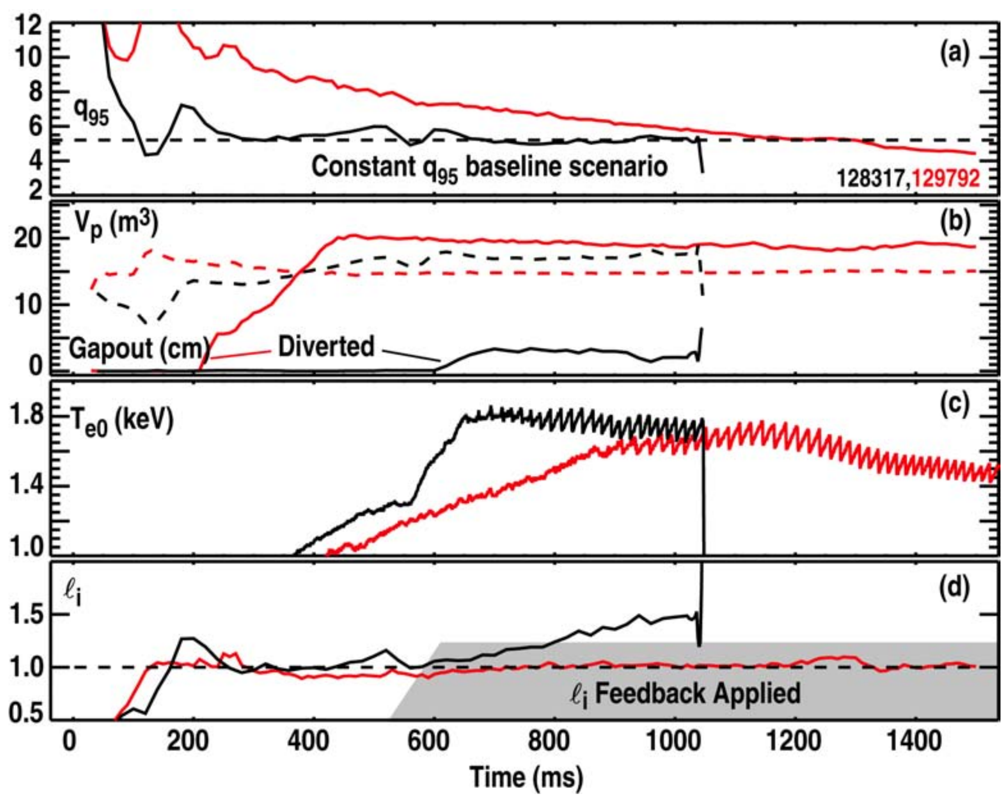

Fig. 2-21. Two ohmic scenarios for ITER startup demonstrated in DIII-D: one with constant q_{95} (black) and one with an earlier diverted phase and constant ℓ_i (red). The figure includes time evolution of (a) q_{95}, (b) plasma volume and gap to outer limiter, showing transition to the divertor shape, (c) central electron temperature, showing sawteeth, and (d) internal inductance ℓ_i.

2.6.1. Research Elements for 2009–2013

DIII-D research in the next five years will focus in two areas: modeling of ITER startup scenarios and validation of these models in the DIII-D tokamak. The goal is to develop models that can be applied to ITER with a high degree of confidence in order to produce scenarios for reliable and robust plasma initiation and startup with the required initial condition at the start time of the burn phase.

The modeling will use present codes such as CORSICA, TRANSMAC, and ONETWO. We expect that enhancements to these codes will be necessary as the modeling proceeds. This work will be done in close collaboration with the USBPO, ITPA groups, and the ITER central team. We will also work closely with other U.S. laboratories. For example, we are already working with Lawrence Livermore National Laboratory (LLNL) in modeling with CORSICA.

The DIII-D experimental work will focus on validating the predictions from the startup modeling and developing successful startup scenarios for ITER. In particular we plan to address the following issues:

- Low voltage inductive startup, ≈ 0.3 V/m
- ECH heating (the recent ITER scenario is compatible with our present ECH second harmonic system)
- ITER shape simulation
- ITER PF coil geometry and power supplies
- Low ℓ_i startup scenarios for AT and hybrid scenarios
- Heat flux to the 3 LFS limiters (and uniformity between the limiters)

2.6.2. Diagnostics and Hardware Upgrades

The DIII-D facility is quite versatile and most of the capabilities are available to carry out these studies. However the following enhancements will allow a thorough assessment of the issues in the last section.

1. Fast IR camera viewing the divertor region.
2. Visible and IR cameras to view the three poloidal limiters.
3. Operation without the vertical field inductor (VFI) constraint on the DIII-D coil system. This will allow better simulation of the ITER coil set and also produce plasmas with shapes closer to the ITER shapes during startup. (Section 7.4.2)
4. Real time thermocouples at and around the outer midplane and the three DIII-D poloidal bumper limiters.

We note that fundamental EC gyrotrons are planned for ITER. The DIII-D second harmonic EC gyrotrons can provide pre-ionization and heating during burnthrough and the current ramp for the large bore ITER startup scenario.

2.7. DISRUPTION MITIGATION, CHARACTERIZATION AND AVOIDANCE

Disruptions represent a potential limitation to the lifetime of ITER's plasma-facing components. Here we discuss DIII-D research plans for methods to mitigate the effects of disruptions by a pre-emptive fast shutdown of the discharge (Sections 2.7.1 to 2.7.3) and to characterize disruptions and their effects in order to improve predictions of disruptions in ITER (Sections 2.7.4 to 2.7.6). Of course, the best solution would be to avoid disruptions entirely. Plans to develop reliable means of disruption avoidance through plasma control are discussed here (Sections 2.7.4 to 2.7.6) and also in the chapter on plasma control (Sections 7.3.7 and 7.3.8).

2.7.1. Disruption Mitigation – Status

In the previous decade, disruption-related experiments on DIII-D have focused primarily on understanding the physics of massive gas injection (MGI). MGI is a potentially attractive scheme for fast tokamak shutdown because (1) it is relatively inexpensive and straightforward to implement and (2) it has, in many tokamaks, demonstrated reduced divertor heat loads and vessel forces. For example, experiments on DIII-D have demonstrated that MGI-induced shutdown leads to reduced vessel forces (≈5x) and reduced divertor heat loads (≈10x) when compared with natural disruptions [Taylor 1999]. Figure 2-22 shows divertor heat loads measured across the DIII-D lower divertor using IR thermography during various types of disruptions. Plotted heat fluxes are averaged over the thermal quench (TQ) duration. It can be seen that neon MGI has much lower divertor heat loads than typical disruptions, e.g., the current-limit disruption, Fig. 2-22(b).

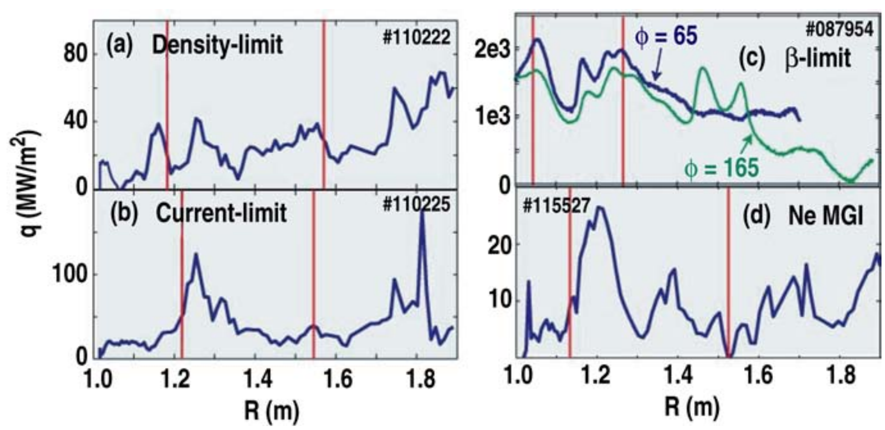

Fig. 2-22. TQ average divertor heat loads measured during different types of disruptions: (a) density-limit, (b) current-limit, (c) β-limit, and (d) neon MGI shutdown. Note the wide variation of vertical scales.

As a result of a variety of dedicated DIII-D experiments, understanding of the physics of MGI shutdowns has greatly improved in recent years. Neutral stopping at the plasma edge is observed over a wide range of target plasma conditions ranging from low energy W_{th} = 0.02 MJ Ohmic discharges to high energy W_{th} = 1 MJ H-mode discharges, demonstrating that neutral stopping at the discharge edge is very robust [Hollmann 2006, Granetz 2007]; these results are consistent with the low ram pressure neutrals being stopped by the high ablation and magnetic pressures at the plasma edge [Parks 1998]. Inward radial diffusion of resulting impurity ions has been shown to be relatively slow, of order $D_\perp \approx 1\ m^2/s$ [Hollmann 2005]. Additionally, the central role of low-order MHD, in particular the (2/1) and (1/1) modes, in enabling rapid radial thermal transport during the TQ was confirmed in q-scan experiment, where target plasmas with different q-profiles were shut down with argon MGI. A clear increasing delay in the onset of the TQ was observed for higher q_{95} target plasmas where the q=2 surface was buried deeper in the plasma [Hollmann 2007]. The current channel spreading (I_p spike) amplitude measured in these experiments at the end of the TQ was consistent with ergodization (due to reconnection during the large TQ MHD) out to $q \leq 2$.

Mixing efficiencies of MGI injected impurities are observed to be fairly low, of order 5% for high-Z gases like neon or argon [Fig. 2-23(a)] and somewhat higher, of order 10%–20% for low-Z gases like helium or hydrogen [Hollmann 2008a]. Mixing appears to occur dominantly during the TQ; i.e. impurities injected during the subsequent cold current quench (CQ) arrive too late and are not assimilated effectively into the current channel.

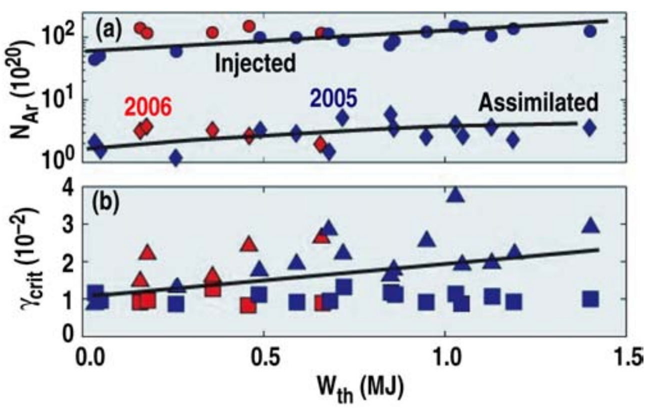

Fig. 2-23. 2005 and 2006 MGI experiments showing (a) injected and assimilated number of argon atoms from argon MGI by middle of CQ and (b) runaway electron suppression parameter as a function of initial thermal energy W_{th}.

Impurity mixing during MGI is important for achieving collisional suppression of runaway electrons (RE). This requires very high core densities e.g., $n_{tot} = n_e + n_B/2 \approx 10^{16}$ cm^{-3} for DIII-D, where n_e is the free electron density and n_B is the bound electron density. In DIII-D, sufficient core densification has not been achieved [Fig. 2-23(b)]. Here, $\gamma_{crit} = E_{crit}/E_\varphi$ where $E_{crit} = 2\pi e^3 \ln \Lambda\ (2n_e + n_B)/mc^2$ is the critical electric field for generation of a RE avalanche [Rosenbluth 1997] and E_φ is the 0-D toroidal electric field estimated from the measured decay of the plasma current: $\gamma_{crit} > 1$ is required to ensure suppression of REs.

A conceptually simple way to improve the neutral delivery rate and γ_{crit} is to increase the number of gas valves. This was attempted in 2007, using a “Medusa” flange consisting of 6 MGI valves, shown in Fig. 2-24. By firing many (5-6) valves simultaneously, record free electron densities of $n_e \approx 10^{15}$ cm^{-3} were achieved. However, RE suppression parameters were still well below unity, with $\gamma_{crit} \approx 0.1$. Additionally, the large amount of injected gas (>1000 torr-l) was found to cause operational problems in subsequent discharges.

In summary, achieving $\gamma_{crit} = 1$ with MGI is extremely challenging because (1) the long delivery tube necessarily limits the impurity delivery rate, (2) neutrals are robustly stopped at the plasma edge, (3) impurity ion mixing appears to be relatively slow except during the brief TQ phase, and (4) the huge required amount of injected gas causes operational problems in subsequent discharges.

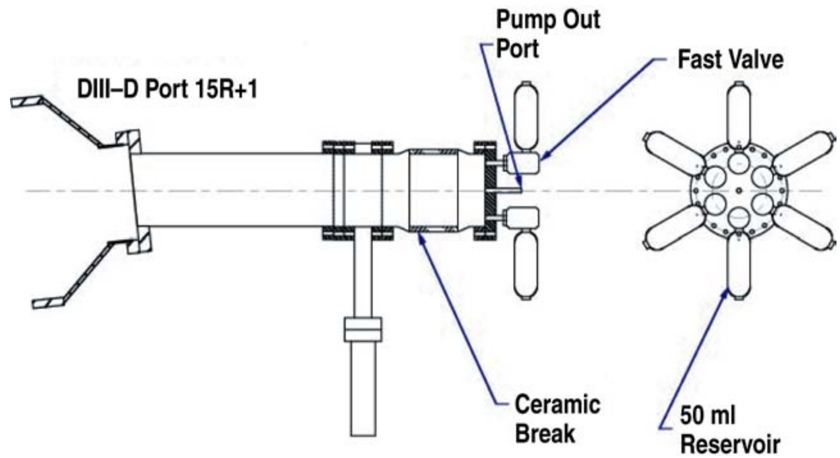

Fig. 2-24. Medusa flange of six valves used in 2007 MGI experiments.

2.7.2. Disruption Mitigation – Research Plans

Presently, it is thought that the halo current and heat load reduction seen in MGI will scale well to larger tokamaks. However, RE generation and amplification is an outstanding concern because the longer CQs in future tokamaks will allow more time for RE amplification to occur via avalanching than in present devices. Because of this larger RE amplification, even the small ($I_{RE}/I_p < 1\%$) RE levels seen in present MGI experiments could be amplified to significant levels in a larger tokamak (e.g., ITER with a 10 MA current and 50 ms CQ time).

Many aspects of RE in disruptions are poorly understood. For example, present estimates indicate that disruption CQ parallel electric fields in DIII-D are ≈100x too small to generate seed RE via the standard Dreicer mechanism, despite frequent indications of RE. Also, the prediction of RE avalanching during disruptions has not been tested experimentally, nor has the level of magnetic fluctuations $\tilde{B}/B$ necessary to prevent RE formation.

One of the main future goals of disruption mitigation work at DIII-D is the development of an impurity delivery technique which can achieve collisional RE suppression $\gamma_{crit} = 1$ in DIII-D, and to extrapolate this technique, as well as possible, to ITER. This should, in principle, be possible to achieve: Fig. 2-25 shows predicted (0-D current channel modeling [Hollmann 2008b]) values of γ_{crit} as a function of injected particles for hydrogen, carbon, and argon. A standard H-mode discharge and ideal (delta-function) deposition of impurities directly into the current channel is assumed. It can be seen that the direct injection of about 10^{23} particles (less than 1 g) of H atoms should be sufficient to achieve unity γ_{crit}.

In order to address these issues, the major elements of DIII-D research in disruption mitigation for the next five years will include:

- Validation of the predicted RE multiplication during the CQ
- Physics understanding of impurity and heat transport during massive gas injection, leading to validated models for extrapolation to ITER
- Development of techniques to deliver sufficient impurities for collisional suppression of REs
- Exploration of alternative means of RE suppression

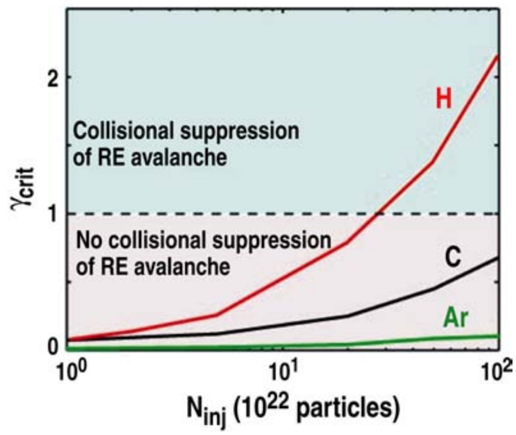

Fig. 2-25. Predicted values of γ_{crit} versus number of injected particles in DIII-D for ideal (delta function) deposition of impurities directly into current channel.

2.7.3. Disruption Mitigation – Hardware/Diagnostics Needed

Since MGI is presently still the method of choice for disruption mitigation of ITER, it is important to continue MGI research. From a hardware prespective, is is planned to use the existing "Medusa valve" for the bulk of the experiments. However, one new valve which might be tried is the 2 cm diameter TEXTOR eddy-current valve. The advantage of this valve is that it has a very clean opening pulse and is not affected by DC magnetic fields (unlike solenoid valves), so it can be mounted very close to the machine.

In addition to continuing MGI work, it is planned to investigate several alternate fast impurity delivery schemes on DIII-D. The first of these is the "inverse jet" (I-jet) proposed by P. B. Parks [Parks 2007a]. Instead of starting with a long empty tube connected to a distant supply tank (as is done in MGI), we use a long cylindrical tube pre-filled with gas at high pressure. The tube is connected to a fast valve (and small secondary reservoir) at one end and the outlet (plasma-facing) end is sealed off with a thin diaphragm or rupture disk. The advantage of this design is that when the disk bursts (after opening the fast valve), the outflow rate rises immediately from zero to the full rate, so the plasma will be heavily fueled while the plasma discharge is still in the thermal collapse phase.

A potentially effective modification to rupture disk MGI would be to include dust with the gas pre-fill. High-speed dust injection is interesting because increased penetration (over neutral gas) might result, but (unlike fast pellets) no potential first wall damage is anticipated from small, fast dust grains. Also, dust size and composition could be tailored to achieve optimum core deposition. Some drawbacks of rupture disk MGI are duty cycle (rupture disks are single-use items and must be replaced after each MGI shot) and the possibility of metal shards going into the plasma. The possible shard issue will be carefully investigated in bench tests at ORNL before implementation of rupture disk MGI on DIII-D.

Another alternate fast shutdown scheme, being investigated by ORNL for implementation on DIII-D in the next several years, is large (D ≈ 1 cm) pellet injection. Previously, small (<10^{21} particles) cryogenic neon and argon pellets were studied for disruption mitigation in DIII-D [Taylor 1999]; however, this research path was discontinued because of the large amounts of RE generated. Now, a large "pipe gun" is being developed which will be capable of firing large cryogenic pellets (>10^{23} particles) – this should be able to deliver sufficient particles in a delta-function manner for collisional RE suppression. A concern with such a large pellet is optimizing core ablation: it is possible that the pellet will pass entirely through the plasma due to a self-cooling effect and damage the opposing wall tiles. To avoid this, the possibility of using sharp delivery tube bends or a target plate to break up the pellet into ice shards or a chain of smaller pellets (similar to a liquid jet) will be investigated.

Another approach which will be investigated for addressing the pellet ablation problem is coated pellets. These refer to a variety of possible designs using pre-fabricated solid pellets consisting of various layers of materials chosen to optimize core impurity deposition. For example, a thin coating of high-Z material such as tungsten could protect the pellet during its trajectory into the core; then a soft low-Z interior (such as Li) could be exposed and ablate rapidly, depositing most of the pellet mass in the core region. The primary goal here is delivery of the pellet interior into the plasma core before the pellet-induced cold front destabilizes core MHD and causes the core thermal collapse. Interesting alternate ideas for the pellet core composition include high-pressure gas [Evans 2007] or dust [Parks 2007b]. These pellets will be fabricated at the GA Inertial Confinement Facility (ICF) pellet facility and designed in close collaboration with the GA theory division, which is developing modeling capabilities for variable-Z pellet ablation. Coated pellet injection will be done using the GA lithium pellet injector, which is being re-installed on the vessel in 2008. Key physics issues to study include pellet shell ablation rates for different Z materials, and plasma cold front generation and propagation by different Z shells.

Finally, it is possible that even the most innovative impurity deposition schemes will not be able to achieve $\gamma_{crit} = 1$ in DIII-D (or ITER). To investigate noncollisional methods of RE suppression, we plan to perform experiments on the destruction of RE confinement using resonant magnetic perturbations (RMPs). RMP application using the existing internal I-coils in DIII-D (generally in a n=3 configuration) has been successful in ELM suppression experiments. Because RE are thought to be extremely sensitive to stochastic fields [Harvey 2000], RMP application might also be used to destroy RE confinement; preliminary indications of magnetic perturbations ruining core RE confinement have been seen in JT-60U [Yoshino 2000].

An overview of different fast shutdown schemes possibly to be attempted (or already attempted) in DIII-D are listed below. Liquid jets are listed for completeness, although it is expected that one (or some combination) of the other, more easily implemented techniques will be sufficient for ITER:

- **Small cryogenic pellets** (1995–1997 DIII-D experiments)
  - Advantages: nearly 100% impurity assimilation, proven technology, good halo current and divertor heat load mitigation.
  - Disadvantages: create large RE seeds.
- **Massive gas jet** (1998–2007 DIII-D experiments)
  - Advantages: straightforward to implement, good halo current and divertor heat load mitigation.
  - Disadvantages: poor impurity assimilation – most gas goes into pumps and NB ducts and can cause problems in subsequent discharges.
- **Large cryogenic pellets** (next five years in DIII-D)
  - Advantages: expected to have good core penetration, could deposit enough particles into core for avalanche suppression.

- — Disadvantages: new technology, need shatter plate to avoid wall damage, impurity deposition profile in plasma uncertain.

- **Custom solid pellets** (next five years in DIII-D)
  - — Advantages: easier to implement than cryogenic pellets, could be designed for optimum core impurity deposition.
  - — Disadvantages: possibility of wall damage, impurity deposition uncertain.
- **Inverse jet** (next five years in DIII-D)
  - — Advantages: rapid impurity delivery without possibility of wall damage.
  - — Disadvantages: possibility of small shards in machine, requires rupture disk replacement between shutdowns, core penetration uncertain.
- **Liquid jet** (possibly within next five years in DIII-D)
  - — Advantages: probably good core penetration without possibility of wall damage.
  - — Disadvantages: new difficult technology, possible heavy load on vacuum system.

In addition to more experiments and modeling, improved understanding of disruptions and fast shutdowns will depend on improved disruptions diagnostics. Because of the fast time scales, machine vibrations, and large radiation brightness which occur during disruptions, many diagnostics which work well during normal tokamak discharges give poor or unusable data during disruptions. This includes crucial core diagnostics such as Thomson scattering, charge-exchange recombination (CER), and MSE.

A variety of diagnostic improvements useful for disruption work are planned for DIII-D. Presently, a new set of SXR arrays is being brought online. These novel photodiode arrays feature rotatable pinhole/filter wheels allowing the arrays to be used for a variety of fast radiated power measurements, such as high-T_e bremsstrahlung measurements or fast radiated power tomography. A diagram, filter wheel photo, and preliminary fast radiated power data from this diagnostic is shown in Fig. 2-26. A possible upgrade to this diagnostic under investigation is the use of Ross filters to isolate e.g., argon K_α emission for tracking RE emission during argon MGI shutdowns. Assuming this SXR diagnostic upgrade is successful, similar upgrades are planned for the three other SXR arrays located at other toroidal locations.

Another presently ongoing diagnostic upgrade which will be very valuable for disruption work is a fast polarimeter for electron density measurements. Unlike the present CO_2 interferometers in DIII-D, the polarimeter is not susceptible to fringe hops during disruptions. Preliminary data for this diagnostic has already been obtained during disruptions in 2007.

In 2006, a fast-framing visible camera was commissioned on DIII-D, primarily for disruption studies. This camera has since provided useful data for disruption, ELM, and turbulence studies. Presently, this camera has a relatively limited tangential view of the midplane. Future work is planned to open up the present view port to allow an expanded view of the main chamber, and also, to develop periscope and fiber bundle expansions to allow the camera access to other view ports. In some experiments, for example for studying pellet ablation plumes, a different (e.g., vertical instead of tangential) view would be highly desirable for the fast-framing camera.

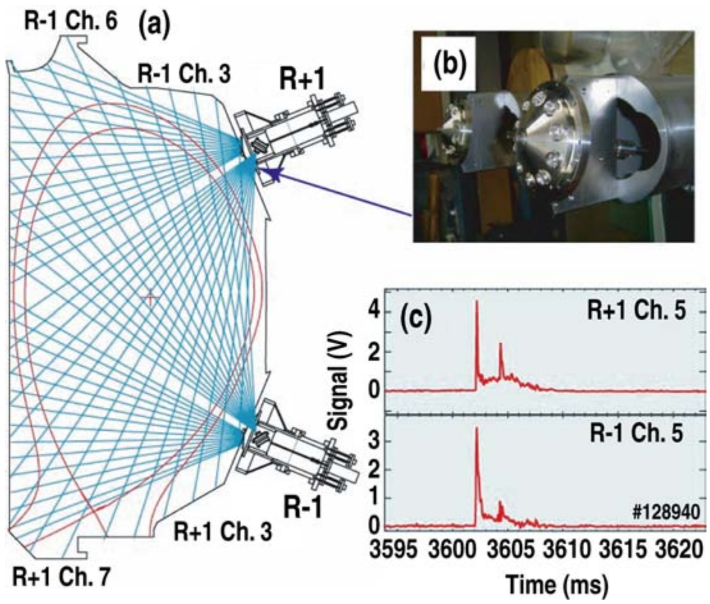

Fig. 2-26. (a) Schematic of new soft x-ray diagnostic showing R+1 and R-1 photodiode arrays; (b) photo of filter wheels; and (c) radiated power data obtained for disruption showing radiation flash in both R+1 and R-1 arrays.

In the longer term, a possible diagnostic upgrade which is being considered is narrower polychromators in the core Thomson scattering system. Presently, Thomson scattering is optimized for ~ 1 keV temperatures and provides core n_e and T_e profiles during normal operation. However, during disruptions, this system becomes swamped by background light, especially during the CQ. By narrowing the polychromator bandpass to measure CQ temperatures of ~ 5 eV, this system might be able to get profiles during the CQ phase as well.

An area where diagnostics are found lacking in tokamaks in general (not just DIII-D) is RE diagnostics. Although RE are created in DIII-D disruptions and fast shutdowns, their existence can only be inferred by their effect on the I_p decay or by hard x-ray flashes if RE beams hit the wall, Fig. 2-27. Better RE diagnostics are crucial for understanding RE seed formation, RE avalanche amplification, and RE loss due to collisions or stochastic fields. Presently, two RE diagnostics are planned for DIII-D. The first, which is presently beginning development, is a RE energy diagnostic. This will use small carbon pellets fired vertically from the existing DIII-D lithium pellet launcher into a pre-existing RE beam. By measuring the spatial distribution of the resulting gamma ray flashes, the energy of the RE beam can be deduced. A schematic of this diagnostic is shown in Fig. 2-28. Important physics to be studied with this diagnostic are RE formation and amplification mechanisms.

Additionally, an extreme ultraviolet (EUV) camera diagnostic to measure the RE spatial distribution and transport is being designed. This would involve coupling EUV mirrors and down-converting image intensifiers to couple EUV images into the existing fast-framing visible camera. The advantage of this is that core K_α emission from the RE beam could be imaged so that fast images of the RE structure and position could be obtained. This could enable first studies of RE radial transport and filamentation.

Finally, plans are being made to image RE beam structure and loss using synchrotron radiation imaging [Finken 1990]. This involves a collaboration with TEXTOR involving borrowing their fast-framing IR camera. This IR camera was already loaned to DIII-D in the 2007 run campaign to image divertor heat loads; the main modifications required at DIII-D to image RE synchrotron emission is replacement of the tangential 90R0 glass window with CaF_2 to allow transmission of IR radiation and some CaF_2 lenses to transmit light to the camera.

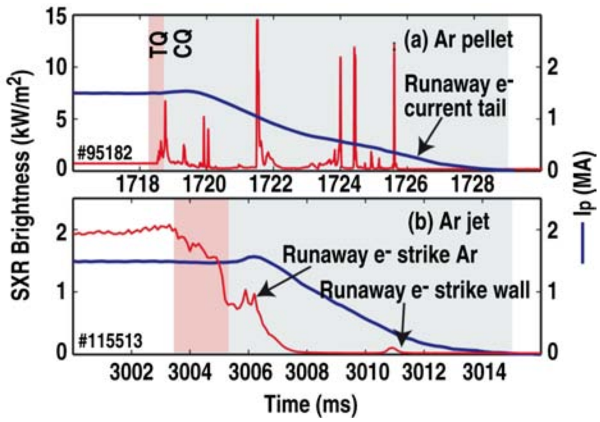

Fig. 2-27. Presence of RE seen in (a) Argon pellet and (b) Argon gas jet experiments by hard x-ray flashes in SXR arrays.

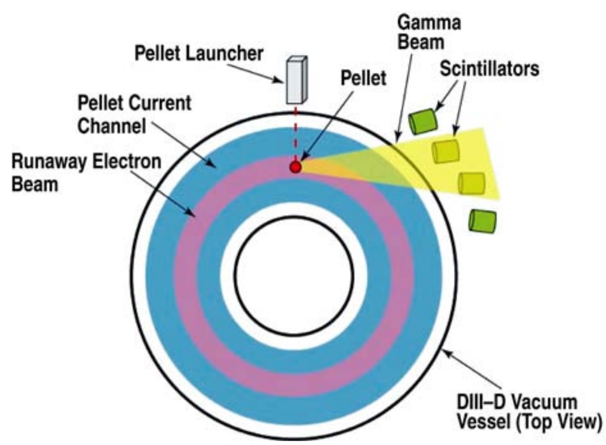

Fig. 2-28. Runaway electron energy diagnostic consisting of carbon pellet launcher and array of fast scintillators.

2.7.4. Disruption Characterization and Avoidance – Status

Characterization of disruptions is important for avoiding the onset of disruptions and for timely detection of disruption precursors if a disruption is imminent. Additionally, being able to predict vessel force and heat load distributions due to disruptions is critical for design of the vacuum vessel and first wall components, since these components might, in a worst-case-scenario, need to withstand several completely unmitigated disruptions in their lifetimes. Disruptions might also affect wall conditioning and subsequent discharge startup. From Fig. 2-22, it is clear that large conducted heat loads can be expected well outside the divertor floor during disruptions. Consistent with this, fast bolometry and probe data in DIII-D suggest that significant plasma heat loads and sputtering can occur at both the inner and outer main chamber walls during disruptions [Hollmann 2003, Hollmann 2005] and are therefore not confined to the plasma divertor. These observations are qualitatively consistent with large main-chamber heat loads seen with IR thermography in MAST [Counsel 2005].

In addition to characterizing the spatial distribution of the disruption heat loads, understanding the magnitude and effect of the large main chamber heat loads is important. Disruption measurements in DIII-D suggest that large quantities of carbon and deuterium are released from the walls during disruptions [Hollmann 2003]; to a lesser extent, this also appears to occur during ELMs [Yu 2007].

Recently interest has focused again on the issue of halo currents, as a result of unpublished reports of nonaxisymmetric halo currents in JET that could extrapolate to radial forces on the ITER vacuum vessel several times larger than previously expected. Previous DIII-D work has shown good agreement between modeling and experimental measurements of axisymmetric halo currents and halo current forces during disruptions [Humphreys 2000]. This work now must be extended to cases with strong asymmetries.

2.7.5. Disruption Characterization and Avoidance – Research Plans

In the future, more accurate spectroscopic and main chamber probe measurements will be performed to attempt to understand the carbon sputtering and deuterium release which occurs during disruptions. A preliminary Divertor Materials Evaluation (DiMES) experiment has already been performed to attempt to measure H_2/D_2 isotope exchange in the divertor graphite surface during a disruption – this experiment showed that significant H_2 appeared to replace implanted D_2 in a single disruption, a promising result for reducing the tritium inventory in ITER’s divertor and other plasma-facing components. Future experiments along these lines will hopefully elucidate some of the disruption plasma-wall interaction physics, such as the role of chemical sputtering and hydrocarbon release during disruptions, which is poorly studied at the moment. Additionally, it is anticipated that some measurements of main chamber heat loads during disruptions will be possible with the fast IR camera to be borrowed from TEXTOR for viewing RE synchrotron emission – here the main challenge is separating main chamber thermal emission with RE emission. If necessary, the two effects can be separated conclusively by reversing the direction of the plasma current. Plans are also underway by LLNL to eventually purchase a fast-framing IR camera which could be used for this purpose also.

Validated models of halo currents are crucial for prediction of the electromagnetic forces on ITER’s vacuum vessel and plasma-facing components, and DIII-D work in this area will continue in the next five years. An improvement which is desirable is the development and validation of current decay and halo current models which do not assume axisymmetry in the plasma current. Also, the inclusion of wall sputtered impurities into the halo current modeling is important for accurately modeling the plasma current decay rates.

One important element of DIII-D research in the next five years will be to develop reliable algorithms that will allow the PCS to detect impending disruptions and avoid them, and to trigger the disruption mitigation system only as a last resort when a disruption becomes unavoidable. Disruptions resulting from causes such as a rotating or locked tearing mode, or a density limit, typically have magnetic or other precursors that can be detected well ahead of the disruption, allowing time for remedial action. However, instabilities at the ideal MHD beta limit often have only a magnetic precursor with a very fast, sub-millisecond growth time. In present disruption precursor detection efforts, e.g., in JT-60U [Yoshino 2003], timely (with 10 ms warning) β-limit disruption detection is achieved less than 10% of the time, compared with over 90% for other types of disruptions. Clearly, improved precursor detection is needed for β-limit disruptions, especially considering the large divertor heat loads that can result from an unmitigated β-limit disruption. Real-time stability analysis, warning of the approach to an ideal stability boundary, is likely to provide the most reliable means of avoiding such disruptions, and we plan to develop this technique in the next five years. Plans to develop disruption avoidance are also discussed in Section 7.3.8.

2.7.6. Disruption Characterization and Avoidance – Hardware/Diagnostics Needed

Improvement of the DIII-D tile current monitors is needed in order to understand halo currents and the resulting electromagnetic loads on plasma-facing components and on the vacuum vessel itself. The DIII-D tile current monitors were disconnected during the 2005 vent divertor shelf work. However, these constitute the key diagnostic for studying halo currents in DIII-D and will therefore be reconnected in a multi-step operation beginning in 2007. For better validation of future modeling of nonaxisymmetric halo currents, greater tile current coverage (poloidal and toroidal) is highly desirable and is being investigated for installation in the next several years.

In the area of disruption avoidance, a variety of diagnostics (some unique to DIII-D) will be used in real time for detection of nonmagnetic disruption precursors. These include fast interferometery, fast polarimetry, fast CER, fast SXR at different cutoff energies, fast bolometry, fast ECE, and fast beam D_α spectroscopy. In addition, we plan to develop real-time calculations of the plasma's stability and proximity to stability limits, using an ideal MHD code such as DCON. A reliable, real-time "kinetic" equilibrium reconstruction, using measured density and temperature profiles as input, is a prerequisite for the stability calculations and therefore must be developed early in the five year period. Implementation of the disruption avoidance techniques outlined here will require a significant effort by both physicists and software engineers. The demands of the real-time stability calculation may also require an upgrade of the PCS hardware.

2.8. PLASMA FACING MATERIALS

2.8.1. Motivation and Status

The choice of the plasma facing materials (PFM) is proving to be one of the most difficult problems for the ITER team. The requirements for the PFM in ITER are much more extreme than those presented in today's tokamaks, due to the high power density from DT fusion and the long pulse, high duty cycle operation. The more extreme conditions in ITER presented to the PFM and their primary effects include:

1. Higher steady state heat flux >>> Over heating
2. Much higher time averaged incident ion flux >>> Erosion
3. Much higher impulsive heat loading >>> Erosion
4. Tritium flux and the potential for tritium erosion >>> Radiological Safety
5. High duty cycle operation also leads to problems with
   — Materials migration and co-deposition >>> Radiological Safety
   — Dust accumulation >>> radiological and explosion hazards

In this section, we will focus on studies of co-deposition, tritium retention, main chamber plasma materials interactions, and material migration and dust. These topics are ITER urgent issues, affecting reliability, safety and licensing. A better understanding of these basic issues, including the detailed mechanisms of tritium uptake through transport processes, codeposition and dust formation, is needed for qualifying a first wall design [Federici 2001].

Closely related topics are also discussed in other sections of this document, including heat and particle flux control (Section 3.8), control of impulsive heat loading by edge localized modes (Section 2.2) and disruptions (Section 2.7), and PFM-related diagnostics (Section 5.6).

On DIII-D we will maintain a strong focus on graphite as a primary plasma facing material. Graphite has been shown to be a very durable material, compatible with long-pulse, high-performance operation [West 2007a]. This focus fits nicely into the context of world-wide tokamak PFM studies, which include high Z refractory metals on Alcator C-Mod (molybdenum) and ASDEX-U (tungsten) and mixed materials (Be, W, C) as the next upgrade of the JET first wall.

During the past decade, laboratory and tokamak studies of the interaction of hydrogen plasma with all promising plasma facing materials have shown that retention of the fueling species is significant. The key problem for the use of graphite in a burning plasma device such as ITER is the retention of tritium in the form of co-deposited layers of fuel ions with redepositing carbon ions that originate from a region of net erosion and are transported to a region of net deposition. Over more than a decade, the DiMES has played a seminal role in the documentation of erosion rates and mechanisms of graphite in the DIII-D divertor [Wong 2007]. DiMES experiments will continue to provide fundamental data on erosion of relevant materials in ITER divertor regimes, such as argon induced detachment.

One promising technique for reduction of the co-deposition of the fuel ions is to increase the temperature of the plasma facing graphite tiles. Laboratory studies and recent data from JT-60U have observed reductions in the H-isotope fraction of co-deposited layers of almost two orders of magnitude at surface temperatures > 700 K compared to room temperature tiles. DiMES experiments on codeposition in tile gaps [Fig. 2-29(a,b)] and on recessed in-vessel diagnostic mirrors [Fig. 2-29(c,d)] have shown a dramatic reduction of codeposited carbon and an even further reduction in the codeposited deuterium as the surface temperature is raised by only a few hundred degrees Celsius [Rudakov 2007, Litnovsky 2007, Krieger 2007].

This year we have initiated a study to determine the feasibility of conducting a high temperature bake in the presence of a few Torr of O_2 to study the efficacy of oxidation in the removal of co-deposited a C:D layers in DIII-D. This is a promising technique for the mitigation of an important component of in vessel tritium in a device with graphite PFM.

Codeposited layers in DIII-D are typically quite thin, a few μm, compared with those that will occur in ITER, of order mm. Recent laboratory data on the oxidation of thick redeposition films is very promising. Thermal oxidation removal rates, (μm/hr), appear to increase roughly linearly with thickness. Thick codeposits can also exfoliate, creating flakes and dust, which opens up the possibility of tritium recovery by mechanical means via collection and removal of the fragments. Thermal oxidation is also likely to be more rapid for fragments. On the negative side, flakes and dust constitute potential hazards, including the risk of causing disruptions. It is therefore important to study thicker codeposits in DIII-D.

The heat and particle flux to the main chamber may be also a serious issue for ITER if the plasma-facing surface there is beryllium due to its relatively low melting temperature and high sputtering rates.

This is of special concern during the breakdown and rampup phase of an ITER discharge when it is limited on the main chamber outer wall.

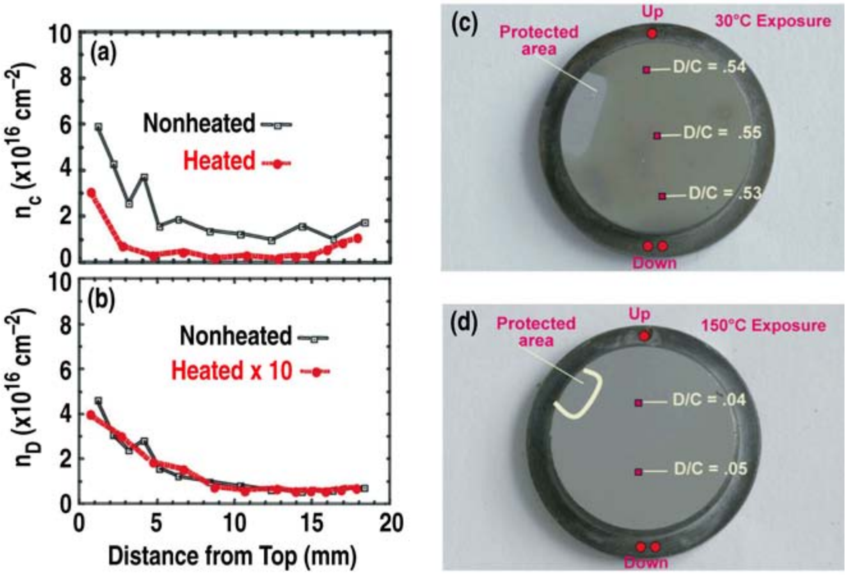

Fig. 2-29. Measured (a) C content and (b) D content in redeposition films in tile gap experiments using DiMES are reduced by an order of magnitude when the surface and gap temperature is increased from room temperature to 200°C. (c) Film deposition is visually apparent on a room-temperature molybdenum mirror placed in a shadowed region of the divertor using DiMES and measured D/C ratio is high. (d) When the molybdenum mirror is heated to 150°C, very little film is deposited and the D/C ratio drops by an order of magnitude.

In long-pulse, high-duty cycle devices such as ITER erosion of PFM from the main chamber and divertor and transport of the eroded material to new wall locations can result in a re-deposition layer of mixed materials with properties, thermo-mechanical and chemical (e.g., fuel ion uptake), significantly different from the underlying plasma facing material. Migration can also lead to the deposition of contamination layers onto surfaces in shadowed regions. The build up of mixed-material codeposition layers is of particular concern in ITER, where three plasma facing materials are planed, beryllium in the main chamber, graphite in the high heat flux regions of the divertor, and tungsten along main chamber/divertor transition.

ITER will be the first device where dust presents a significant safety hazard. A build up of dust from intense plasma/materials interactions, especially from impulsive loads from ELMs and disruptions, and from thermo-mechanical interactions of codeposition layers, is expected due to the high heat throughput and the high duty cycle. Accurate predictions of dust inventory, and removal/mitigation techniques are presently unavailable. Studies of dust, initiated in DIII-D many years ago, will be continued. The first of these studies, conducted by Idaho National Engineering Laboratory (INEL), collected dust from a variety of regions of the first wall soon after a major vent and used standard materials analysis techniques to characterize the size, composition, and deuterium content. About two years ago, studies of dust during plasma operation were initiated using the Rayleigh channels of the Thomson scattering system to detect submicron dust in the SOL and divertor during plasma operation [West 2007b]. Dust transport studies have been initiated by introducing well-characterized dust onto the DiMES probe and using standard and fast-framing video cameras to track dust trajectories and spectroscopy to measure the contamination from the dust.

2.8.2. Research Plans

During the period of 2009 through 2013, we will attack the problem of co-deposited layers of fuel ions with redepositing carbon ions, using a three-pronged approach:

1. Basic erosion, transport and redeposition studies
2. Reduction of the rate of codeposition of deuterium by operation with hot walls
3. Removal of codeposited deuterium by chemical, thermal, and mechanical processes

Each of these approaches will be briefly discussed below.

In the past three years we have begun a set of basic studies of carbon transport and deposition based on the injection of $^{13}CH_4$ [Allen 2005, Wampler 2007]. We will continue these experiments to map the flow of chemically sputtered carbon from various regions of the main chamber and divertor.

We plan to improve the in-situ DiMES temperature control to extend the temperature range substantially. As a part of the DiMES program, a new Midplane Materials Evaluation Station (MiMES) was implemented in 2007. The present station is designed for passive exposure of material samples in the main chamber near the midplane of DIII-D. The goal is to measure erosion, redeposition, codeposition, and implantation of first wall relevant materials. We are proposing a next generation main chamber material study module including a movable bumper limiter, along with instrumentation for diagnostics and sample manipulation. These upgrades are also discussed in Section 5.6.

On DIII-D several techniques appear to be feasible for obtaining higher tile temperatures over large regions of the wall. Raising the ambient temperature of the vessel and tiles to temperatures up to 100°C can be achieved by heating the water flowing in the double layered vessel. Temperatures up to 200°C can be achieved by replacing the water with heated air. Large regions of the tiles can be thermally isolated from the vessel and heated to high temperatures using the heat from the plasma. During a discharge, tile surface temperatures up to several hundred degrees Celsius can be achieved. We will choose between the various options after detailed studies of the engineering issues are completed. Preliminary engineering studies of the thermal and structural issues have been carried out with promising results. A confirmation of the large reduction factor in the uptake of fuel atoms when the plasma facing surface is operating at a temperature close to that expected in a long pulse device would be a very valuable result for future burning plasma experiments.

We will continue to pursue the feasibility of conducting a high temperature bake in the presence of a few Torr of O_2 to study the efficacy of oxidation in the removal of co-deposited aC:D layers. Should this study indicate these experiments can be carried out safely, we will continue this line of research.

As the pulse length and total heating energy is increased during the five year period, we expect codeposits to become significantly thicker on both plasma-facing and plasma-hidden surfaces. The latter will reproduce the tritium codeposition pattern that occurred in the Joint European Torus DT Experiment #1 (JET DTE1) experiment, where most of the tritium retained in the vessel was in the form of codeposits which formed on the plasma-hidden water-cooled louvers located in a pumping duct, but which then exfoliated and dropped as tritiated fragments to the bottom of the vessel. Removal of these thick films and fragments can be studied in-situ, using e.g., baking in a low pressure oxygen atmosphere. Coated tiles can also be removed and sent to surface chemistry laboratories for more detailed studies.

To better characterize the plasma interaction with the outer main chamber wall, we propose to install an advanced main chamber surface station that will incorporate a fast stroking probe, an instrumented materials testing station that is extractable between shots, and a movable bumper limiter, also well instrumented for the measurement of plasma heat and particle flux. A detailed understanding of the heat and particle flux to the ITER bumper limiters and main chamber wall during the relatively long duration plasma startup period is of immediate concern. These data will also help characterize sputtering of main chamber wall material and subsequent migration onto divertor surface, discussed in more detail below.

To study the migration of material from the main chamber to the divertor, we are considering the possibility of adding coatings, e.g., SiC, to regions of the main chamber. Monitoring of the long-term build up of Si in the divertor would yield valuable information on the migration of material due to erosion and transport. The durability of CVD SiC and B_4C coatings on graphite DiMES samples and divertor tiles has been tested in the past with very good results [Buzhinskij 1999].

Dust transport studies, using dust injection from the DiMES probe with fast-framing video cameras to track dust trajectories and spectroscopy to measure the contamination from the dust, will be continued in order to help identify the key formation mechanisms, transport, and rates of accumulation of the dust. The results will also be used to benchmark modeling, such as DustT code developed at University of California, San Diego (UCSD).

2.8.3. Hardware Improvements

To accomplish the goals described above, several hardware and diagnostic improvements are proposed:

- Upgraded diagnostics, sample manipulation, and advanced main chamber surface station featuring a movable bumper limiter
- Improved in-situ temperature control for divertor materials station (DiMES)
- Tests of various surface coatings (e.g., SiC) in the main chamber
- DIII-D operation with heated walls
- Removal of hydrocarbon deposits by oxygen bake

2.9. HYBRID SCENARIO DEVELOPMENT

2.9.1. Status and Goals

The hybrid scenario is a long duration, high performance plasma discharge that has been proposed as a robust operating regime for ITER [Luce 2001]. In its performance characteristics it is intermediate between the standard, high current, ELMy H-mode scenario, and the steady-state AT scenario, hence the name “hybrid”. The hybrid scenario regime is inductively driven, with bootstrap current fraction of 0.3–0.5 and a fully relaxed current profile with $q_0 \sim 1$. Compared to standard H-mode operation, the hybrid mode has a broader current profile that often eliminates sawteeth, further improving overall confinement and removing a trigger for the 2/1 NTM [Wade 2005]. Hybrid mode discharges have demonstrated stationary, high performance operation over many current relaxation times (Fig. 2-30). Projections to ITER based on DIII-D hybrid mode discharges indicate that a pulse length longer than 1 hour should be achievable, with a power gain Q ≥ 10.

The mission of hybrid scenario research on DIII-D is to develop, assess, and qualify candidate high performance, pulsed tokamak regimes for next-generation devices. The corresponding long-term goals are to:

- Provide next generation devices with robust, reliable operating regimes that offer the potential of a substantial increase in performance over the conventional, sawtoothing, ELMy H-mode regime
- Develop a detailed physics understanding of the processes that lead to improved performance
- Convince the worldwide community to adopt the hybrid scenario as the new benchmark in pulsed tokamak performance.

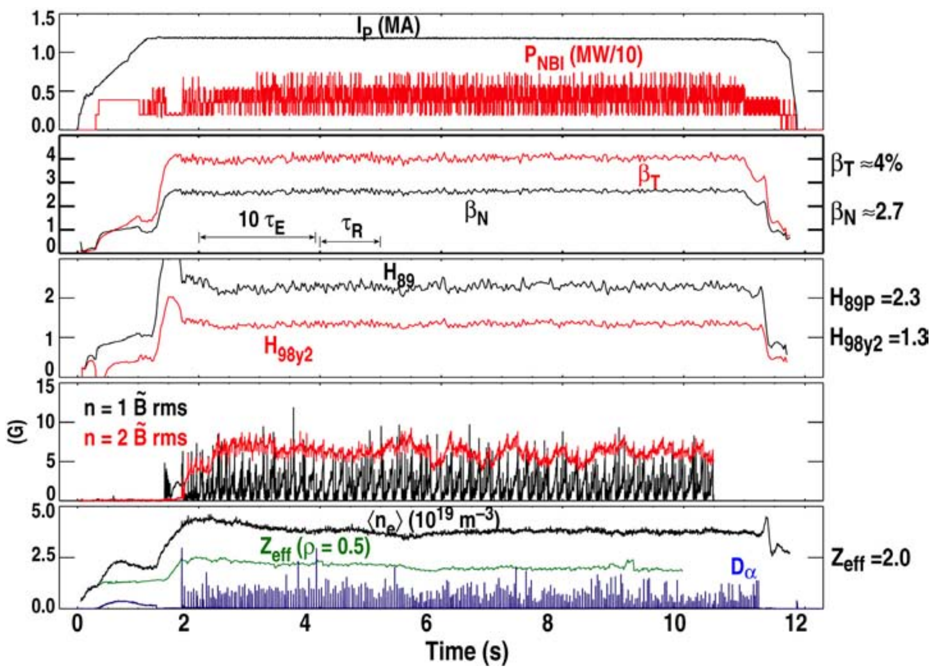

Fig. 2-30. Hybrid mode discharge is sustained for more than nine current relaxation times, with high beta (β_T ~ 4%) and good confinement (H_{89P} ~ 2.3).

The Steady State Operation (SSO) group of the ITPA includes research on hybrid scenarios in several of its joint experimental proposals, to which DIII-D has and will continue to contribute. Experimental research on the hybrid scenario is organized into four areas:

1. Expansion of operating range
2. Validation in reactor-like conditions
3. Improved understanding of important physics
4. Assessment of extrapolation issues.

Initial research on hybrid scenarios, before 2006, concentrated on (1). During the 2006–2007 experimental campaigns, significant progress was made in (2), including studies of low rotation plasmas using balanced NBI, and initial investigation of plasmas with more equilibrated electron and ion temperatures using ECH. Thus, during the next five-year plan for DIII-D, the main emphasis of hybrid scenario research will be on areas (3) and (4), *i.e.*, improved understanding of important physics and the assessment of extrapolation issues, although additional progress is also expected on the topic of reactor-relevance. Specifics regarding the research to be done over the next five years are given below.

2.9.2. Research Elements for 2009–2013

Validation in Reactor-Like Conditions. Most of the hybrid mode discharges to date have been obtained with $T_i > T_e$ and high toroidal rotation, which are plasma conditions quite different from those expected in ITER. Because both of these effects can improve confinement, it is important to assess $T_i \approx T_e$ and low rotation conditions to be able to extrapolate to ITER with more confidence. ECH can be used at either 1.3 T (third harmonic) or 1.9 T (second harmonic) to heat the core electrons to obtain a more reactor-relevant value for T_i/T_e. Based on recent DIII-D experiments, it is estimated that ~6 MW of ECH is needed to obtain $T_i \approx T_e$ for hybrid mode plasmas at 1.3 T, which is comparable to the amount of injected ECH power expected in the next five-year plan. To study hybrid mode plasmas with equilibrated temperatures at higher B_T, the ECH will need to be supplemented by electron heating from the FW system. These rf heating experiments naturally lead to the study of low rotation hybrid mode plasmas owing to the low torque injection, which can be aided by the use of balanced NBI if additional power is needed to obtain high β. The physics to be studied in this area include:

- The dependence of transport and turbulence on T_i/T_e and rotation. Figure 2-31 shows results from recent experiments where the injection of 2.4 MW of ECH at fixed β and Mach number led to a 15% reduction in the confinement factor and a dramatic increase in the long wavelength turbulence amplitudes.
- The dependence of stability limits, especially the onset of the 2/1 mode, on T_i/T_e and rotation.

Improved Understanding of Important Physics. Several important characteristics of hybrid scenarios are not well understood. The hybrid regime has achieved remarkably high confinement factors, especially for co-NBI with strong rotation. While it is well known that rotational E×B shear can suppress turbulence and reduce transport, it is not clear why the hybrid regime should have much better confinement than the standard H-mode regime when using the same co-beams. Possible areas to investigate over the next five years are the effect of the H-mode pedestal height, and the sensitivity to the current profile, which is broader in hybrid mode plasmas compared to standard H-mode plasmas. Since the EC, FW, and NB systems can all drive current, the sensitivity of confinement and stability limits to the current profile can be studied. For example, experiments can determine whether onset of 2/1 NTM (the effective β limit) is sensitive to the exact q_0 value by using ECCD and fast wave current drive (FWCD) used to vary q_0 near 1. Another distinguishing aspect of hybrid mode discharges is the ability of the 3/2 NTM to sustain a stationary current profile with $q_0 > 1$. The resulting sawtooth suppression (when $q_{95} > 4$) is beneficial as the sawteeth can spoil central confinement and trigger the 2/1 tearing mode. However, the mechanism by which the 3/2 NTM maintains $q_0 > 1$ has not been conclusively identified. Proposed mechanisms and the experimental tests to be conducted during the next five-year plan are:

- Poloidal magnetic flux pumping from the coupling between the 3/2 NTM and ELMs. This can be tested by making ELM-suppressed hybrid mode discharges with $q_{95} > 4$.
- Beam ion redistribution by the 3/2 NTM. This can be evaluated using the fast ion d-alpha (FIDA) diagnostics, and rf heated hybrid mode plasmas with minimal fast ion density can be studied.
- Direct counter-current drive by the 3/2 island acting as an Alfven wave antenna. Since this mechanism is dependent upon having a rotational difference between the q=1 and q=3/2 surfaces, the sensitivity of sawteeth suppression to difference rotation profiles (peaked, flat, hollow) can be investigated.

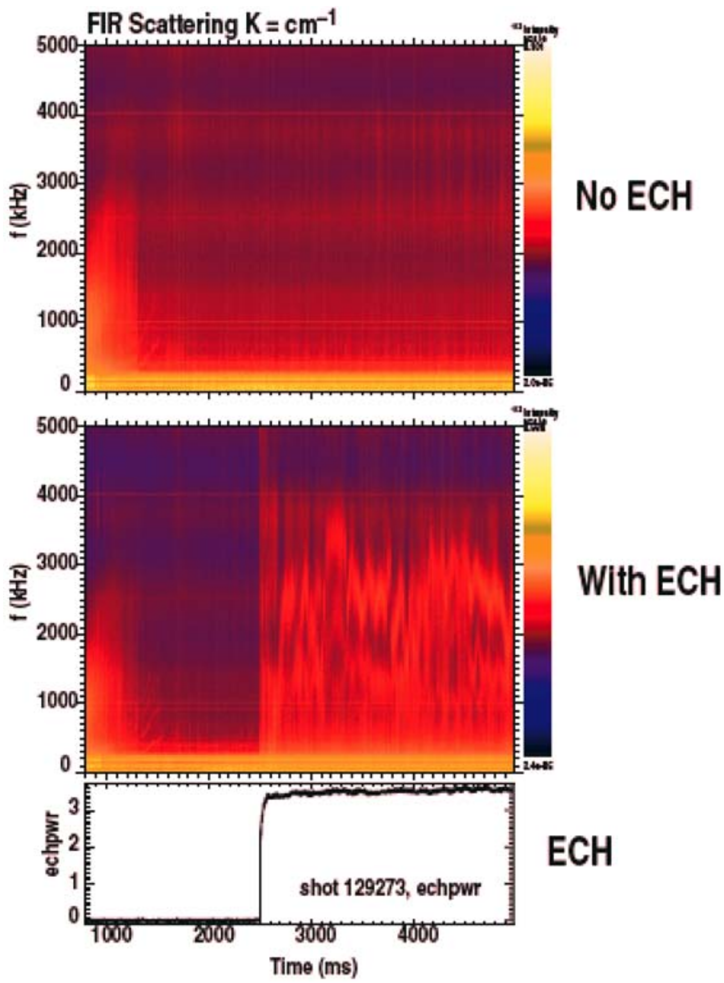

Fig. 2-31. Measurement of long wavelength fluctuations from far infrared (FIR) scattering with and without ECH for hybrid mode plasmas on DIII-D.

Also the current drive systems on DIII-D can be used to compare the characteristics of “natural” and “artificial” hybrid mode plasmas, where the latter are made using off-axis ECCD and on-axis counter-FWCD to suppress tearing modes while maintaining $q_0 > 1$.

Assessment of Extrapolation Issues. All high performance scenarios need to address the accompanying issue of ELM/divertor high heat flux. Research over the next five years on DIII-D will determine whether the high β hybrid scenario is compatible with ELM suppression (or small ELMs) and a radiative divertor. The ELMs can be modified using with the RMP from the I-coil or by counter NBI. Experiments need to determine how the high confinement, high stability properties of the hybrid scenario are affected by these heat flux mitigation techniques, and what the impact is for high fusion performance on ITER. To extrapolate confinement and stability properties of the hybrid scenario to the larger effective size of ITER, ρ^* scaling studies will need to be done. To obtain the smallest possible ρ^* on DIII-D, the hybrid regime will need to be extended to higher I_P and B_T, which will result in higher values of Q_{DD} being obtained. Experiments will also study NTM threshold scaling with ρ^*.

In addition to the experimental research on DIII-D, an important effort will be made over the next five years to continue and expand the hybrid scenario modeling activity. The modeling is done mainly by transport codes such as CORSICA, ONETWO, and TRANSP, which can simulation the evolution of the current, density, temperature, and rotation profiles. This activity will strive to model both the start up phase and stationary phase of hybrid mode discharges, using data from DIII-D as a benchmark, to assess the extrapolation of the hybrid scenario to ITER.

The research on hybrid scenarios in DIII-D over the next five years will benefit from several of the planned improvements to the heating systems and nonaxisymmetric magnetic field coils. The upgrade in the ECH injected power to 6 MW is necessary to obtain hybrid mode plasmas with $T_i \approx T_e$, and the combination of the increased rf power and balanced NBI will allow us to study these plasmas with low torque injection. To obtain the lowest possible rotation rates, however, an increased ability to reduce the error fields in the plasma likely will be necessary, and the improved internal coil set will aid in this task, as will improvements in dynamic error field correction. Finally, the new internal coil set will also likely expand our ability to suppress ELMs using RMPs in high β hybrid mode plasmas, which will increase the relevance of this scenario for ITER.

2.10. HYDROGEN PLASMA OPERATION

2.10.1. Present Status

It is anticipated that the first operational phase of ITER will be carried out in pure hydrogen plasmas. Therefore, a particularly important issue for ITER is the power and/or techniques required for the access to H-mode with pure hydrogen plasmas. Experiments worldwide have consistently shown a strong dependence of various performance parameters (L-H transition power threshold $P_{L\rightarrow H}$, energy confinement, pedestal height) on the plasma species mix. Empirical scalings of $P_{L\rightarrow H}$ indicate that sufficient auxiliary power will be available in ITER to enable access to H-mode with DT operation, but the inverse mass dependence of this scaling suggests that access to H-mode in hydrogen operation will be marginal at best. The inability to obtain H-mode in the hydrogen phase will limit the ability to develop control techniques needed for high Q operation in ITER including ELM and NTM control.

Once H-mode access is achieved in hydrogen operation, other issues related to the ion species become important. Empirical scalings for thermal energy confinement such as the IPB98(y,2) scaling [ITER 1999c] have shown a weak but positive dependence on the ion mass; this is unfavorable for achieving full operating parameters, equivalent to ITER's Q=10 scenario, in hydrogen. Other critical needs for ITER are the capability to control the species mix during DT operation in order to optimize fusion power production, and the capability to change the species mix between pure deuterium and DT.

The effectiveness of resonant magnetic perturbations and other ELM control techniques in hydrogen plasmas is also an open question with important implications for the initial hydrogen phase of ITER operation.

2.10.2. Research Elements for 2009–2013

In the next five years, experiments on DIII-D will be carried out in pure hydrogen, mixed hydrogen/deuterium (H/D), and pure deuterium to assess issues related to ion mass, and to develop potential control techniques for use on ITER. Using DIII-D's capability to produce the ITER shape, a detailed mapping of $P_{L\rightarrow H}$ versus density, toroidal field, and H/D species mix will be developed. As part of this exercise, information will be obtained on

- Dependence of $P_{L\rightarrow H}$ on the ion species mix
- Dependence of the energy confinement time on the ion mass
- Measurement and control of the species mix
- The time required to obtain equilibrated wall conditions as the species mix is changed
- Dependence of ELMs and ELM control methods on the ion mass

The ITER methodology of controlling the species mix using pellet injection will be assessed using the DIII-D pellet injector, and the effects of wall conditioning will be studied. Joint experiments with JET may provide an opportunity to separate ion gyroradius (ρ^*) effects from other effects of the ion mass.

2.10.3. Diagnostic Improvements

An important part of this research effort will be developing a real-time capability to measure the H/D species mix. Ion-ion hybrid layer reflectometry as an ion species mix diagnostic has been successfully demonstrated [Heidbrink 2004] in a proof-of-principle test in DIII-D, and could be developed to provide the input for real-time species mix control. Development of such a diagnostic on DIII-D would also provide valuable information for application of the technique in ITER.

2.11. FAST-WAVE COUPLING TESTS FOR ITER

2.11.1. Present Status

Coupling of FW power to the plasma edge has proven challenging in H-mode plasmas, because of the large antenna-plasma gaps required for high-performance H-mode plasmas and because of the large transient changes in the coupling caused by edge-localized modes (ELMs). Research in both antenna design and operational techniques to improve coupling will be highly relevant to ITER.

A key issue for ITER is how well the FW antenna design will couple to the core plasma. The anticipated ITER antenna [Swain 2007] will use triplets of short straps arranged poloidally and operated in parallel (Fig. 2-32), in order to reduce the maximum voltage at the antenna. However, the present ITER design no longer incorporates capacitors near the antenna current straps to limit voltages in the torus "vacuum" region. The Joint European Torus – Extended Pulse (JET-EP) antenna is ITER relevant with regard to having relatively short antenna strap elements poloidally, but the antenna feed is no longer ITER relevant as it uses capacitors to maintain a relatively low rf voltage in the torus "vacuum". Thus, the ITER antenna design is not being fully tested in any present experiment.

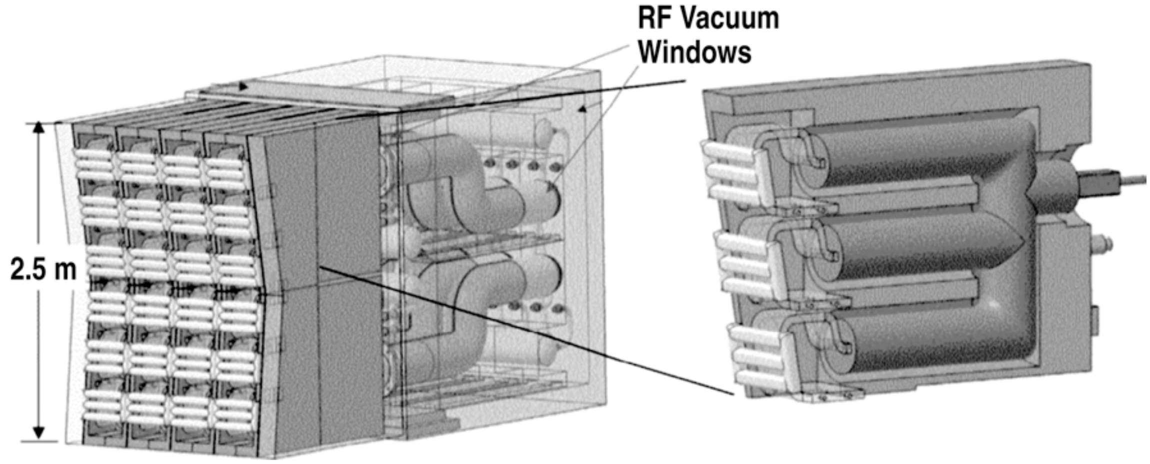

Fig. 2-32. Triplet straps operate in parallel from a single transmission line. The ITER antenna will consist of eight such triplets [Swain 2007, Reprinted from Fusion Engin. Design, Vol. 82(5-14), D.W. Swain et al., "ITER ion cyclotron system: Overview and plans," p. 7, Copyright (2007) with permission from Elsevier.]

2.11.2. Research Elements for 2009–2013

In the next five years, we plan to study and characterize the use of gas injection to raise the density in the scrape-off layer near the antenna. Such injection is being proposed for ITER to augment FW coupling. Coupling experiments in DIII-D are directly relevant to ITER, and successful use of gas injection to augment coupling on DIII-D will give considerable confidence that the use of gas injection will facilitate coupling on ITER.

In addition, we plan to test coupling to H-mode plasmas with an ITER-like antenna. Optimization of coupling with such an antenna on DIII-D and use of the results to benchmark coupling codes will demonstrate the power level that can be achieved on ITER. These results will be even more valuable for ITER if the ITER matching scheme is adopted as well.

In addition to the benefits for ITER, successful operation of the redesigned antenna will benefit the DIII-D program by increasing the power coupled to H-mode plasmas. Over the next five years it should be possible for DIII-D to support antenna and matching development which first provides for delivering the maximum source power available to the DIII-D AT plasmas (Section 10.6.2) and then supports studies of relevance to ITER.

2.11.3. Hardware Improvements

We propose to modify one of the FW antennas in DIII-D to an ITER-like configuration as described above. Changing the present folded strap design to triplets of short straps will yield an antenna that looks like a section of an ITER antenna. As in ITER, the straps would be joined at a high voltage point away from the antenna but still within the torus vacuum, and thus would have the maximum rf voltage along the line in torus "vacuum". Finally, if successful operation is obtained for the DIII-D FW regime, conversion of one of the FW systems to the frequency regime of ITER would be warranted.

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3. ADVANCED TOKAMAK SCENARIO DEVELOPMENT

Building on its roots in the Doublet series of tokamaks, the DIII-D Program has played a leadership role in the global research program on advanced tokamak scenario development. The research program described here defines a path for continuing that leadership role beyond 2013, focusing on the development of both true steady-state tokamak scenarios and stationary inductive scenarios with performance beyond that of conventional H-mode. The scenarios under development are designed to be compatible with high fusion gain in a power plant. The research program includes the integration of solutions for dealing with heat and particle flux in steady state, the demonstration of required control techniques, and the development of a suitable physics basis to allow confident projection and optimization of tokamak designs on the path to a power plant.

A number of the hardware upgrades proposed for DIII-D are essential to the development of the advanced tokamak scenarios described in this chapter. Examples are off-axis neutral beam injection, increased gyrotron and fast wave power, increased pulse length capability, and improved divertor heat handling and diagnostics.

Sections 3.1 and 3.2 provide a brief overview of the mission and plan for advanced tokamak scenario development. The sections that follow provide details, including a description of the four classes of operating scenarios under study and a discussion of the role that the hardware upgrades to the DIII-D facility will play in advanced scenario research (Section 3.3), an outline of development of the physics basis for advanced scenarios (Section 3.4), an overview of discharge control (Section 3.5) and details on instability avoidance and control (Section 3.6), and a description of solutions for handling of steady-state heat and particle flux (Section 3.7).

3.1. MISSION OF ADVANCED TOKAMAK SCENARIO RESEARCH

The goal of advanced tokamak scenario research is to realize the full potential of the tokamak as a source of fusion energy. Two distinct paths for improvement upon the conventional H-mode scenario that was chosen for the reference scenario of ITER are under investigation. One path points toward true steady-state operation of the tokamak while maintaining high fusion gain. This improves the tokamak from the engineering point of view by elimination of thermal cycling, but introduces complexity and reduces flexibility of the operating conditions. The other path retains pulsed operation of the tokamak with its inherent thermal cycling, but seeks to maximize the fusion output with minimal control requirements to reduce the complexity and retain a degree of flexibility in the operating conditions.

The approach proposed for DIII-D advanced scenario research is to provide an existence proof of a plasma that exhibits the characteristics desired for fusion power plant operation under stationary conditions. This is a necessary milestone for the research, but it is not sufficient. In addition to the existence proof, it is necessary to also supply the scientific characterization of these scenarios such that extrapolation to and design optimization of future devices can be carried out with confidence. This characterization cannot be accomplished on the DIII-D tokamak alone; joint research with other tokamaks is critical to gaining and validating the physics understanding of these scenarios.

The DIII-D advanced scenario research program has chosen a clear set of target applications to provide direction to the program. The realization of advanced scenarios on the ITER tokamak is the nearestterm goal. ITER has set operation at Q=10 for 400 s and Q=5 steady-state (in principle) as its top-level physics objectives. ITER has a technology requirement to also maintain the possibility of limited nuclear testing. Candidate advanced scenarios are under development on DIII-D to meet each of these goals. (See Section 3.3 for more details.) A critical step beyond ITER toward a fusion power plant is the operation of a device that demonstrates tritium self-sufficiency. This requires both significant power density and fluence to succeed. Based on preliminary design studies of a Fusion Development Facility (FDF), candidate advanced scenarios for this application are also under development. Finally, the goal of electricity production by fusion energy places very high demands on power density and reliability to reduce the cost of electricity. The DIII-D advanced scenario research program will increase its focus over the next five years on establishing the basis on which the fundamental decisions will be made for a demonstration power plant (DEMO). The three target applications for the DIII-D advanced scenario research program are summarized in Fig. 3-1. Having these clear goals in mind helps to define the necessary research and the relative priorities of the various threads that must come together to reach these goals.

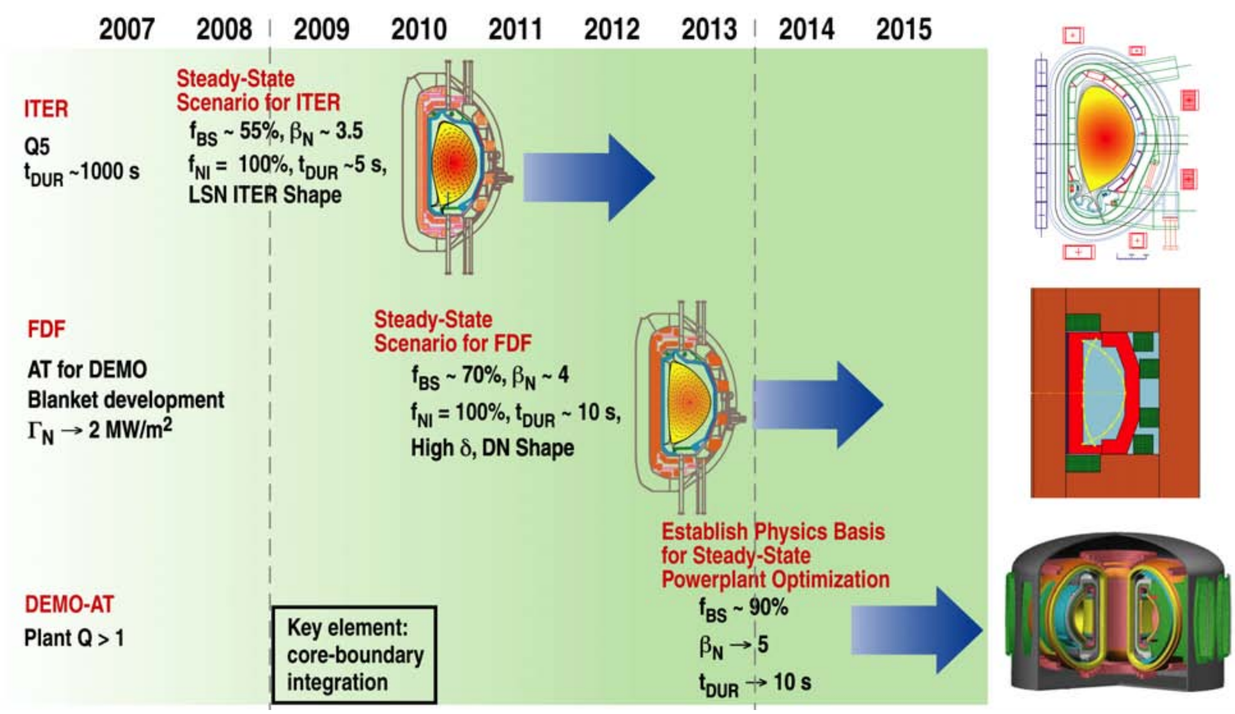

Fig. 3-1. The DIII-D five year plan for advanced scenario research focuses on validating the physics basis for three target applications: ITER, FDF, and DEMO-AT. The left-hand column lists the major goals for the target applications. The remainder of the figure shows the proposed parameters of relevant discharges in DIII-D and the timescales for development of these discharges.

The DIII-D program plays an essential role in the worldwide development of advanced scenarios. It possesses a unique combination of attributes relevant to this research and will be without peer in this respect for at least 10 years. The tokamak itself is of sufficient size to investigate plasma parameters relevant to fusion power plants, while retaining the flexibility to modify in-vessel components, heating systems, and diagnostics on a reasonable timescale to address issues as they arise. The tokamak is engineered to allow experiments to probe instability limits to validate models and to investigate scenarios systems, and diagnostics on a reasonable timescale to address issues as they arise. The tokamak is engineered to allow experiments to probe instability limits to validate models and to investigate scenarios reaching stationary conditions on the longest plasma physics timescales with control methods relevant to power plants. The DIII-D team occupies a leadership position both in advanced scenario development and plasma control research. This expertise, matched with the DIII-D tokamak capabilities, facilitates rapid progress in this area. Demonstration of the use of the physics knowledge gained on DIII-D in shortening the learning curve for applying advanced scenarios in larger tokamaks such as JET will facilitate the eventual use of such scenarios on ITER and other tokamaks.

3.2. KEY ELEMENTS OF ADVANCED TOKAMAK RESEARCH ON DIII-D

Development of advanced scenarios requires integration of a large number of research disciplines. This integration occurs primarily in three distinct ways. First, the core plasma solution must integrate confinement, stability, and current drive physics to yield a self-consistent, stationary plasma. Second, the heat and particle fluxes (sources and sinks) must be handled in a manner that integrates the needs of the core plasma solution, for example, low impurity density and control of the total particle density, with the requirements for heat removal and particle control at the boundary in order to ensure a long lifetime for the first wall. Third, this overall solution has to be integrated into a comprehensive control environment that can initiate, maintain, and exit from the desired plasma conditions in a controlled fashion in both normal operation and in the face of off-normal events. Advanced scenario research in DIII-D in this five-year plan addresses all three areas of integration.

Scenario development on DIII-D relies heavily on strong interaction between theory/modeling and experiment. Theory-based models are used to identify regions in the large potential operating space of DIII-D that are favorable for finding plasma conditions suitable for the various scenarios. As these predictions are tried, refinement of the models occurs, leading to a more complete description of the plasma behavior. It is to be expected that some plasma models will be shown to be quite accurate (those with extensive validation in the past), while other models will be shown to be inadequate, leading to a new and more complete description of plasma behavior in advanced scenarios. The outcome of the research is not simply a proof that the desired scenarios exist, but also a validated physics basis from which projections, optimizations, and intelligent control methods can be derived.

The target application discharges illustrated in Fig. 3-1 will be developed using a staged approach as new hardware capability becomes available for DIII-D. Moreover, many of the planned hardware upgrades are essential to maximizing the output of the advanced tokamak scenario development program. An outline of the plan is illustrated in Fig. 3-2. Discharge parameters will initially be limited in total stored energy and pulse length by the capabilities of the DIII-D facility. Thus, advanced tokamak scenarios will be studied at relatively low toroidal field and short pulse length during the first two years of the plan. As the heating and current drive and pulse length capabilities of DIII-D are upgraded, the discharges under study can be extended to higher beta and pulse length greater than the resistive time. In addition, upgrades to heating and current drive capability will allow more detailed study of control algorithms. When the complete set of heating system upgrades is complete, advanced scenario discharges will be able to reach the maximum projected parameters at the highest available toroidal field and with stationary conditions at the longest available pulse length. The DIII-D facility will be in a position to establish and optimize controlled high beta steady state scenarios in preparation for DEMO in the following five-year program. Development of the physics basis will continue throughout the period of the five-year plan. Complete details of the parameters of the advanced tokamak scenario discharges and the research plan on which the outline shown in the figure is based are provided in Section 3.3.

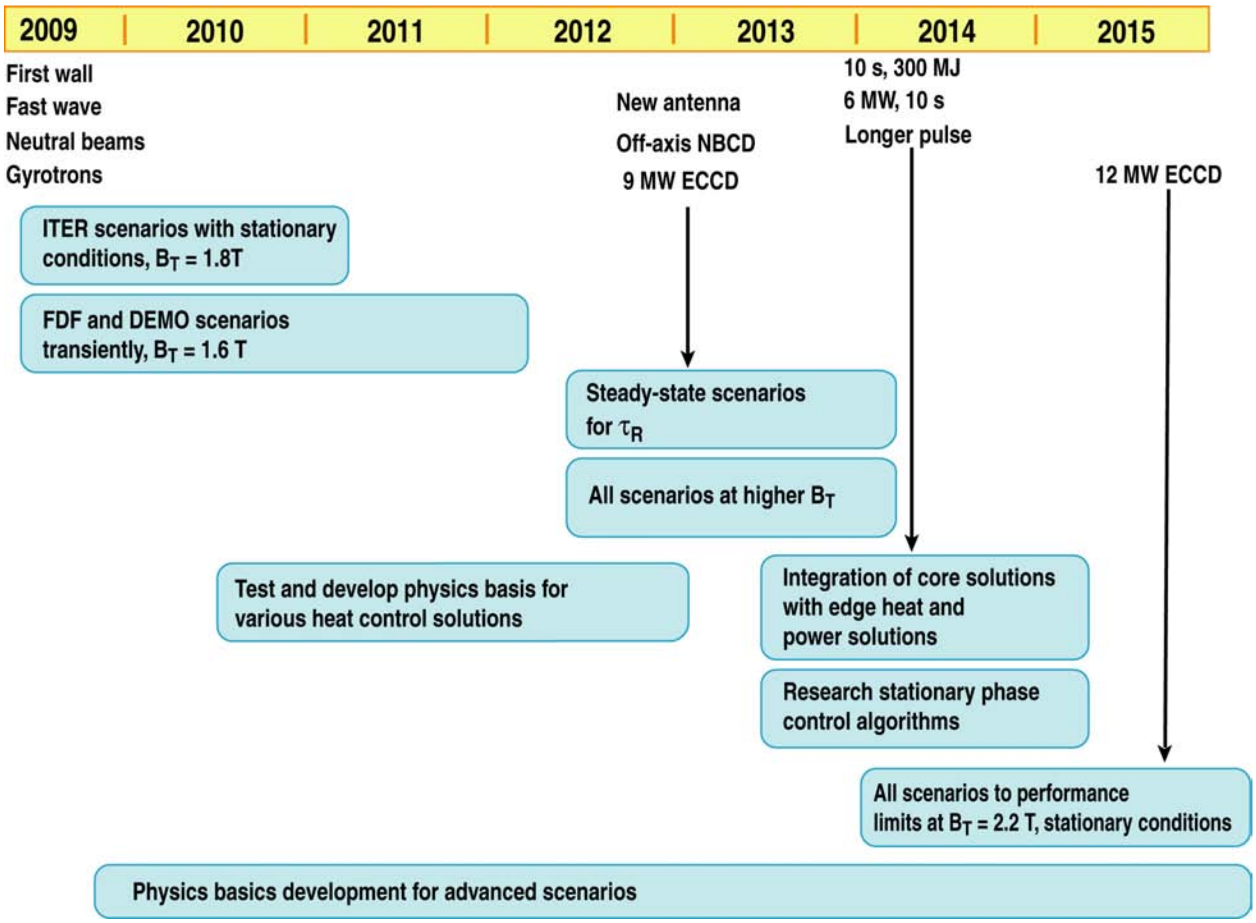

Fig. 3-2. Advanced scenario development during DIII-D five year plan will advance with the availability of enhanced heating and current drive capability.

3.3. INTEGRATED SCENARIOS

Four distinct classes of operating scenarios are under active study. All four have an H-mode edge pedestal. The ITER baseline scenario is a conventional H-mode plasma with q_{95} = 3.0. Sawteeth are expected, and the pressure is limited to modest values (β_N < 2) to avoid destabilization of tearing modes. The advanced inductive scenario pushes to higher pressure (β_N < 3), which allows higher fusion power and fusion gain at fixed *B* and *I*. The pressure limit in this case is set by the m=2/n=1 tearing modes. Higher *n* tearing modes are often present and give only a modest reduction in confinement. This scenario fills the role of a candidate for pulsed power plant operation with the highest inductive performance without active profile or instability control requirements. The hybrid scenario was originally targeted at maximum fluence in an inductive pulse in ITER to allow limited nuclear testing. At reduced current (q_{95} ≈ 4) and higher normalized pressure than the baseline scenario (β_N < 3.5), this scenario not only can maximize the fluence/pulse of ITER, but also has the potential to deliver the same fusion power and fusion gain as the ITER baseline scenario with reduced magnetic free energy available in the event of a disruption. With the development of a solid physics basis, this scenario is an alternative scenario to achieve the Q = 10 physics objective for ITER. The final class of operating scenario under study on DIII-D is the steady-state scenario. The goal is to attain fully noninductive sustainment at reduced current ($q_{95} \approx 5$) to enable continuous operation of a tokamak. In order to maintain adequate fusion power, the normalized pressure must be increased. For the ITER Q = 5 goal, $\beta_N \approx 3.5$ is probably sufficient. For the FDF mission of demonstration of tritium self-sufficiency, $\beta_N \approx 4$ is needed. For the economic production of electricity in a steady-state tokamak, a goal of $\beta_N \approx 5$ has been set. In DIII-D, multiple paths to these steady-state goals have been identified. Fortunately, the same mixture of upgrades to the heating and current drive systems, the tokamak itself, and the power systems at the DIII-D site are sufficient to explore each of these paths.

The relative locations of these scenarios in tokamak operating space are illustrated in Fig. 3-3. In this plot, the vertical axis is related to fusion power output, and the horizontal axis is related to the fraction of the plasma current supplied by the self-generated bootstrap current, which is required to be high for steady-state high fusion gain. The dashed diagonal lines are different values of q_{95} and lines representing higher q_{95}. The lines of constant β_N are shown as hyperbolas. The current limits occur at low q_{95} (q_{95} < 2), while the pressure limit is indicated at β_N = 5 at all q_{95} for the purposes of illustration only. It clear that the pressure limit sets the highest q_{95} for which the fusion power can be maintained at a specified level. What is not clear from this graph is that the fusion gain drops as $1/(q_{95})^2$, so that steady-state operation at the relatively high q_{95} necessary to maximize the bootstrap current demands operation near the pressure limits in order to maximize the fusion power. For advanced inductive scenarios, both the fusion power and fusion gain are maximized at the highest pressure and lowest q_{95}.

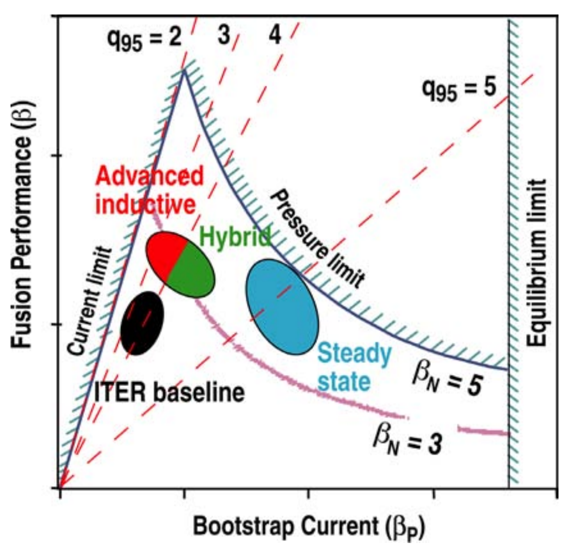

Fig. 3-3. Relative locations of scenarios in tokamak operating space.

The candidate plasmas developed in DIII-D to fulfill missions of the various scenarios described above can be most easily distinguished by the radial profile of q (Fig. 3-4). The ITER baseline scenario (green dashed curve) has a q profile with $q(0) \leq 1$ and rising smoothly to the edge. The advanced inductive and hybrid scenarios have very flat q profiles in the plasma core and very steeply increasing q profiles at the edge. In both scenarios, $q(0) \approx 1$, slightly higher than 1 in the hybrid scenario and slightly lower in the advanced inductive scenario. The main difference in these two scenarios is the value of q_{95} — the hybrid has $q_{95} > 4$ and the advanced inductive has $q_{95} < 4$. There is a much larger variety of q profiles proposed for steady-state scenario operation, as indicated by the shaded band in the figure. The q profiles for two leading candidates are indicated in the figure. The solid line is typical of high q_{min} scenarios on DIII-D, with a radial profile similar to the hybrid scenario except elevated at all radii. The dashed line is the extreme case of a high ℓ_i scenario, where the goal is to maximize the magnetic shear ($dq/d\rho$) across the radius.

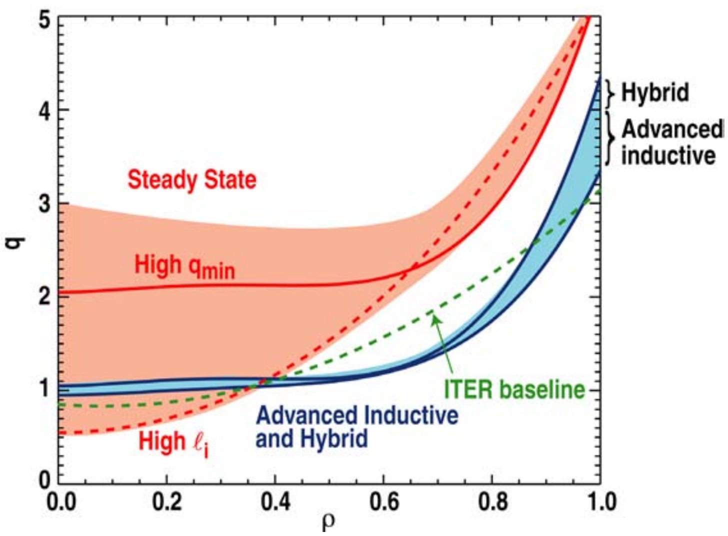

Fig. 3-4. Radial profiles of q for the four classes of discharge scenarios under study at DIII-D. The high q_{min} and high ℓ_i cases are two examples of steady-state scenarios.

The remainder of this section will focus on the development of the various advanced scenarios by demonstrating their existence in DIII-D. Establishment of the physics basis and the required operational control, instability control, and heat and particle flux solutions will be discussed in subsequent sections of this chapter.

3.3.1. Status of Advanced Scenario Development

As stated previously, the demonstration of stationary scenarios with characteristics suitable for their intended mission is a high priority for the DIII-D research program. The typical development path begins with transient discharges to provide a proof that such profiles can be obtained. The development then continues through pulse lengths sufficient for equilibration of the pressure to pulse lengths sufficient for equilibration of the current profile. The pressure equilibrates on the timescale of several energy confinement times τ_E, which is on the order of 100–200 ms in DIII-D. The current equilibrates on the timescale of several current profile relaxation times τ_R, which is on the order of 2–4 s in DIII-D. This separation of timescales allows independent validation of the relevant models for the relaxation processes in DIII-D. When the core plasma is stationary for several τ_R, the first level of integration (stability, transport, and current drive) can be considered complete. The integration with the control environment occurs at the stage where the relaxation time of the quantities required for access and operation of these scenarios matches the scenario development. Integration of the edge solutions requires a scenario at least to the pressure equilibration stage to assess the impact.

The present status of the various scenarios will be reviewed briefly here. The references point to the most recent publications, whose references serve as a guide to the history of the scenario development.

For the purposes of this plan, the conventional H mode scenario is assumed to be typified by the ITER baseline scenario design (q_{95} = 3.0, β_N = 1.8, Q = 10). The physics basis of this operating scenario has been summarized [IPB 1999, PIPB 2007], and outstanding issues to be addressed on DIII-D in this plan are detailed in the chapters on ITER Physics and Fusion Science. The level of performance to achieve Q = 10 in ITER in a conventional H-mode plasma is used in the advanced scenario research as a benchmark for performance comparison. The figures of merit used for this comparison are the normalized pressure (β_N) and the confinement quality compared with scalings derived by regression analysis (for example, H_{89P} or $H_{98y,2}$). A composite figure of merit for fusion gain, G = β_N H_{89P}/q_{95}^2 [Luce 2004], indicates the relative gain performance of a plasma. For ITER, a plasma with G = 0.42 will yield performance near Q = 10, while G = 0.30 will yield performance near Q = 5. This figure of merit does not translate universally to these gain factors; tokamaks with different size and B will require different values of G to reach Q = 10.

Advanced inductive and hybrid scenario discharges have been demonstrated under stationary conditions in DIII-D [Wade 2005]. An example is shown in Fig. 3-5 of an advanced inductive discharge operated for almost 10 τ_R with G > 0.6. Hybrid scenario discharges with G > 0.4 have been demonstrated for > 4 τ_R. Limits to pressure have been determined as a function of q_{95} and density. The limiting instability is an m=2/n=1 tearing mode. For discharges with q_{95} > 4 (hybrid discharges), sawteeth are generally absent and the m=2/n=1 tearing mode appears for β_N > 3, near the ideal n=1 limit in the absence of a wall. For discharges with q_{95} < 4 (advanced inductive), sawteeth are present, but are quite small, and the m=2/n=1 tearing mode appears at about 85% of the ideal n=1 limit in the absence of a wall. Both hybrid and advanced inductive discharges in DIII-D have a m=3/n=2 tearing mode that leads to a current profile, stationary for several τ_R, with less central current density [or higher $q(0)$] than predicted by current relaxation using a neoclassical Ohm's law. The confinement in these discharges is at or above the ITER H-mode confinement scaling (IPB98y,2). Work has begun to assess the confinement under conditions more relevant to plasmas in power plants (lower rotation, equal electron and ion temperatures) [Politzer 2007] and to test the compatibility of these scenarios with solutions to handle the heat flux at the walls [Petrie 2007]. While some work remains to document the operational boundaries and characteristics of these plasmas, the focus of the work in this plan will be the establishment of the physics basis to confidently extrapolate these scenarios to future tokamaks and to optimize them for this level of performance.

As noted previously, the optimal approach to a steady-state tokamak scenario remains an open question. Therefore, the demonstration of stationary plasmas with no inductive current drive is the highest priority research topic. Two approaches are under investigation on DIII-D. The high q_{min} approach is limited in pressure by kink instabilities. Reaching the highest pressures in this scenario will require stabilization of these modes either by rotation sufficient to make the wall with finite resistivity act as a perfectly conducting wall or by direct feedback stabilization. This approach has received the majority of the experimental effort in DIII-D. The high ℓ_i approach is also limited in pressure by kink instabilities; however, this scenario optimizes the current profile for stability even in the absence of a conducting wall.

Both approaches have the potential for stable noninductive operation at $\beta_N \approx 5$ according to modeling. The main focus for steady-state scenario development is the demonstration of fully noninductive current sustainment for > 2 τ_R at progressively higher pressures to meet the requirements of ITER, FDF, and DEMO. These efforts will entail as a matter of course establishing a physics basis for optimization and for devising the required control methods in DIII-D.

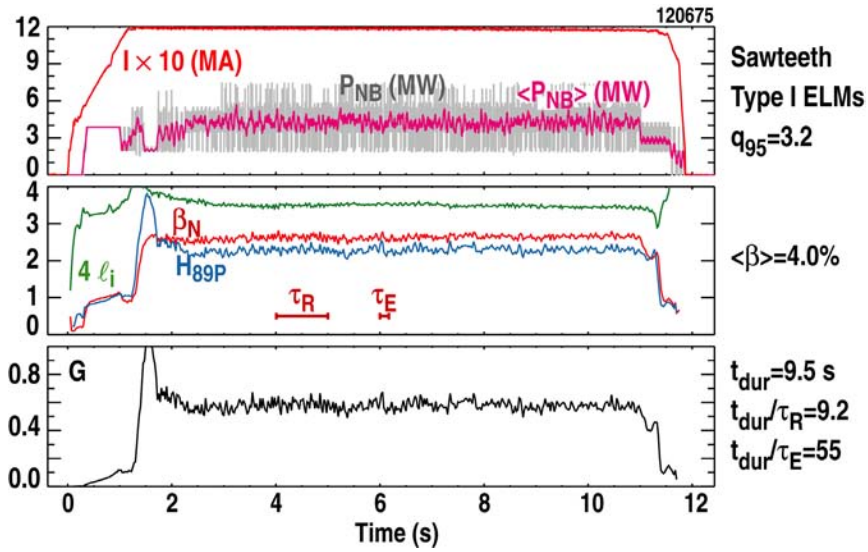

Fig. 3-5. An example of an advanced inductive discharge operated for almost 10 τ_R with G > 0.6.

The high q_{min} scenario has demonstrated stationary performance for slightly longer than τ_R at G = 0.3, which is sufficient to meet the ITER physics objective on steady-state operation (Fig. 3-6). The plasma shown does not have fully noninductive current sustainment. To reach that goal, more off-axis current drive or more bootstrap current is required. Optimization of performance through variations of the plasma shape and the radial distribution of the off-axis current drive by ECCD has begun. This scenario has achieved $\beta_N > 4$ transiently.

The high ℓ_i scenario was investigated transiently in the 1990s, producing the required current profiles by means of current ramps and elongation ramps. These techniques are not consistent with the constraints expected for tokamaks with superconducting coils producing burning plasmas. A new technique, based on the reference ITER startup scenario, was demonstrated in DIII-D in 2007 with transient performance at $\beta_N > 4.5$.

3.3.2. Goals for Advanced Scenario Development (2009–2013)

As introduced above, the targets for advanced scenario development are both steady-state and advanced inductive scenarios for use in ITER, a tokamak to demonstrate tritium self-sufficiency such as FDF, and a prototype electricity production tokamak (DEMO). The ITER parameters are naturally more definite than for the other two targets. ITER has a single-null divertor design, while the FDF and DEMO configurations are expected to require the higher pressure limits and distributed heat fluxes associated with double-null divertor configurations. DIII-D is the only operating tokamak that can compare directly the benefits and costs of single-null and double-null divertor operation directly.

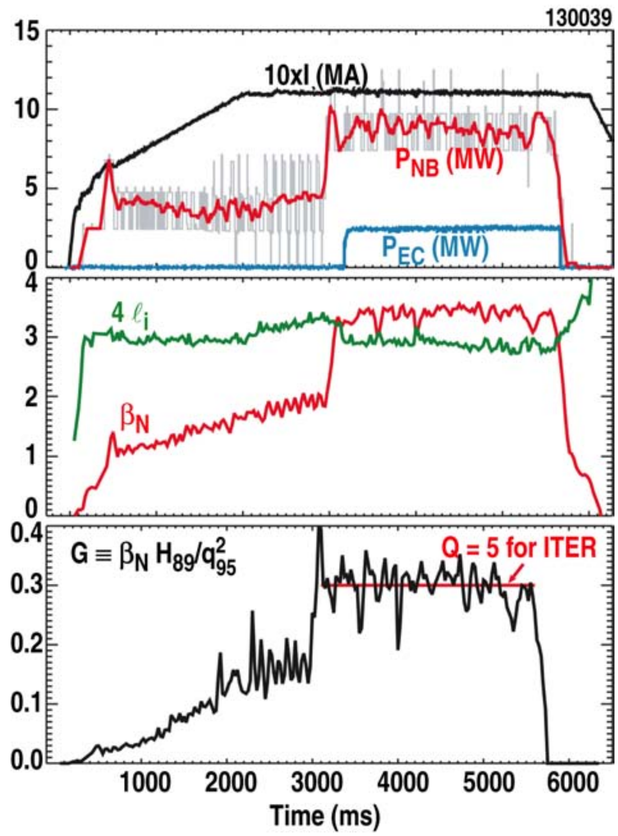

Fig. 3-6. The high q_{min} scenario has demonstrated stationary performance for slightly longer than τ_R at G = 0.3.

The portfolio of operating scenarios to be researched in this plan is outlined in Table 3-1. The operating parameters for all of these scenarios are calculated for the maximum *B* allowed now in DIII-D (2.2 T). Each of the columns is based on equilibria and profiles already achieved on DIII-D in similar operating conditions, but at lower performance and in many cases not under stationary conditions. The ITER scenarios are for equilibria that match the plasma boundary of the ITER design equilibrium. In all cases, the values of β_N and q_{95} are chosen. With the equilibrium shape and B chosen, these parameters specify the plasma current and stored energy. The density is chosen to be consistent with the pumping capabilities of the present DIII-D divertor cryopumps. For the baseline and advanced inductive scenarios, the value of H_{DS03} (scaling taken from dimensionless scaling experiments primarily from DIII-D and JET [Petty 2003]) is taken from existing plasmas and used to calculate the required power for the scenario. From the table, it is clear that the main difference between the DS03 scaling and the ITER H-mode scaling (IPB98y,2) is that the confinement degradation at high β is much weaker for DS03, consistent with the β scaling experiments. For the steady-state scenario for ITER, this same procedure is followed, but for the FDF and DEMO steady-state scenarios, the power is the maximum specified in this plan and the H factors to achieve the chosen parameters are the derived quantity. The calculations of the driven current from the various sources are derived from 1-D calculations based on scaled profiles from existing DIII-D plasmas. For the steady-state scenarios, the current profiles are generated self-consistently, and the equilibria recomputed based on the distribution of the current sources, assuming no inductive current. No attempt has yet been made to assess the ideal MHD stability of these profiles. The main point of this exercise is to determine the heating and current drive powers necessary to reach the target parameters.

Table 3-1
The Portfolio of Discharge Types to be Studied in the DIII-D Advanced Scenario Research Program

	ITER Baseline (#1)	Advanced Inductive (ITER Shape) (#2)	Advanced Inductive (DN Shape) (#2)	Hybrid (ITER Shape) (#3)	Hybrid (DN Shape) (#3)	Steady-State (ITER) (#4a)	Steady-State (FDF) (#4b)	Steady-State (DEMO) (#4c)	Steady-State (DEMO-High ℓ_i) (#5)
B (T) @1.6955 m	2.2	2.2	2.2	2.2	2.2	2.2	2.2	2.2	2.2
I (MA)	1.93	1.93	2.59	1.45	1.94	1.4	1.55	1.55	1.55
n (10^{19} m^{-3})	5.8	5.9	10.4	4.4	6.8	4.5	6.2	6.2	6.2
P_{aux} (MW)	5.9	15.8	11.6	20.8	21.6	19.5	27.6	27.6	27.6
P_{NB} (MW)	2.3	7.7	8.0	12.7	11.0	8.7	15.	15.	15.
P_{EC} (MW)	0	4.5	0.0	4.5	7.0	7.2	9.	9.	9.
P_{FW} (MW)	3.6	3.6	3.6	3.6	3.6	3.6	3.6	3.6	3.6
I_{BS} (MA)	0.41	0.65	0.72	0.56	0.72	0.92	0.92	1.32	1.21
I_{NB} (MA)	0.05	0.17	0.06	0.39	0.15	0.23	0.25	0.27	0.30
I_{EC} (MA)	0.0	0.0	0.0	0.0	0.0	0.17	0.16	0.17	0.14
I_{FW} (MA)	0.07	0.11	0.03	0.16	0.08	0.0 (0.17)	0.0 (0.09)	0.04 (0.08)	0.0 (0.19)
β_N	1.8	2.8	2.8	3.2	3.2	3.5	4.0	5.0	5.0
H_{89P}	2.1	2.1	2.4	2.2	2.2	2.4	2.3	3.0	3.0
H_{98y2}	1.32	1.37	1.30	1.48	1.45	1.73	1.54	2.07	2.09
H_{DS03}									
q_{95}	3.0	3.0	3.0	4.0	4.0	5.0	5.0	5.0	5.0
G	0.42	0.65	0.75	0.44	0.44	0.34	0.38	0.6	
Design Equilibrium	125946 @3350	125946 @3350	126067 @3375	125946 @3350	126067 @3375	125946 @3350	126067 @3375	126067 @3375	126067 @3375
Design Profiles	125946 @3350	120675 @5500	120675 @5500	113993 @4900	113993 @4900	111221 @4105	126067 @3350	122976 @2900	129613 @1700

The maximum power capabilities assumed to be available at the end of the five-year planning period are as follows:

- NBI (15 MW co-injection, 5 MW counter-injection)
- EC (9 MW absorbed, 12 MW source)
- FW (3.6 MW absorbed, 6 MW source)

This complement of heating and current drive systems implies the re-installation of the eighth NB source, an increase in the EC system from 6 MW to 12 MW at the source, and installation of a new long pulse fast wave antenna.

In Table 3-1, the leftmost column gives the DIII-D reproduction of the ITER baseline scenario. At the level of confinement expected for ITER (H_{89P} = 2.1), this scenario can be achieved with co-NBI, balanced NBI, EC, EC+FW, or counter-NBI+(EC or FW). This flexibility will not only facilitate tests of the main heating systems for ITER, but will test the effects of rotation and electron-ion temperature ratio on confinement and tearing mode stability. Note that the bootstrap current fraction is <25% for this scenario.

Moving to the right, the next scenarios are advanced inductive scenarios in the ITER and nominal DIII-D double-null shapes, respectively. Comparison of the ITER version with the ITER baseline scenario shows that the average pressure is 55% higher in the advanced inductive scenario. The clear advantage in plasma current of the lower aspect ratio and double-null divertor at fixed q_{95} is apparent. Also, the pressure is 34% higher at fixed q_{95} for the double-null case at fixed toroidal field. Despite the higher pressure, the bootstrap fraction remains at about 30%. For the ITER shape, the table shows the heating power split to be the maximum FW+EC power presently available with the remainder made up with NBI. The 7.7 MW of NBI could be co-NBI or balanced. As with the ITER baseline scenario case, the ability to vary the rotation and the ratio of electron to ion heating power will facilitate transport and stability studies. For the double-null shape, the lowest achievable density is above the EC cutoff density, so the heating mix is limited to NBI+FW.

The next two columns to the right are hybrid scenarios with $q_{95} = 4$ in the ITER shape and the DIII-D double-null shape. These hybrid scenarios are just at the full-power limit of the existing heating systems. In the table, the ITER case is shown with the maximum EC+FW power now available, while the double-null case is shown with equal NBI and EC+FW power, assuming the upgrade to the EC system. The latter case is more realistic, in that it allows for feedback control of the power of both the NBI and EC systems to achieve the desired pressure level. Only with the upgraded EC system would it be possible to obtain these hybrid scenarios at $B = 2.2$ T with no rotation. Despite the higher q_{95} and β_N, the bootstrap fraction remains at about 40%.

The rightmost four columns are four different steady-state scenarios with increasingly stringent specifications. For these cases, the FW was used only for heating. The values in parentheses show the capability if the FW was also to be used for current drive. The ITER scenario is consistent with the performance of existing single-null DIII-D discharges. But this level of performance has not been demonstrated in the ITER shape, which is critical due to the observed sensitivity of the pressure limits and confinement to the higher order details of the boundary shape. Demonstration at $B = 2.2$ T requires a significant upgrade to the EC system to supply the off-axis current drive or, alternatively, modification of two of the NB lines in DIII-D to allow off-axis NBCD by vertical steering (Section 3.3.4). Figure 3-7 shows the calculated current density and q profiles for the ITER steady state scenario (#4a in Table 3-1).

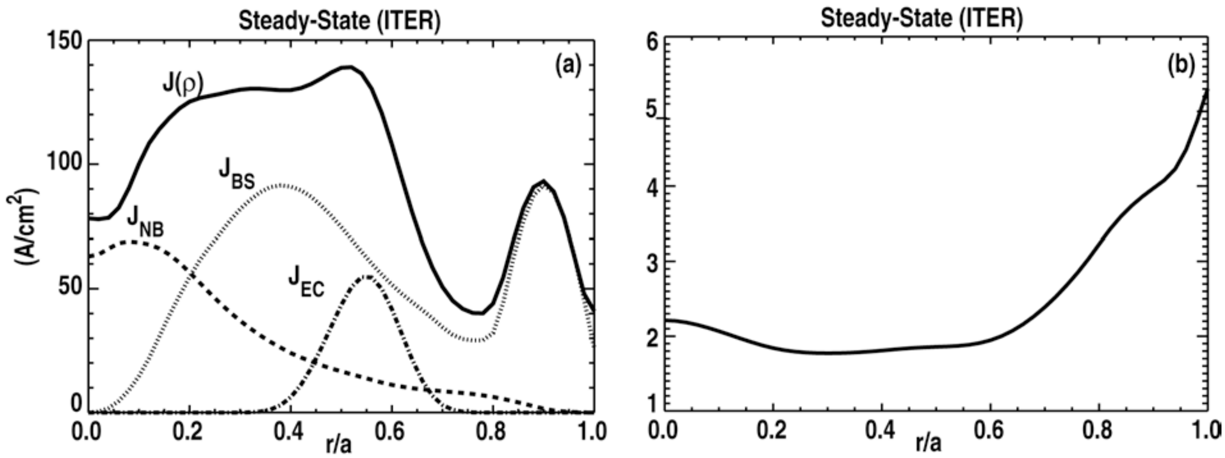

Fig. 3-7. Calculated radial profiles of (a) current density and (b) q for scenario #4a in Table 3-1 (ITER steady state).

The FDF and DEMO scenarios require β_N values larger than presently achieved under stationary conditions in DIII-D. The FDF value of β_N = 4 has been exceeded transiently and stationary discharges have been obtained at ~95% of this level. The DEMO value of β_N = 5 is quite challenging, but the two scenarios identified in the table have been modeled at this level of performance. Each of these last three scenarios would require the full upgraded power capability of 27.6 MW to be realized at *B* = 2.2 T, unless the confinement quality significantly exceeds H_{89P} = 3. To obtain sufficient off-axis current drive in the high q_{min} scenario, both off-axis NBCD and off-axis ECCD will be required. Note that the bootstrap fraction varies from 60%–85% in these cases.

3.3.3. Plan for Scenario Development

The staging of the scenario development across the five-year plan is shown in Fig. 3-2. The first two years of the plan will focus on demonstration of the ITER scenarios under stationary conditions. This will begin at reduced toroidal field (*B* = 1.8 T), both to reduce the power demands to allow adequate control and to shorten the resistive times to match the capabilities of the existing heating and current drive systems. During this time, the FDF and DEMO scenarios will be explored transiently at *B* as low as 1.6 T both to lessen the power demands and to maximize the current drive capability of the EC system. The strong on-axis NBCD will make this work on the high q_{min} scenario challenging. In the middle years of the plan, off-axis NBCD will become available, as will longer pulse NBI, the first of the EC system upgrade, and the longer pulse fast wave heating capability. This will allow the work to begin to extend the steady-state scenarios to the resistive timescale and demonstration of all scenarios under stationary conditions at higher *B*. The longer pulse length is also essential to the integration of the core solutions with the heat and particle flux handling solutions in the edge (Section 3.7) and to development of control algorithms for the stationary phase of the discharge (Section 3.5). With the completion of the EC system upgrade toward the end of the five-year period, all scenarios can be extended to their performance limits under stationary conditions. In concert with the scenario development activity, the work to establish the physics basis for each of the scenarios will be carried out (Section 3.4).

3.3.4. Hardware Upgrades Required for Scenario Development

The major hardware upgrades motivated by advanced scenario development are increased heating and current drive power (20.6 MW -> 27.6 MW, delivered power to plasma) and comprehensive upgrades to enable full performance operation to 10 s. Correlated to the upgrade to the heating and current drive systems is modification of two of the four NBI beamlines to allow injection of 10 MW of NB for off-axis NBCD by vertical steering. The increase in the total power is gained by increase of the EC system power from 6 MW (source) to 12 MW (source) over the five-year period and completion of the installation of the eighth NB source in 2009.

The motivation for increase in the heating and current drive power is to have adequate power to demonstrate steady-state scenarios up to $\beta_N = 5$ under stationary conditions. The proposed power upgrade is sufficient for this mission at B = 2.2 T only if the confinement quality is quite high ($H_{89P} = 3$). At lower B, this same power will be adequate to reach stationary conditions, even with lower confinement. In the near term, the full power of the NBI system is required to reach high β_N; however, the high q_{min} scenario cannot tolerate the central NBCD at full power at lower values of B. The most expedient solution to this conflict is to modify two of the NB lines to allow full power injection vertically steered to drive current as far as half the plasma radius off-axis. New calculations indicate very good current drive efficiency (actually better than on-axis due to increased trapped electron fraction) with good localization off-axis as long as *B* and *I* are in the same direction (for a beam steered downward). Since DIII-D has the unique capability to operate with *B* and *I* in either direction and has full divertor pumping and heat handling capability in the upper and lower divertors, the localization conditions on *B* and *I* require no further modifications of the tokamak. At the end of the five-year plan, the full EC system will be available for either on-axis or off-axis heating and current drive. In concert with the modified NB system, a large range of scenarios can be tested up to the ideal MHD limits. The antenna now attached to the 60 MHz FW transmitter is capable only of 10 MJ of input energy, limiting it to 5 s at full transmitter power. It is proposed to replace this antenna with an advanced antenna design capable of 10 s operation at full transmitter power during the five-year plan.

The capability for 10 s operation enables several essential research elements for the steady-state scenarios. First, it is very important to have demonstrations of stationary performance in the steady-state scenarios for at least a resistive time. Without such demonstration, the confidence in adopting such a scenario for future tokamaks will be lacking. At full field, the resistive times are 6–9 s for the steady-state scenarios in Table 3-1. The resistive time will scale roughly as *B*, so the highest performance scenarios will require 10 s operation to get to 1.5 τ_R, even at reduced *B*. In addition to this demonstration, tests of the effects of coupling the heat and particle flux handling solutions necessary for the next generation of tokamaks require operation on at least the resistive time scale to observe any response of the current profile. Finally, the demonstration of control of the operational conditions, including an instability avoidance system (Section 3.5) requires demonstration on the resistive timescales. Reaching 10 s at full performance on DIII-D will require significant modification of the NB system to deliver more than three times the energy capability of the current system. Modifications to the toroidal field coil cooling for such operation should be complete by the beginning of the five-year plan. New power supplies to drive the poloidal coil system will be required, as will installation of a new site substation that has already been procured. Modification of the divertor tiles and the cooling circuits in the upper divertor is required to remove the 300 MJ of energy required for the highest performance scenarios. The vacuum vessel is engineered for 400 MJ of input energy; this likely sets the fundamental limit on energy throughput for DIII-D.

For reaching the advanced inductive scenario goals at B = 2.2 T, no upgrade in power is required beyond that projected to be available at the beginning of the 2009 DIII-D campaign. The NBI system would not be capable of operation to 10 s at this power level, which is approximately two resistive times, nor would the FW antenna connected to the 60 MHz transmitter. The total heat loads for this duration are 120–160 MJ, which is likely beyond present heat handling capabilities, especially in the single-null ITER equilibrium.

The hybrid scenarios are at the upper end of the present heating system capability, which implies no feedback control is possible without upgrading the power capability of the EC system. As with the advanced inductive scenarios, the resistive time is ~5 s, so 10 s operation is required to demonstrate stationary conditions. This implies a significant upgrade of the pulse length capability of the NBI system. With a total input power of ~22 MW, the tokamak must handle >220 MJ, which will require significant development of heat flux handling solutions (Section 3.7) and upgrades to the in-vessel components in DIII-D.

No heating system upgrades are required to achieve the baseline scenario for durations of 5 s, which is less than 2 τ_R, since the mix of heating types is flexible. The present FW antenna connected to the 60 MHz transmitter is not capable of handling a 2 MW source at 5 s, but the other FW and EC systems are rated for this duration at full power. Upgrades to the poloidal field coil power supplies and the toroidal field cooling and supplies for 10 s operation would allow operation to >3 τ_R. The present divertor system is capable of handling >60 MJ, so no upgrade is needed for 10 s operation of this scenario.

3.4. DEVELOPMENT OF THE PHYSICS BASIS FOR ADVANCED SCENARIOS

The previous section focused on the plan and requirements for demonstrating various advanced scenarios. As discussed in the introduction to this chapter, that work is necessary, but not sufficient for these advanced scenarios to be adopted as the standard operating mode of a future tokamak. The establishment of a basis for extrapolation to larger tokamaks and optimization of the design of those not yet in existence is also necessary. Here the physics basis for core integration will be discussed. Operational control, instability control, and heat and particle flux solutions will be discussed separately in the following sections.

3.4.1. Stability

The design of the various advanced scenarios discussed here has ideal MHD stability as a basic guide to which classes of q profiles should be investigated. The basic trends indicated by variations in model equilibria of the q profile and boundary shape have been validated by transient experiments. Prediction of absolute pressure limits remains out of reach for now, due to the lack of precise predictive capability of the pressure profile and especially the details of the pressure profile in the H-mode pedestal region. Resistive MHD is important in both the advanced inductive and steady-state scenarios. The hybrid and advanced inductive scenarios rely on the presence of tearing modes that provide a significant benefit by reshaping the current profile while remaining relatively benign with respect to energy confinement. Tearing modes are not beneficial to the steady-state scenarios, since they transport current much faster than resistive diffusion and in ways that are virtually impossible to recover from in high temperature plasmas. For the steady-state scenarios, the key is to know where the resistive MHD instabilities occur (both in the formation phase and the high performance phase) and take action to avoid destabilizing them. Stability to modes driven by energetic particles is also of importance to advanced scenarios. The advanced inductive and steady-state scenarios with $q(0) \approx 1$ should have similar stability characteristics to the ITER baseline scenario, but with the possibility of higher fast particle fractions. The flatter magnetic shear in the core may also alter the gap structure of the Alfven eigenmodes somewhat. The steady-state scenarios with elevated q_{min}, especially those with significant reversal of the q profile in the core, are predicted to be more susceptible to energetic particle modes. The DIII-D advanced scenario program has focused more on flat q profiles even at high q_{min}, partly to avoid unfavorable stability projections to these modes in burning plasmas.

For ideal stability the outstanding issues to address are:

- Sensitivity of low n modes to higher order variations in the boundary shape (e.g., squareness)
- Validate ideal MHD stability as the ultimate limit to plasma performance and implement fast real-time calculations of the ideal MHD stability that can be used for feedback control
- Understand whether the correlation in pedestal stability and global stability optimization is a simple correlation through the MHD energy principle or a more complex interaction through plasma transport

For resistive stability, much less work has been done, both from modeling and experimental validation. The key issues here are:

- Validation of a model that predicts accurately the trends in the linear stability threshold
- Characterization of the dependence of the saturated mode amplitude on rotation, ρ_*, and collisionality
- Understanding the role of error fields and rotation in mode locking

For modes driven by energetic particles, the main effort will be to validate model predictions in DIII-D under conditions as close as possible to the conditions in burning plasmas, then seek ways to validate the key physics differences in the extrapolation.

3.4.2. Transport (Heat, Particle, Momentum)

The profiles of density, rotation, and temperature are the key elements that not only determine the fusion output, but also connect to both stability and current drive. Stability is sensitive to the profile of the pressure, both globally and locally in the pedestal. Operation above the calculated n=1 limits in the absence of a wall is greatly aided by plasma rotation to make the real wall act like a perfectly conducting wall. The bootstrap current is directly related to the profiles of density and temperature, as is the current drive efficiency for all auxiliary current drive schemes (through collisional dissipation). Impurity accumulation in advanced scenarios that have excellent confinement would lead to poisoning of the fusion reactions. All of these issues make characterization of the sensitivity of the density, rotation, and temperature profiles to various plasma parameters an important element of establishing the physics basis for advanced scenarios. The ultimate goal is a full theoretical description of plasma transport; however, simple empirical models that are validated over a limited range of parameters also play a critical role in feedback control methods and design optimization tools.

The highest priority research topics for advanced scenarios include:

- Dependence of energy confinement on q and magnetic shear
- Dependence of energy confinement on electron-ion temperature ratio
- Assessment of helium accumulation in advanced scenarios
- Characterization of momentum transport and separating the effects of error fields, fast-particle losses, and intrinsic rotation
- Determining whether density peaking should be expected in low collisionality plasmas
- Understanding the role of transport in setting the density and temperature pedestal height

3.4.3. Heating and Current Drive (Bootstrap, NBCD, ECCD, FW)

Sustainment of the plasma current profile noninductively is essential to steady-state operation. In general, the four types of current drive employed on DIII-D (neutral beam, electron cyclotron, fast wave, and bootstrap) have predictive models validated through detailed experimental comparisons. However, some issues remain that are relevant to the development of steady-state scenarios on DIII-D. The inductive scenario (advanced and hybrid) are observed to achieve the beneficial current profiles under stationary conditions by some anomalous means. Identifying this mechanism is key to the confident extrapolation of these scenarios to future machines.

Of all of the current drive mechanisms, NBCD is probably has the most need for additional validation of the model, especially for off-axis NBCD. New calculations were carried out for DIII-D to evaluate whether localized off-axis current could be driven by the NB system if the sources were shifted vertically to displace the beam deposition vertically [Murakami 2007]. Because of the constraints of the existing poloidal coils and vacuum vessel, the beam must pass through the existing port, so a vertical shift of the source implies an oblique injection angle in both the toroidal and vertical directions (Fig. 3-8). Surprisingly, these calculations found a significant difference in both the localization and the magnitude of the off-axis NBCD for co-injection with the direction of B. For beam steering to provide beam deposition at the half radius, the case with B in the same direction as I yields a profile of NBCD that is peaked off-axis and reasonably localized, while the case with B in the opposite direction from I (standard DIII-D operation) yields a broader profile with a peak near a normalized radius of 0.3 [Fig. 3-9(a)]. The magnitude of the current is 40% higher in the case with B and I in the same direction. The physics of this difference does not arise from the change in the direction of the particle drifts, but from the fact that the beam is much more parallel to the magnetic field lines in the case of B and I in the same direction. The efficiency of the NBCD is quite good off-axis, because the trapped electron fraction increases, which reduces the electron shielding of the injected ion current. This is in contrast with electron current drive schemes where the increased trapped electron fraction degrades the efficiency. For a given set of density and temperature profiles, the off-axis NBCD efficiency is comparable to that of ECCD at the same radius. The ECCD retains the advantage of higher localization and variation between discharges (and eventually during a discharge). Figure 3-9(b) is an example comparison of calculated NBCD to ECCD profiles. For the scenario development plan, it is recommended to modify two beamlines (10 MW) for vertical steering with the variation being possible only between monthly run periods. A design goal is to retain the ability to operate the beams with the same aiming as at present.

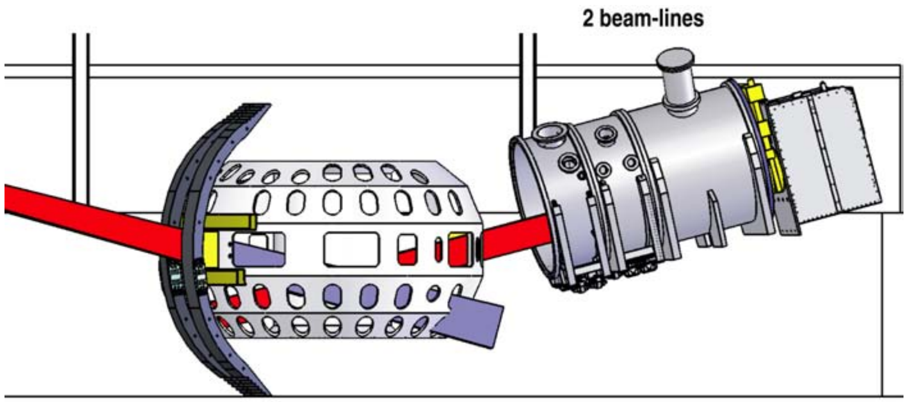

Fig. 3-8. A conceptual drawing of the off-axis neutral beam injection geometry for two of the DIII-D beam lines.

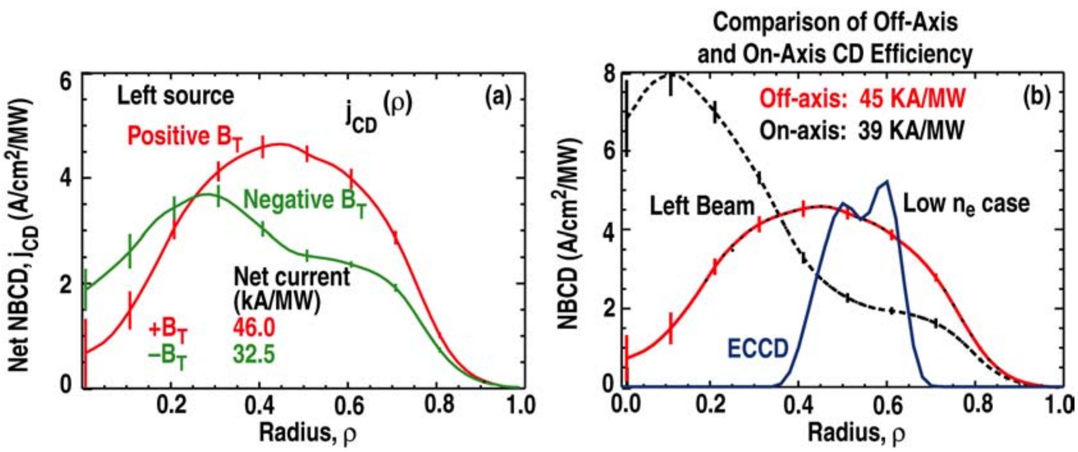

Fig. 3-9. (a) Calculated dependence off-axis NBCD profiles on the toroidal field direction. (b) A comparison of off-axis and on-axis current drive efficiency.

The outstanding physics issues for fast wave current drive are mainly those surrounding the competing absorption mechanisms. The current drive efficiency has been validated experimentally well enough that deviations from the model are used as circumstantial evidence of absorption mechanisms other than direct electron absorption becoming important. The two mechanisms under study are absorption at the edge and absorption on high harmonics of the fast ions injected by the neutral beams. The improved fast ion diagnostics being implemented now on DIII-D will also allow these issues to be addressed fully near the beginning of the five-year plan. While the coupling of the fast waves from the antenna is a technology issue, it should be noted that the increased use of ELM suppression and mitigation techniques for reduction of the transient heat loads will also ease one of the most significant technical challenges for the application of FWCD in advanced scenarios.

Both ECCD and bootstrap current are considered to be described well by the present models. For the bootstrap current, the biggest uncertainty is the applicability of the model to the pedestal region, where the neoclassical approximations break down. The bootstrap current density in the pedestal dominates the current profile locally, so gaining a predictive capability for this region is important for all H-mode scenarios. The main outstanding questions for ECCD for which there is no direct experimental information involve the current drive efficiency at high temperature and transport broadening of the current drive profile in this regime with long slowing down times. These will be addressed in the course of the scenario development and will not require dedicated experiments unless the results deviate significantly from the modeling.

3.5. CONTROL FOR ADVANCED SCENARIO OPERATION

The key challenge for high performance, steady-state operation of a tokamak discharge is production and maintenance of the optimized pressure and current profiles. This is an exciting area of research since the realization of this control will require detailed knowledge of stability, transport, and current drive physics and development of new control methods to be successful. This area will be an important research focus in DIII-D. Many of the technological resources are already in place to begin this work--diagnostics available in real time, a flexible and powerful digital plasma control system to execute the feedback control algorithms, and a flexible set of heating and current drive actuators (NBI, EC, FW). The continuing upgrade of the heating and current drive capabilities and the extension of the pulse length are essential to the activities discussed here.

This section outlines the approach to be taken toward studying the three key issues for advanced scenario control:

- Access to the desired target current profile as the discharge is formed
- Challenges in controlling a discharge with very high bootstrap current fraction
- Maintaining the high-performance discharge in steady-state.

These issues will be discussed in light of operation of steady-state discharges in DIII-D, as well as for the steady-state scenarios for ITER and DEMO.

3.5.1. Development of Feedback Control for Access to Advanced Regimes

Access to an advanced scenario is not possible starting from arbitrary conditions at the end of the current buildup. The challenge in the experiment is to use the current buildup phase to form profiles that allow access to the desired operating conditions and then maintain these profiles in steady state. The primary difficulties in DIII-D are that the time scales for making significant modifications in the current profile are comparable to the discharge length and that the pressure profile shape is primarily determined by transport coefficients and is thus difficult to change. For future tokamaks capable of steady-state operation, the primary issue is that a trial-and-error approach to the current buildup phase will be too costly in time. A focus of DIII-D research will be optimization of feedback control methods to produce and maintain desired current and pressure profiles in steady-state scenario discharges, especially those methods that employ validated models of the plasma response.

DIII-D is unique in its focus on feedback control of the access conditions for advanced scenarios. Control of q_{min} during the current ramp phase has been successfully developed using proportional-integral control to deal with sparse sensor data in time (Fig. 3-10) [Ferron 2006]. This method is very useful in improving the productivity of experiments by using feedback control to give reproducible target conditions in the presence of modest variability of the machine conditions. This method relies on the development of a target trajectory by other means. This reliance on prior knowledge and the limitation to control of a single parameter means it is not extensible to automated searches for new operating conditions.

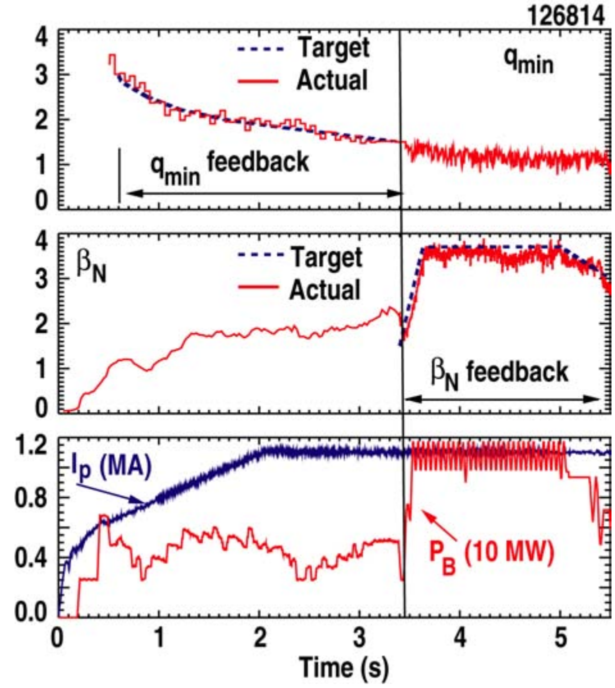

Fig. 3-10. Time evolution of a steady-state scenario AT discharge in DIII-D. During the discharge formation (before 3.4 s) neutral beam power is used as the actuator for feedback control of the evolution of q_{min}. In the last portion the discharge, the heating power is used to control β_N.

The plan for continuing research will focus on development of model-based controllers for the q evolution, and optimization of the control techniques to avoid MHD instability during the discharge formation. The time evolution of the poloidal flux, including resistive diffusion and noninductive current drive, is assumed to be governed by a partial differential equation. Controllers for processes governed by a PDE are at the state of the art in control method research. In addition, the plasma has the unique feature that the diffusion coefficient (resistivity) can be altered by the feedback system as a control variable. The control in this case is also much different than standard control methods. Usually sensors generate signals to be compared with targets, and an error vector is calculated, multiplied by gains, and used to drive the actuators. In the case of the target current profile control, the controller must project forward to determine what actions must be taken to assure that the target profile will be obtained within an allowed time window later in the discharge. As the current builds up, this assessment becomes successively refined and an accurate prediction of when the target will be reached is generated for the control system to prepare for the next control phase. This type of control is beyond the methods presently available and represents an exciting opportunity for the control community. It also seems to be required for future tokamaks, including ITER, to have any hope to access advanced scenarios. To achieve this goal, collaborations with Lehigh University on controller development and with CEA-Cadarache on modeling the current buildup phase have been established.

The plan for controller development entails the following steps:

- Open-loop validation of the plasma response models to the actuators available during the current ramp in DIII-D

- Open-loop validation in DIII-D of the forward projection of the current evolution in time using off-line calculations of the optimum trajectory
- Development of control methods to allow real-time evaluation and optimization of the evolution "in-flight" to facilitate closed-loop control
- Closed-loop tests in DIII-D of the target q controller

The goal is to have a prototype closed-loop system deployed on DIII-D at the middle of the five-year plan.

3.5.2. Control Issues at High Bootstrap Fraction

Control of plasmas with very high bootstrap fraction (>80%) present a unique challenge, similar to that of controlling the pressure in a burning plasma with high fusion gain. Any action taken with auxiliary current drive systems will result in changes to both the current profile directly and indirectly through the bootstrap current response to the heating. The change in the current profile on the resistive timescale may also change the transport, again leading to a change in bootstrap current. In the early part of the five-year plan, research into this type of control will be carried out in high q_{95} plasmas, which have >80% bootstrap current. Preliminary efforts along this line [Politzer 2005] have yielded two important observations. First, the constant current constraint, either by use of the central solenoid or eventually with feedback control, is important to stable maintenance of the operating point. Second, the coupling between the energy and particle transport and the current profile is strong enough to yield oscillations in the plasma. From these observations, it is clear that detailed characterization of the particle, energy, and current transport will be essential to operating steady-state scenarios near the stability boundary simultaneous with high bootstrap fraction. Later in the five year plan, the steady-state operating scenarios at high β_N will require this knowledge for stable operation. By the end of the five-year plan, sufficient power should be available to add in the even more challenging task of including burn control by taking part of the auxiliary power and simulating the fusion power response to pressure variations.

3.5.3. Steady-State Operation Control

Once the desired current and pressure profiles are created during the discharge formation (as described in Section 3.5.1), the control challenge is to maintain these profiles in steady state. The control challenges in DIII-D, ITER and DEMO differ significantly because of the differences in the available control actuators. In all cases, in order to enhance the fusion efficiency, it is desirable to minimize the amount of external current drive and heating power. In DIII-D, all of the heating and current drive is external, while future tokamaks may have substantial heating from α particles. In all cases, the amount of current drive available from any individual actuator will be small compared to the total plasma current, so external current drive is a relatively weak actuator. The challenge will be to maintain a pressure profile that produces bootstrap current that can be made to be consistent with the desired q profile with the addition of only a small amount of external heating and current drive.

This research requires several tools such as 10 s pulse capability, higher heating power, and the physics basis elements related to the coupling of the current profile and the pressure profile. Therefore, the plan is now only in broad outline form. The key elements will be:

- Control of the pressure profile through modification of the rotation and magnetic shear

- Assessment of the optimum simultaneous current and pressure profiles for steady-state operation using feedback control
- Development of burn control techniques in advanced scenarios using dedicated heating systems in a separate feedback loop to simulate the fusion power.
- Assessment of the fractional power above that needed for equilibrium to provide a specified level of control (maximum deviation of control parameter)

3.6. INSTABILITY AVOIDANCE AND CONTROL

Optimization of the performance of advanced scenarios will entail operation in proximity to stability limits. The measure of how near to these limits the scenario can run is the ability of the control system to determine in real-time where the operating point is relative to the limits and what the actuator capabilities are to avoid the limits or compensate for off-normal perturbations. One specific instability that must be routinely stabilized for the steady-state scenarios with high q_{min} is the resistive wall mode (RWM). The research program for stabilization of the RWM and a description of the general instability avoidance and mitigation control will be presented in the following sections.

3.6.1. Rotation and Active Feedback Control

RWM Stabilization Physics. Steady-state scenario plasmas in an AT fusion reactor and in ITER require stabilization of the low-n magnetohydrodynamics (MHD) kink mode for operation at normalized β above the no-wall stability limit [Jardin 1997, Hender 2007]. Conductive structures close to the plasma convert the kink mode into the slowly growing resistive wall mode (RWM), which theory [Bondeson 1996] predicts could be stabilized by sufficiently rapid plasma rotation. Research carried out on the DIII-D tokamak and other high β experiments has demonstrated that the RWM can be stabilized by plasma toroidal rotation [Garofalo 2002a, Sabbagh 2006], even with the relatively slow rotation resulting from near-balanced neutral beam injection [Reimerdes 2007, Tackechi 2007]. Measured rotation threshold are shown, for example, in Fig. 3-11.

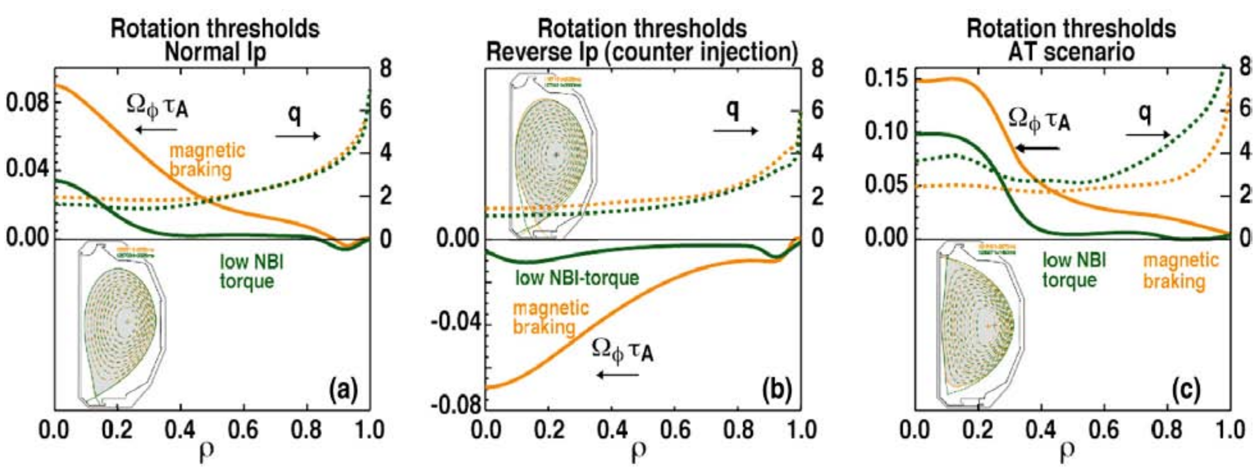

Fig. 3-11. Measured profiles of toroidal rotation and safety factor at the onset of RWM instability, with rotation slowed by resonant n=1 magnetic braking (orange curves) or by low neutral beam torque with minimized n=1 error fields (green curves). Cases shown include low triangularity plasmas with (a) co- and (b) counter-injected neutral beams, and (c) a high triangularity advanced tokamak plasma with q>2 everywhere.

However, DIII-D experiments also show that rotational stabilization provides only shallow stabilization. The stable n=1 RWM is always measured close to marginal stability, independent of the plasma rotation levels or the plasma beta, once beta is above the n=1 no-wall limit. For this reason, essential to maintaining rotational stabilization is the correction of magnetic nonaxisymmetries that could be resonant with the RWM.

If these nonaxisymmetries are not corrected, they can drive the marginally stable RWM to large amplitude, a phenomenon called "resonant field amplification" (RFA). In turn, the RFA applies a strong electromagnetic braking torque on the plasma and eventually can lead, depending on the momentum input and confinement, to torque balance instability and collapse of the plasma rotation as in the induction motor model by Fitzpatrick [Fitzpatrick 1998].

The physics details of what happens during and after the rotation collapse are the subject of active investigation. It is not clear whether the RWM becomes unstable as the rotation collapses, or the large amplitude RWM becomes unshielded by the plasma rotation and proceeds to form a magnetic island and seed a metastable tearing mode. Measurements of the internal structure of the magnetic perturbation are crucial to address this issue, and will require diagnostic upgrades in the next five-year period.

A similar issue arises even when the correction of resonant nonaxisymmetries is optimized, but the input torque from neutral beams is lowered to near zero using simultaneous co- and counter-injection while beta is above the no-wall limit. As the torque is reduced, a beta collapse is eventually encountered at some low value of the rotation because of the growth of a locked n=1 mode. Again, it is not clear whether it is the RWM becoming unstable at a rotation below a rotation threshold for stabilization, or it is the threshold amplitude of the magnetic error for plasma locking and reconnection that, with near-zero input torque, is lower than our ability to correct nonaxisymmetries. An improved capability to correct magnetic nonaxisymmetries requires additional power supplies (and possibly additional coils) and improved diagnostics in order to control individual I-coil segments as opposed to hardwired combinations of them.

Resolving the above described issues is essential in order to correctly determine, in the experiment, the rotation threshold for RWM stabilization. This determination, and the comparison of the experimental result to model predictions is essential to validating the RWM theory for reliable extrapolations to ITER and future devices.

Feedback Control of the RWM. DIII-D experiments have demonstrated that the magnetic feedback system for RWM control is effective at improving the error field correction in a plasma with beta above the n=1 no-wall stability limit [Garofalo 2002b]. This is the process called "dynamic error field correction" (DEFC). The feedback system is driven by the increased RWM response in the plasma to an intrinsic resonant error field. The increased response is detected as the plasma beta exceeds the n=1 no-wall limit. With large enough gain, the feedback system does not simply cancel the plasma response, but acts in such a way as to correct the error field that drives it.

Besides being used to quickly find the optimal correction for intrinsic error fields, the role of DEFC is also to maintain or quickly restore optimal correction during unexpected changes of the equilibrium field. These field changes include perturbations due to plasma instabilities having a magnetic component resonant (in the frequency and space domains) with the RWM, such as ELMs. In DIII-D advanced tokamak plasmas, it has been observed that some ELMs can drive the RWM to large amplitude. The

RWM tends to remain at finite amplitude long after the perturbing kick has vanished. This leads to magnetic braking of the plasma rotation and can eventually result in loss of stability even in discharges with large torque input. The results shown in Fig. 3-12 demonstrate the effectiveness of the DEFC for recovery from the perturbation caused by an ELM.

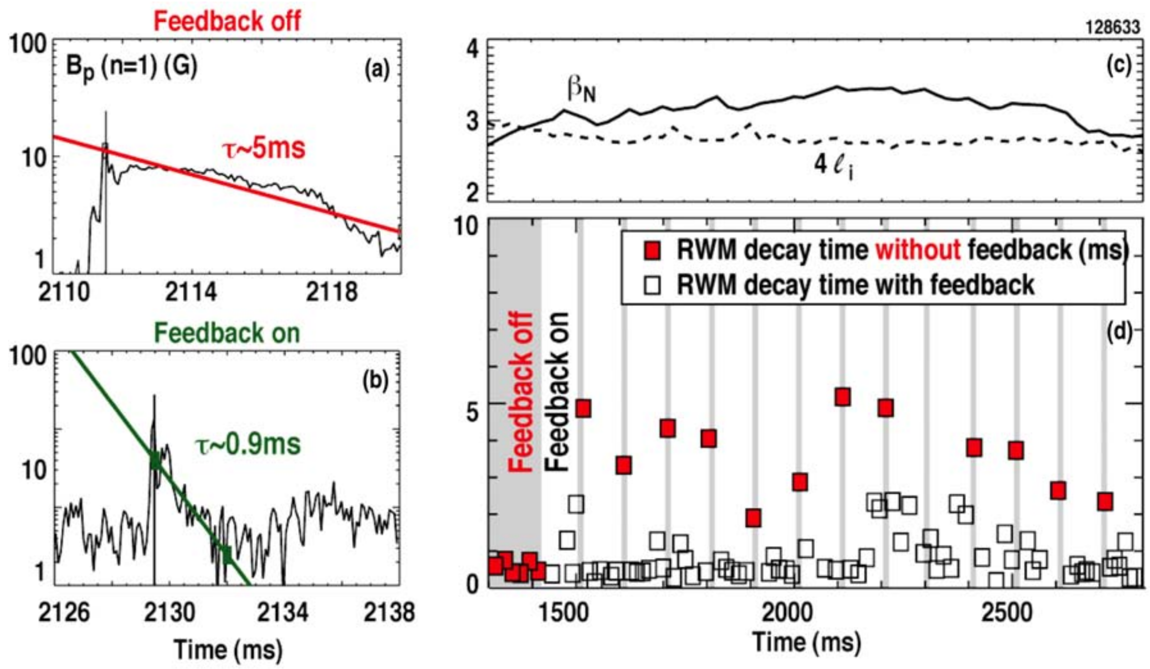

Fig. 3-12. Feedback control of stable RWMs that are resonantly excited by ELMs in a high-beta advanced tokamak plasma. The decay of the n=1 perturbation is shown for two successive ELMs (a) without feedback and (b) with feedback. (c) Normalized beta and estimated no-wall stability limit. (d) Decay times of many ELMs during the discharge, including ELMs with feedback control (open symbols) and without feedback control (orange filled symbols). Vertical gray bars indicate the time intervals with feedback turned off.

Active control using magnetic coils provides additional stability against these magnetic disturbances, but can require large currents at fairly fast response time. In DIII-D adequate protection against the n=1 RWM is provided by the I-coil powered by Switching Power Amplifiers (SPAs) [Garofalo 2006]. In a very high beta plasma where the n=2 and n=3 RWM may become marginally stable, the individual I-coil elements would need to be powered separately, for simultaneous application of n=even and n=odd control fields. The capability to individually feedback-control each I-coil segment may also be a crucial issue at the slow plasma rotation resulting from low input torque. Only the control field required to oppose the RWM field should be applied to the plasma. Any extra field with little shielding from plasma rotation could easily open a magnetic island.

At low input torque, feedback will need to act more promptly to a transient braking phenomenon. Right now, the delay of the power supplies (~500 μs) does not seem to be the limiting issue. If, at low input torque, the error field threshold for torque balance instability is below the detection threshold, the fastest supply would not be of help. Instead, improvement of the mode detection capability would be essential.

Hardware Upgrades. DIII-D research in both the areas of rotational RWM stabilization and feedback control of the RWM will require significant upgrades in the next five-year period. Improvements in the mode detection capability are required for: lower detection threshold, earlier detection, detection of toroidal mode number n>1, measurement of the internal mode structure. To address these issues, we will pursue the installation of additional magnetic sensors, the reduction of noise and improvement of rejection of axisymmetric field changes also for the existing sensors, the addition of sensors that can detect the RWM inside the plasma, such as multiple toroidal locations of SXR detectors and ECE radiometers, and the development of advanced algorithms such as Kalman filters or matched filter techniques to discriminate against unwanted signals in the mode detection system, including other MHD modes as well as true noise in the instrumentation.

Power supply improvements are required for control of individual I-coil segments for simultaneous application of n=even and n=odd control fields. The various I-coil connections and corresponding current limitations for the present set of 24 audio-amplifiers (AAs) and four SPAs is described in Table 2-4 of Section 2.4.3 of this document. In order to use individually-powered I-coil segments to carry out correction of the intrinsic error fields and dynamic error field correction in an AT DIII-D plasma, each I-coil segment would have to be energized by a relatively fast power supply with high-current capability. There are three possible approaches: (1) double the number of AAs and build a crossover network to add 700 A from audio-amplifiers to 1.7 kA from a SPA sub-unit powering each I-coil segment; (2) replace each existing AA with a higher current version (dc capability no longer required); (3) new 12-channel amplifier with intermediate current and bandwidth.

3.6.2. Comprehensive Control for Disruption-Free Operation

The ultimate goal of instability control is the reliable achievement of effectively disruption-free operation. Disruptions are the result of two broad categories of events: crossing of a deterministic stability boundary beyond which the control system cannot prevent growth of a disruptive mode, or nondeterministic failure of a critical machine subsystem which results in a loss of ability to maintain the discharge. Advanced tokamak, steady state plasmas are particularly challenging to the goal of disruption-free operation in that they are defined (among other things) by their operation beyond several well-known stability boundaries. Despite these challenges, because disruptive instabilities are still fundamentally deterministic, reliable stability control can in principle provide quality of service similar to high-performance aircraft and other highly-regulated open-loop unstable systems. While it is fundamentally impossible to guarantee fault-free operation of any system, the probability of a disruption-producing fault can be reduced to an economically attractive level, and the damaging consequences of such faults can be effectively mitigated. The viability of the tokamak and the AT concept in particular depend critically on demonstration of an economically acceptable level of reliability of control and fault response.

The 2009–2013 DIII-D program will include a significant initiative to develop the tools for “disruption-free” operation, and to demonstrate these solutions experimentally. The integrated solution will follow a layered response strategy: advanced controllers will enable robust operation beyond various (open loop) stability boundaries including the RWM no-wall beta limit and NTM thresholds, realtime algorithms will be developed to monitor distance past open loop boundaries and proximity to relevant (closed-loop) stability boundaries, intelligent corrective response algorithms will enable discharges to recover from control excursions that threaten to violate stability margins, robust disruption avoidance algorithms including soft-landing and rapid shutdown approaches will ensure that failed corrective action does not produce a damaging disruptive termination. Reliable and effective disruption mitigation will ensure that rare (system fault-driven) but potentially destructive disruptions are not permitted to significantly damage the device.

Advances in both computational platforms and codes will be required in order to achieve accurate disruption proximity identification. Active methods such as MHD spectroscopy may be needed to augment realtime stability calculation. Complex but reliable decision algorithms may be required to take the necessary realtime avoidance and correction actions in a wide range of operating regimes. In addition to fundamentally robust stability control physics, new approaches to maintenance of robust control operation as the operating regime changes (e.g., adaptive control or gain scheduling) may be explored. Significant study of disruption and effects mitigation physics will continue in order to define effective rapid shutdown approaches. Areas of disruption mitigation physics and development of new technologies required may include multi-valve and multi-species impurity gas injection, use of rupture disks, and combined use of nonaxisymmetric magnetic fields along with various forms of impurity injection.

This ambitious program will also require expanded diagnostics, such as a high degree of toroidal coverage by poloidal field probes, enabling full 3D magnetic field reconstruction. Improved internal profile measurements such as soft x-ray chords enabling full tomography could also contribute to both control needs and improved physics understanding of impurity transport. Improved hard x-ray diagnostics will greatly improve understanding of runaway electron production and confinement, and support the development of critical runaway mitigation techniques.

3.7. INTEGRATION OF STEADY-STATE HEAT AND PARTICLE FLUX SOLUTIONS

Advanced tokamak scenarios offer the best chance for optimization of fusion performance to make future tokamak power plants commercially attractive. However, the high power loads on the divertor structures expected for DEMO present a major challenge for steady-state operation. Therefore, for steady state (or even long pulse) operation of advanced scenarios, the high heat flow out of the plasma core must be handled without compromising the advanced scenario performance. While the high heat flux is the primary concern, it is also important to have regulation of the plasma density and impurities. For DIII-D experiments, the solutions must be in place to handle up to 300 MJ heat input per pulse with peak power levels approaching 30 MW by the end of the five-year plan. The present plan to deal with this heat flux in DIII-D is to explore three promising methods for achieving heat flux reduction at the divertor targets and improved density control:

- Nonaxisymmetric coil arrays
- Axisymmetric divertor flux expansion coils in the divertor region
- Modification of the divertor structures

These methods are discussed individually in the following sections along with the possible installation of carbon fiber composite (CFC) tiles.

3.7.1. Heat Flux Spreading by Resonant Magnetic Perturbations

Nonaxisymmetric applied magnetic fields, such as those from the DIII–D I-coils and C-coils, have been observed to split the usual narrow axisymmetric heat flux on the divertor into two or more spirals.

Recent DIII-D experiments, where a nonaxisymmetric field has been applied for the purpose of ELM suppression, have shown a reduction of peak divertor heat flux by about 30%, due to this splitting [Evans 2007] (Fig. 3-13). This splitting is predicted by field line tracing codes that incorporate the magnetic perturbations resulting from applied currents in I-coils and C-coils, together with measured intrinsic field errors. Images of the divertor in D_α light show structures consistent with the field line tracing calculations (Fig. 3-14).

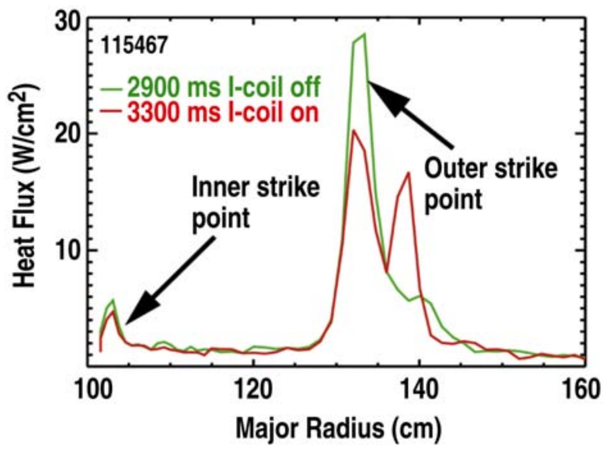

Fig. 3-13. Radial heat flux profiles taken before the magnetic perturbation pulse (green curve) and during the perturbation pulse (red curve).

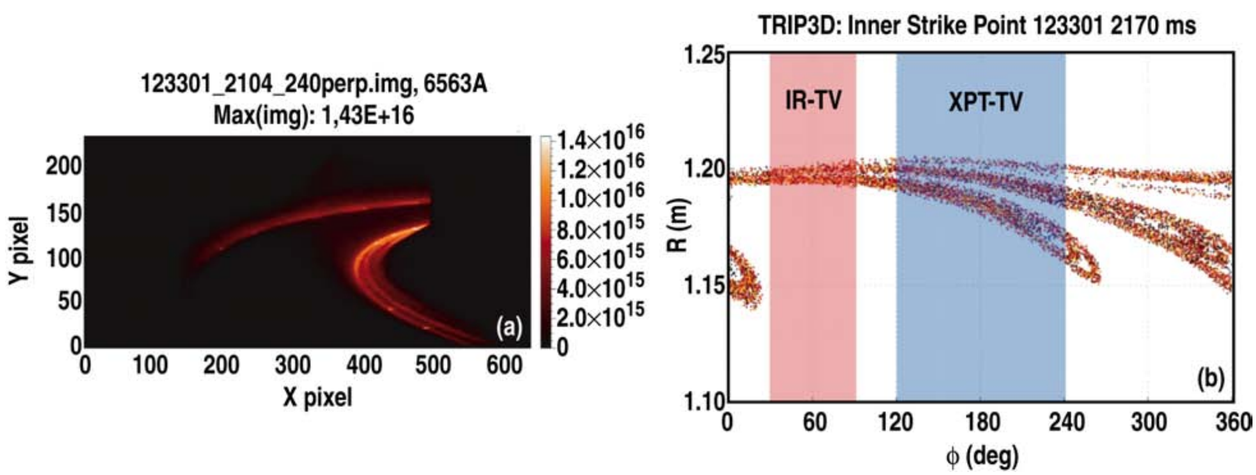

Fig. 3-14. (a) Tangential X-point TV image of filtered D_α light displays three striations along the inner strike point during I-coil operation. (b) Calculations of the magnetic field lines as they exit the inner divertor target, including C-coil, I-coil, and field errors, show significant n = 1 and n = 3 components. Shaded regions cover the approximate field of view of the current infrared camera view (red) and the tangential X-point camera view shown in (a) (blue).

The application of nonaxisymmetric magnetic fields offers a means to control the effective divertor area over which power is deposited, although local increases in heat flux could be possible. A more reliable approach to reduce the average peak heat flux would be to rotate the toroidal phase of the spiral pattern. This leads to a time average of the heat flux over the full radial extent of the spiral. A low frequency (≤ 10/n Hz, where n is the toroidal mode number of the nonaxisymmetric field) should suffice.

The existing I-coils and C-coils are not well suited to this application. Both coil sets are composed of six toroidally distributed coils, which can only make magnetic fields of three or fewer toroidal periods. This implies that only fields with $n = 1$ and 2 symmetry can be rotated. Unfortunately, these harmonics interact strongly with the least stable MHD modes, and therefore need to be carefully controlled to avoid destabilizing MHD modes and reducing the plasma rotation. A new coil set having $n \geq 4$ is proposed for both this application and ELM suppression. Sets of $n = 8$ or 12 would fit most naturally in DIII–D. It is possible to install 12 internal coils per horizontal row with minor extensions of existing I-coil design and technology. However, connecting power to 12 coils of a row below the vessel midplane is considered very challenging, because of severely limited space. The decision as to which coils to install and the timeline will be determined by the ELM suppression mission (see Section 2.2). The heat flux reduction application will make use of whatever coils are installed for the suppression of ELMs.

3.7.2. Heat Flux Reduction Using Divertor Poloidal Flux Expansion Capability

The poloidal flux expansion near the outer divertor target can be modified by axisymmetric coils placed close to the strike point. Such coils provide the capability to change the flux expansion without altering significantly the geometry of the main plasma. This would be valuable because it isolates the effects of flux expansion from those of plasma shape changes. Because this method works well only for a narrow range of strike point locations relative to the coils, it will be studied further during the early part of the five-year plan. Implementation would only be initiated toward the end of the five-year plan, after the final divertor geometry for the bulk of the advanced scenario development is determined.

3.7.3. Improvements in Density Control and Radiated Divertor Performance

Modification of the lower divertor geometry by installation of a barrier that isolates the inner divertor target from neutral particle flux at the outer divertor target is under consideration. This modification of the lower divertor should improve particle control in order to control both the fuel density and the impurities injected to radiate power for heat flux reduction. Presently, the private flux region between the inner and outer targets in the lower divertor is open. This openness allows the unimpeded flight of both hydrogenic and impurity neutrals from the outer divertor target to the inner divertor target. Because there is no active pumping capability at the inner divertor target of the lower divertor, density and impurity control can be compromised. A barrier that physically isolates the inner and outer strike points in the lower divertor offers a credible way to block neutral particle migration between the two lower divertor targets.

To achieve better density control, the access of neutral fuel particles to the high field side of the plasma across the lower divertor must be reduced. This is especially important for a double-null divertor shape, the primary shape used in advanced tokamak scenarios. On the high field side, the probability of particles entering the main plasma is higher, because the characteristic electron temperature for the inboard SOL plasma of a double-null plasma is less than that of a comparable single-null plasma. The SOL density profile is also narrower for the double-null plasma. The inner divertor target offers little resistance in preventing hydrogenic neutrals from reaching the SOL on the high field side of the double-null plasma.

The main method proposed for reducing divertor power loading is to "seed" the divertor plasma with impurities that radiate the conducted power before it can reach the divertor targets. For this to be practical, the confinement and the stability of the plasma cannot be compromised by leakage of the impurity into the main plasma. Previous studies have shown that the concentration of impurities in the divertor can be increased by raising the flow of deuterium ions into the divertor by a combination of upstream deuterium gas puffing and pumping at the divertor targets [Schaffer 1997]. The enhanced deuterium particle flow toward the divertor targets exerts a frictional drag on impurities, inhibiting their escape from the divertor. This approach is more effective in a closed divertor configuration, because the increased baffling inhibits the flux of neutral impurity atoms from the closed divertor region back into the main chamber.

The same arguments for the beneficial effects of a barrier to neutral fuel particle fluxes between the inner and outer divertor legs in the lower divertor hold for neutral impurity fluxes. The unimpeded flow between the inner and outer targets allows neutral impurities access to the high field side of the plasma, which can lead to impurity accumulation in the main plasma. A physical barrier between the inner and outer targets would be advantageous in inhibiting impurity migration in the lower divertor.

A properly designed barrier can also aid heat flux reduction by increasing the wetted area of the divertor targets. The concept is to minimize the angle between the divertor structure and the separatrix field lines. A smaller angle gives a larger wetted area in the divertor and lower peak heat flux. The peak heat flux can be made arbitrarily low, but finite ion gyro-orbits may lead to a practical lower bound. The present upper dome is not useful for testing this technique, because the dome configuration does not allow the required measurements, e.g., infrared camera measurements of the heat flux distribution are not possible.

3.7.4. Proposed Hardware Modifications

As discussed above, three significant new hardware initiatives are proposed for enhancing the heat and particle flux handling capabilities for advanced scenarios. The nonaxisymmetric coil design is primarily motivated by the ELM suppression goals and is discussed in Section 2.2.2. Here the proposals for axisymmetric coils for modifying the flux expansion and for a lower divertor baffle for a neutral particle barrier are described.

Coils for Controlling Flux Expansion. Independent control over divertor flux expansion is possible in DIII-D, with some modification to the present lower divertor. A coil array is proposed for this application that consists of four coils located under the protective tiles of the lower outer shelf. The location of these coils is shown in Fig. 3-15. Although the four-coil array shown is a conceptual design and is by no means optimized to produce the maximum differential in flux expansion, this configuration can produce a variation of ≈3.3 without a significant change in the shape of the plasma (Fig. 3-15). The current through any of these coils is limited to ≤10 kA. The main technical problems to be resolved are redesign of the graphite tiles to protect the coils from the plasma, space for the coil leads to pass into and out of the vessel, and design of adequate mechanical support structure to react the electromagnetic forces.

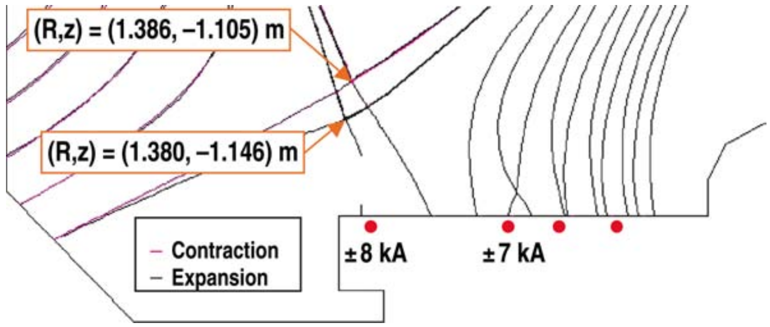

Fig. 3-15. The poloidal cross-sections of the contraction and expansion cases using the new divertor coils. The flux surfaces in the SOL are 10 mm apart at the outer midplane. The maximum flux *expansion* = 16.09 and the maximum flux *contraction* = 4.83. The contraction-to-expansion ratio is ≈3.3

Improved Closure of the Lower Divertor. As discussed above, a barrier in the lower divertor that inhibits the transit of neutrals across the lower divertor is highly recommended to improve the density control and the trapping of seeded impurities in the divertor during radiating divertor operation. The barrier would be a graphite-protected structure in the lower divertor that meets the following criteria:

- Block transit of neutral particles between the inner and the outer targets
- Configuration that allows straightforward interpretation of the heating profiles in the vicinity of the inner and outer divertor targets, e.g., straight-edged when projected onto the poloidal plane
- Withstand moderate heat loading for the duration of the shot
- No pumping capability required
- Does not allow neutral particle leakage between divertor targets underneath the structure.
- Does not to interfere with the plasma configurations used for advanced scenarios

A major change to the lower divertor configuration as outlined above should not be done early in the five-year plan. A careful assessment of particle control in advanced scenarios with the current divertor configuration should be completed to confirm the need for the barrier. If needed, the divertor modification could take place following the 2010 campaign.

3.7.5. Diagnostic Upgrades

Three key diagnostic upgrades are recommended for the evaluation of the core-edge integration work on advanced scenarios—an IR camera system to view the entire set of plasma-facing components, an improved bolometry system, and expansion of the Langmuir probe arrays. The motivation for these upgrades is outlined in the following sections.

Whole-Vessel IR Camera System. The measurement of the surface temperature evolution on tiles in both upper and lower divertors and along the nondivertor vessel walls is important from the standpoint of understanding divertor physics and for protection of vessel tiles exposed to divertor power loading. Important issues that require good IR imaging are:

- Distribution of ELM energy between upper and lower divertor, inner wall, and outer wall
- Toroidal asymmetries in heat flux during operation of nonaxisymmetric coils

- Observe the heat flux move from the divertor targets to the main chamber during radiative divertor operation
- Local heating of divertor tiles due to asymmetric power deposition during high power, long pulse discharges (up to 300 MJ)
- Monitoring antennas and limiters.

A three camera system that can monitor almost the entire vacuum vessel is proposed. Coverage of the divertor region is planned for 2009 with additional capability to be added incrementally.

Auxiliary Divertor Bolometer Arrays. Two bolometer arrays are used presently to determine the distribution of the radiated power in the plasma. The lines-of-sight of the arrays have an obstructed view of the divertor regions due to the overhang of the outer divertor baffles and provide relatively poor spatial resolution of the radiated power distribution in the divertor regions that they can view. These deficiencies result in a significant degree of uncertainty in the spatial distribution of the radiated power in either divertor. Analysis of plasmas in the radiating divertor operating modes is particularly vulnerable to these deficiencies. One possibility is to install an auxiliary bolometer array under the upper outer divertor baffle and a second array inside the upper dome. Additional bolometer coverage is planned for installation in 2010–2011.

Expanded Langmuir Probe Capability. Fixed Langmuir probes embedded in the divertor tiles and centerpost tiles are the only way to measure electron density, temperature, and particle flux near the inner and outer divertor strike points. It is essential that this Langmuir probe capability be maintained in the divertor, since these measurements are needed in work that involves modeling of the divertor plasma behavior.

Particle recycling in the main chamber and related impurity generation can be a significant contributor to core plasma fueling and core impurity content. Measuring the particle fluxes to the wall with Langmuir probes is an important technique to advance our understanding of the spatial and temporal distributions of the particle sources in a tokamak. In DIII-D, areas of the most intense plasma-wall interaction outside the divertor were identified to be the toroidally symmetric surfaces of the inner wall, and the divertor baffles of the lower and upper divertors, and the toroidally nonsymmetric bumper and guard limiters along the outer equatorial plane. To further advance our understanding of plasma-main chamber wall interaction, it is proposed to increase the coverage of the main chamber wall with Langmuir probes in these crucial areas.

The standard Langmuir probes in the divertor are highly vulnerable to erosion in operations with up to 30 MW and 300 MJ per discharge, leading to a short useful lifetime. Investigations into probe designs that can withstand the high heat loading in the divertor are well underway. It is recommended that probe arrays be added at multiple toroidal locations in the divertor in order to explore nonaxisymmetric issues in the divertor and add redundancy, so that the loss of one probe does not compromise key measurements. New Langmuir probes will be added incrementally throughout the five year plan.

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4. DIII-D CONTRIBUTIONS TO A LONG-TERM DEVELOPMENT PLAN

4.1. CONSIDERATIONS FOR THE FUTURE

On November 26, 2007, the U.S. signed the ITER agreement. We are now well on our way to the next step in Fusion Energy Science research. As described in Chapter 2, DIII-D is and will continue to be actively engaged in preparation for and support of ITER. However, the Fusion Science Community has long recognized that ITER is just one part of a larger development strategy aimed at the eventual realization of fusion energy production.

Following the signature of this agreement, the Fusion Energy Sciences community was challenged by the Undersecretary of Energy for Science: “To assist planning for the ITER era, it is critical that FESAC identify the issues arising in a path to DEMO, with ITER as a central part of that effort.” A FESAC panel, led by Martin Greenwald, was formed to respond to these charges, and recently concluded the requested analysis of the knowledge base needed to be ready to build a DEMO device.

The panel’s report [Greenwald 2007] identified a set of 15 scientific and technical questions, organized into three themes (Table 4-1), which must be resolved to proceed to a DEMO device. The report goes on to identify “gaps” that might be filled by future initiatives.

It is the role of present day devices, including DIII-D, to establish the necessary scientific basis to proceed to these initiatives. The list of issues is particularly well aligned with the DIII-D mission *to develop the physics basis for the optimization of the tokamak approach to fusion energy production*. The planned research program will continue to address all but one of the issues in Theme A (Creating predictable high-performance steady-state plasmas), as well as the first Theme B issue (Plasma-wall interactions).

We envision working toward the eventual realization of a DEMO device patterned after the ARIES AT study [Najmabadi 2006]. As an intermediate step, a device such as FDF [https://fusion.gat.com/fdf/Home] is envisioned to complement ITER through demonstrations of steady-state high performance operation and control, tritium self-sufficiency, blanket testing, and other issues (Fig. 4-1).

The following sections are organized around these issues. The topics shown in the tables were separately identified as areas of specific need for FDF and DEMO that will be addressed, fully or in part, by DIII-D in the next five years. We wish to particularly highlight issue 3, “Validated Theory and Predictive Modeling.” This area is strongly aligned with our emphasis on scientific understanding, and permeates all areas of our research. This goes beyond the Fusion Science program described in Chapter 5. Both the ITER Physics and Steady-State Scenarios research also strongly emphasize scientific understanding. An overriding goal of the DIII-D Research Program is to work toward an accurate and fully predictive capability that can be applied to the design and execution of experiments on ITER, FDF, DEMO, and any other future burning plasma device. The embodiment of such understanding will be integrated models, as described in Chapter 6.

Table 4-1
Scientific and Technical Issues Requiring Resolution Prior to Proceeding to DEMO

Issue	Addressed in DIII-D
Theme A. Creating predictable high-performance steady-state plasmas	
1. Measurement	√
2. Integration of high-performance, steady-state, burning plasmas	√
3. Validated Theory and Predictive Modeling	√
4. Control	√
5. Off-normal Plasma Events	√
6. Plasma Modification by Auxiliary Systems	√
7. Magnets	
Theme B. Taming the plasma material interface	
8. Plasma-Wall Interactions	√
9. Plasma Facing Components	
10. RF Antennas, Launching Structures and Other Internal Components	√
Theme C. Harnessing fusion power	
11. Fusion Fuel Cycle	
12. Power Extraction	
13. Materials Science in the Fusion Environment	
14. Safety: *Demonstrate the safety and environmental potential of fusion power*	
15. Reliability, Availability, Maintainability, Inspectability	

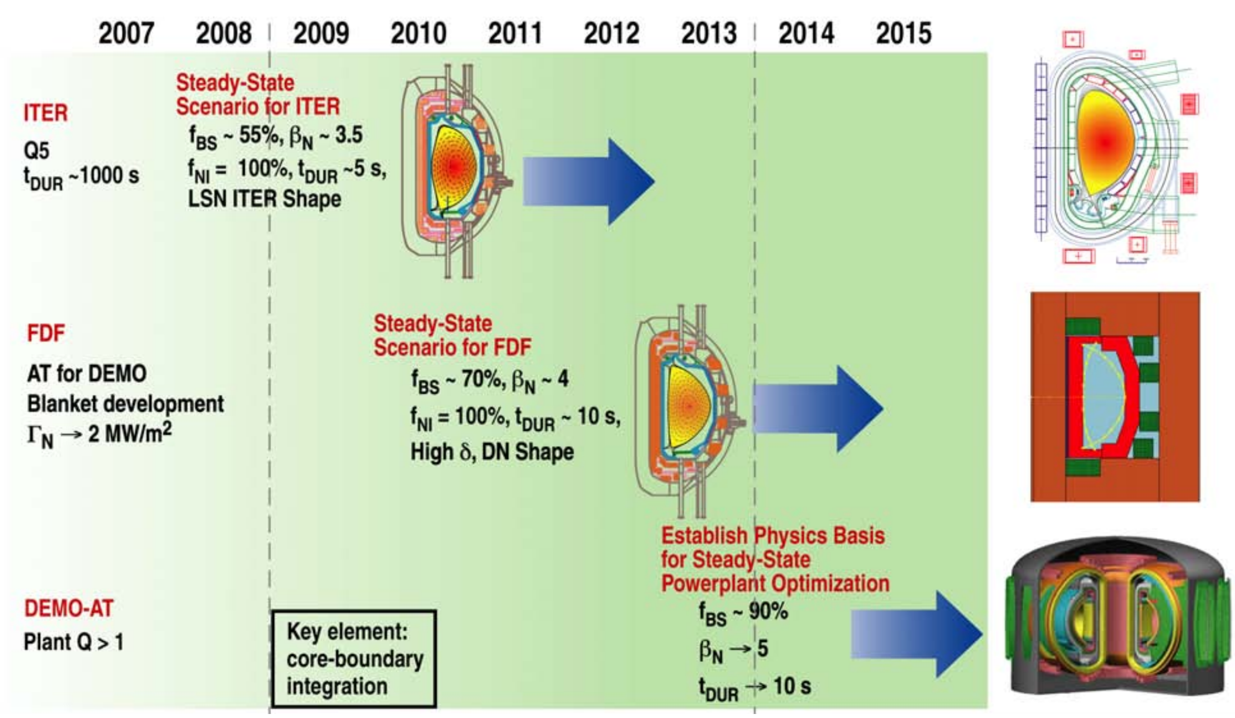

Fig. 4-1. DIII-D five year plan (2009–2013) emphasizes validation of the physics basis for Advanced Scenarios in future devices.

4.1.1. Measurement

Make advances in sensor hardware, procedures and algorithms for measurements of all necessary plasma quantities with sufficient coverage and accuracy needed for the scientific mission, especially plasma control.

The DIII-D five-year program plan provides for an active program of development and implementation of diagnostics and measurement techniques (Chapter 8). These diagnostics are of great importance for the scientific and technical missions of DIII-D. In many cases, diagnostic development on DIII-D can provide a test bed for prototypes of diagnostics designed to meet the challenges of a burning plasma environment in next-step experiments.

Diagnostic development in DIII-D is motivated by the needs of research areas. In particular, measurement techniques and their application to control are considered within Sections 4.1.2 and 4.1.4, below.

4.1.2. Integration of High-Performance, Steady-State, Burning Plasmas

Create and conduct research, on a routine basis, of high performance core, edge and SOL plasmas in steady-state with the combined performance characteristics required for Demo.

Tokamaks require significant plasma current for confinement. In ITER, this may be as much as 17 MA, and future burning plasma tokamaks will operate in the same range. The "conventional" H-mode scenario, ITER's primary operating mode, is inherently limited in duration by the inductive drive used to produce and maintain this current.

In a fusion power plant, the situation will be somewhat different. It is highly desirable, if not absolutely essential, that such a device operate in steady-state. Even in post-ITER experiments, such as FDF, we envision 2 week pulse duration with a 30% duty cycle. This creates a host of challenges, many of which will be addressed in the DIII-D program (Chapter 3). Although the most visible product of this research, as called for in the Greenwald report, will be demonstrations of steady-state scenarios, we believe the most *important* product will be its contributions to scientific understanding, which in turn contribute to the predictive capability discussed in Section 4.1.3.

This is also the area where "multiphysics issues" will be addressed, since these scenarios will challenge limits in many areas, including stability, confinement, power and particle handling. So integration is an emphasis here.

In developing the Advanced Tokamak (AT), or high beta steady-state scenario, we must first define the operating point. Conceptually, there are multiple approaches with several common features. First, all of the current in the plasma will be driven noninductively, by a combination of bootstrap and external sources, in order to remove the restrictions imposed by the transformer current drive. The external sources available in DIII-D during this program plan include neutral beam current drive (NBCD, both on- and off-axis), electron cyclotron current drive (ECCD, both on- and off-axis), and fast wave current drive (FWCD, centered at the magnetic axis). Second, the plasma must operate at high β. This is important not only to maximize fusion performance, but also to maximize bootstrap current. The elevated q scenarios in particular require very high bootstrap current, up to 90%. Third, the current, pressure, and rotation profiles must be optimized in such a way as to allow steady, stable operation close to stability and other limits. They must be sufficiently controllable as to avoid transient events moving the operating point past these limits. Finally, the scenario must be designed while taking into account the fact that the vast majority of the heating power will be provided by fusion α's, leaving little or no ability to exercise control over the plasma state through heating. Although this is the case for all burning plasma scenarios, whether pulsed or steady, this is much more challenging in a steady-state scenario due to its inherent control requirements. Definition of this operating point, therefore, requires sufficient understanding of the multiphysics couplings to develop self-consistent integrated scenarios that minimize control requirements.

DIII-D has studied two classes of potential noninductive scenarios. The "high ℓ_i" scenarios have peaked current profiles, and require significant current drive, about 60% of the total, near the magnetic axis. The remaining ~40% of the current would be supplied by bootstrap. It is believed that this is feasible, since on-axis current drive is significantly easier than off-axis. This scenario has the advantage of being able to reach high β without active RWM stabilization.

The leading scenario in DIII-D, and the one that has been studied most thoroughly for application to ITER and FDF, is characterized by elevated q profiles with a broad, weakly reversed, magnetic shear region centered on the axis. This scenario has been demonstrated with β_N up to 4, and projects to β_N as high as 5. Most of the current in this scenario is produced by bootstrap, with about 10% of the current supplied by off-axis ECCD and/or NBCD. Additional control can be exercised by application of on-axis FWCD. This scenario requires RWM control to reach β_N above the no-wall limit to the n=1 kink mode.

The key to obtaining such scenarios is controllability. The existing set already allows access to this scenario, and has demonstrated stationary, fully noninductive operation on a confinement time scale and nearly (~90%) noninductive operation on a current penetration time scale, at high β_N. The planned current drive and stability control tools are expected to allow thorough exploration of the relevant parameter space. As previously mentioned, this experimental effort will continue to be closely coupled with integrated modeling in order to develop a predictive capability that can be applied to ITER, FDF, DEMO, and eventually, a power reactor.

In 5–10 s discharges in DIII-D, heat flux control is in general not a concern, and has not been emphasized in previous experiments. Looking forward to future burning plasma ATs, however, there is a great deal of concern. There is at present no demonstrated technique for handling the heat flux anticipated in the divertor. Although radiative divertor techniques have been successful in some scenarios, the low density and collisionality in the AT make this very challenging. As described in Section 3.7, these issues are addressed by several device upgrades in the five-year plan.

The final challenge for AT scenarios is to integrate all of the above considerations, along with others noted in Table 4-2. We believe that development of both integrated physics-based models for the plasma behavior (Chapter 6), and a sophisticated Plasma Control System (Chapter 7) will allow us to optimize use of the tools described above.

Table 4-2
Many Issues Specific to Steady-State Operation are Addressed in the DIII-D Five-Year Program Plan

Topic	Five-Year Plan Section(s)
Operating point	Chapter 3
• High β essential for bootstrap and fusion performance	
• Optimized profiles (current, pressure, rotation)	
• Limited controllability: most heating power from fusion α's	
Noninductive current drive	3.4.3
• Bootstrap (≤90%)	
• NBCD (on/off-axis)	
• ECCD	
• FWCD	
Divertor power handling	3.7
• High power/heat flux	
• ELMs	
• Radiative divertor at low density and collisionality	
Fueling and exhaust	3.7
• Controlling the fuel mix (gas valves and pellets)	
• Ash/exhaust removal (divertor cryopumps, must be consistent with nuclear environment)	
Control	3.5, 7
• Model-based: benefits from predictive modeling	6
• Actuators (current, pressure, rotation,…)	
— All CD tools also heat; NBI is particle and momentum source	3.4.3
— Shape (FDF has κ=2.3)	3.4.1
— Fuelling (see above)	3.7
— Stability control	
» RWM	2.4, 3.4.1
» ELMs	4.1.7
• Diagnostics (actuators) must produce real-time information, preferably profiles	8.1
• Burn control simulations	3.5.2

4.1.3. Validated Theory and Predictive Modeling

Through developments in theory and modeling and careful comparison with experiments, develop a set of computational models which are capable of predicting all important plasma behavior in the regimes and geometries relevant for practical fusion energy.

In future burning plasma devices, each long pulse discharge will be relatively expensive and will require advanced planning. In addition, off-normal events that would not be considered a problem (and may even be deliberately triggered as part of an experiment) in a current device could be damaging. These considerations both point to the importance of an accurate capability for predictive modeling of the entire discharge, both spatially and temporally. The results of such a capability would provide essential contributions not only in the design of new devices, but also in development of operating scenarios.

Development of such a capability requires sufficient understanding of the underlying individual plasma physics issues, as well as an understanding of integrated multiscale and multiphysics issues. Individual fusion science issues have been and will continue to be a major emphasis of the DIII-D program, and are described in Chapters 5 (Fusion Science) and 6 (Integrated Modeling). Research supporting ITER (Chapter 2) requires targeted, near-term research on DIII-D. In doing so, we emphasize understanding as a key component of our contribution to ITER, since that enables transfer of our results to ITER. Much of the integration of individual scientific issues, and thus the multiphysics component of this gap, takes place in the Advanced Tokamak (Chapter 3) research program. In particular, increased understanding of multiscale, multiphysics coupling is essential to any demonstration of AT scenarios and their transfer to ITER, FDF, DEMO, *etc*. The embodiment of progress here will also have impact in development of sophisticated control systems for DIII-D and beyond (Chapter 7).

Table 4-3 lists some physics issues needing resolution for a fully predictive capability to develop hardware and discharge scenarios in future burning plasma devices. Most of the issues discussed in other parts of Section 4.1 apply here as well.

Table 4-3
Some Physics Issues Needing Resolution for Development of Fully Predictive Capabilities

Topic	Section(s)
Transport in all channels (ion and electron thermal, particle, momentum) • Theory-based understanding • Control and integration	5.2, Chapters 6 and 8
Multiscale, multiphysics coupling	Chapters 3, 5, 6
Energetic particles • Handling large populations • Instabilities driven by EPs • Affect on transport in all channels • Measurement • Control and integration	5.4, 6.2.2.3, 8.6
… Many other issues …	4.1

The greatest challenge in developing this predictive capability is transport. Over the last 20 years, much progress has been made in understanding ion thermal transport, to the point where we find good agreement in many detailed comparisons with theory-based models. However, our understanding of the electron thermal, particle, and momentum transport channels is at a much lower level, and theory-based models can accurately predict none of these over a wide range of parameters.

DIII-D is well positioned to address these areas. Currently, DIII-D has arguably the most comprehensive array of turbulence diagnostics in the world, with measurements over a wide range of length scales, spatial resolution, and fluctuating fields (density and electron temperature). As described in Chapters 5 and 8, the diagnostic set will continue development in these directions. DIII-D also has a broad range of actuators to probe the individual transport channels (heat, particle, and momentum).

These tools will be systematically applied to fundamental turbulence studies aimed at definitive validation of theory-based models (Chapters 5 and 6). The DIII-D experimental program is closely coupled with the Theory and Modeling communities, and we will continue to work together to improve these models together.

Another area where challenges remain in predicting behavior is the interaction of energetic particles with the thermal plasma (Section 5.4). In most current tokamaks, such as DIII-D, these are introduced by neutral beams or accelerated by injected waves and are in the tens or hundreds of kiloelectron volts. In a reactor environment, with a large population of fusion α particles born at 3.5 MeV, the situation will become much more interesting and more challenging.

Although our understanding of MHD stability has become rather advanced, there are several outstanding issues. Many of these are discussed in following sections. One class of instabilities, the Alfvén eigenmodes, are particularly relevant here, since they are driven by and strongly affect the distribution of energetic particles. Recent experiments have made significant progress in validating models for particular classes of these instabilities, but a detailed theory that predicts how energy and particles are redistributed by AE modes remains elusive. Efforts in this area will continue with the same approach described above for transport: An expanded diagnostic set (Sections 5.4, 8.6) will allow increasingly sophisticated validation experiments, with the results being used to further develop the models.

In each of these areas, the ultimate goal is to achieve a predictive understanding, and some ability to control these affects in a burning plasma device. Where our ability to control these effects is limited, predictive understanding will at least allow us to include the important physics in the design of integrated scenarios.

4.1.4. Control

Investigate and establish schemes for maintaining high-performance, burning plasmas at a desired, multivariate operating point with a specified accuracy for long periods without disruption or other major excursions. (Provision for sensors is included under issue 1 and for actuators under issue 6.)

The vast majority of DIII-D efforts in this area are described in other sections. The development of steady-state scenarios, described in Section 4.1.2, heavily relies on these techniques. The Plasma Control system development is described in Chapter 7. Table 4-4 lists several research areas where DIII-D will make a contribution.

The main area not addressed elsewhere is control of neoclassical tearing modes (NTM). It is hoped that these can be avoided in AT scenarios simply by not allowing the current profile to evolve to an unstable state at high β. Since we have observed higher order NTMs (*e.g.*, m,n=3,1) appearing at high q, this approach may not be ultimately successful, and NTM control techniques might even be needed in steady-state scenarios. The main approaches to NTM control involve off-axis ECCD. Since this is planned for ITER, only limited additional development may be needed to extend the technique to FDF and DEMO.

Table 4-4
Additional Considerations for Control that are
Addressed by the DIII-D Five-Year Program Plan

Topic	Five-Year Plan Section(s)
Plasma Control	Chapter 7
NTM stability control • Dependence on ECCD width and alignment • ECCD modulation • Rotating n=1 fields to prevent mode locking	2.3
Advanced Tokamak relevant techniques	See 4.1.2
ELM control	See 4.1.7
Large-scale off-normal events	See 4.1.5

4.1.5. Off-Normal Plasma Events

Understand the underlying physics and control of high-performance magnetically confined plasmas sufficiently so that 'off-normal' plasma operation, which could cause catastrophic failure of internal components, can be avoided with high reliability and/or develop approaches that allow the devices tolerate some number or frequency of these events. (Because of their implications and importance, these "off normal events" are called out separately from the control issues listed above).

One of the most damaging events potentially facing a reactor class device is a disruption, where the plasma current rapidly drops to zero, with large thermal and mechanical loads stressing the device. On the other hand, a DIII-D class device is an ideal laboratory in which to study disruptions and their effects, as well as techniques to prevent and mitigate them. DIII-D, operating at significantly lower stored energy, is not damaged by disruptions, and experiments are not usually impacted by a few disruptions (no repairs or cleanup discharges are usually required). Table 4-5 lists some of the issues surrounding disruptions that are addressed in the DIII-D research program.

Table 4-5
Avoidance and Mitigation of Large-Scale Off-Normal Events

Topic	Five-Year Plan Section(s)
Disruption avoidance and mitigation • Prediction/detection • Avoidance • Mitigation – Runaway electrons	2.7, 7.3 (in particular, 7.3.8)
Disruption effects • Mechanical loads • Thermal loads	2.7

Often tokamak discharges disrupt because we often intentionally challenge known stability limits. Most disruptions occur when the plasma crosses a known stability boundary, so a demonstration of disruption avoidance might be done in DIII-D with a long series of repeated, disruption-free discharges. In general, as our control and predictive ability both increase (as discussed throughout this chapter), our ability to avoid disruptions will improve as well. However, disruptions might still occur due to control failure, so development of mitigation techniques will continue as well.

DIII-D has done pioneering work in the area of disruption mitigation with massive gas puffs. The goal is to inject enough impurity gas into plasma to more gently, but rapidly, eliminate the plasma current while maintaining a high enough electron density to prevent runaway electrons. Continued efforts with massive gas puffs, as well as several other techniques are under consideration in this plan.

Having a technique to mitigate disruptions is necessary, but not sufficient. A companion goal is for the Plasma Control system to "know" when to initiate mitigation. Algorithms for fault detection and response, described in Section 7.3.8, are being developed for this and related applications.

4.1.6. Plasma Modification by Auxiliary Systems

Establish the physics and engineering science of auxiliary systems which can provide power, particles, current and rotation at the appropriate locations in the plasma at the appropriate intensity.

Tools for plasma modification are a particular strength of the DIII-D device (Table 4-6). The long-range plan described in this document includes several initiatives to strengthen present capabilities as well as add new ones. The comprehensive diagnostics set will facilitate detailed scientific exploration of the effect of these systems both individually and as integrated into advanced scenarios.

All of these tools will be integrated into the Plasma Control system (Section 4.1.4)

Table 4-6
Tools for Plasma Modification

Tool	Five-Year Plan Section(s)	Comment	Power	Particles	Current	Rotation
Neutral Beams (on and off-axis)	3.4.3, Chapter 10	Two beamlines will have off-axis capability	√	√	√	√
ECH/CD	3.4.3, Chapter 10	On- and off-axis Increased power	√		√	
Fast Wave	3.4.3, Chapter 10		√		√	
Pellet injection	Chapter 10			√		
Gas valves	Chapter 10			√		
Divertor and cryopumps	3.7, Chapter 10	Optimized configuration		√		
Bootstrap	3.4.3, Chapter 10	Connects current profile modification with kinetic profiles			√	
Nonaxisymmetric coils	3.6.1, Chapter 10	RWM and error field control				√

4.1.7. Plasma-Wall Interactions

Understand and control of all processes which couple the plasma and nearby materials.

Plasma wall interactions are listed in the Greenwald Report as an issue for DEMO. However, many of these issues are not yet resolved sufficiently well for ITER, and so are for the most part included as part of DIII-D's ITER Physics research area. These issues, shown in Table 4-7, will have applications potentially in all subsequent plasma devices.

Table 4-7
The DIII-D Program Plan Includes Major Efforts Aimed Toward Understanding and Control of Interactions Between the Plasma and the First Wall

Topic	Five-Year Plan Section(s)
Plasma-wall interactions	
• Erosion, co-deposition and tritium uptake	2.8.1, 5.6.3, 6.2.4, 6.3
• Hot wall operation	2.8.1, 5.6.3, 6.2.4, 6.3
• Hydrogenic removal	2.8.1, 5.6.3, 6.2.4, 6.3
Transport of energy and particles between the boundary and the wall	
• SOL flows	2.8.1, 2.8.3, 5.6.2, 6.2.4, 6.3
• Material migration	2.8.3, 6.2.4, 6.3
• Thermal and particle loads to the first wall	5.6.1, 6.2.4, 6.3
Off-normal plasma events	
• Disruptions	See Section 4.1.5
• Pedestal and ELM Control	2.2, 5.2.4.2, 5.3.3
– Nonaxisymmetric coils	2.2.2
– Axisymmetric control (QH-mode)	2.2.3
– Decoupling edge pressure and current density	2.2.4
– Pellet ELM pacing	2.2.5
– Small ELM regimes	2.2.6

The scientific underpinnings of these issues can be addressed on present day devices. DIII-D is particularly well positioned to undertake such efforts due to its comprehensive, high quality, diagnostic set. Our approach is to obtain high quality data, and use it to validate theory-based models. The models can then be applied with some confidence to extrapolate the results to ITER, FDF, DEMO, and other future devices.

Many of the issues listed in the table for plasma-wall interactions are specific to carbon. DIII-D is taking on the role as the pre-eminent carbon-walled tokamak in the world, and we consider it part of our mission to determine the suitability of carbon as a first wall. As such, the five-year plan includes expansion of DIII-D's already considerable capabilities in this area.

Issues regarding transport of energy and particles between the boundary and the wall are generic to any wall material. The DIII-D program plan continues a history of significant achievement in measuring and characterizing transport of particles and energy in this region, with model validation as an important emphasis.

There are classes of instabilities that have the potential for damage to the first wall. Foremost among these are the edge localized modes (ELM). The allowed ELM energy load in ITER being adjusted downward to a level well below that scaled from current experiments. One might anticipate even more stringent requirements in a subsequent device such as FDF or DEMO. DIII-D has taken a leading role in the development of operating scenarios and techniques to control or eliminate ELMs with minimal impact on fusion performance. Five different techniques, listed in Table 4-7, have been identified along these lines, and will be studied in DIII-D. At this time, the leading technique is resonant magnetic perturbations (RMP), utilizing a set of nonaxisymmetric coils. A large, new, set of internal coils is planned to allow further optimization of the RMP technique. Other techniques, including the ELM-free QH-mode regime discovered in DIII-D, will also be explored with new tools called for in the five-year plan.

4.1.8. RF Antennas, Launching Structures and Other Internal Components

Establish the necessary understanding of plasma interactions, neutron loading and materials to allow design of rf antennas and launchers, control coils, final optics and any other diagnostic equipment that can survive and function within the plasma vessel (Table 4-8).

Table 4-8
The DIII-D Program Plan Includes Major Efforts Aimed Toward Understanding and Control of Interactions Between the Plasma and the First Wall

Topic	Five-Year Plan Section(s)
RF antennas	
• High power, long-pulse antenna	10.2.4, 10.6
Other internal components	
• Burning plasma diagnostic development	8.8

Although DIII-D cannot reproduce many aspects of the reactor environment, especially the nuclear aspects, some issues can be addressed. The high power, long-pulse rf antennas discussed in chapter 10 are mainly targeted at operation scenarios being developed in DIII-D. However, much of what is learned in such an exercise will contribute to future antenna designs for similar operating scenarios in a nuclear environment.

Similarly, some of the issues relevant to development of burning plasma diagnostics can be addressed in DIII-D, as described in section 8.8. This includes development of burning plasma compatible measurement techniques, as well as materials tests, e.g. testing mirrors in a high heat and particle flux environment.

4.2. DIII-D SUPPORT OF FDF

In order to address most of the issues in Table 4-1, we envision a future Fusion Development Facility (FDF), consisting of a tokamak with capabilities complementing those of ITER. FDF would be capable of operating at a 30% duty cycle, with continuous steady-state operation for up to 2 weeks with $P_{fusion} \approx 100–250$ MW. The main mission of FDF is to develop the nuclear technology needed for a

DEMO while demonstrating Advanced Scenario operation using a physics basis being developed on DIII-D.

The technology mission of FDF would include

- High degree of maintainability
- High neutron fluence operation (3–6 MW-yr/m^2)
- Tritium self-sufficiency demonstration
- Fusion blanket development, including
  - Electricity production
  - Tritium production
  - Hydrogen production

Along with ITER and IFMIF, we anticipate an FDF would provide a basis for a magnetic fusion based DEMO power plant.

The development of steady-state operating scenarios is not part of FDF's mission. Instead, it relies on the scientific basis being provided by ongoing research in DIII-D. FDF will require a physics basis consisting largely of the following issues:

- Steady-state scenario development
  - Required stability values (β_N = 4 at DIII-D aspect ratio) can be available in steady-state in 2–3 years
  - Confinement quality required already obtained in long pulse DIII-D plasmas
  - Bootstrap fractions already achieved
  - Pumped, high triangularity plasma shape
- Stability control
  - RWM feedback already developed
  - NTMs already stabilized
  - ELM suppression/control demonstrated in QH-mode operation and with stochastic edge field
- Uses DIII-D plasma control system, most advanced in the world
- Power exhaust more challenging than DIII-D and comparable to ITER
- Main challenge is tritium retention in plasma facing surfaces

In order to move forward to an FDF, demonstration of solutions to these issues alone is not sufficient. A viable operating scenario for FDF will need to demonstrate integration of all of these issues, and will require the scientific understanding to allow accurate extrapolation based on validated, theory-based models. This is a major goal for the DIII-D program.

References for Section 4

[Greenwald 2007] M. Greenwald, et al., "Priorities, Gaps and Opportunities: Towards a Long-Range Strategic Plan for Magnetic Fusion Energy," A Report to the Fusion Energy Sciences Advisory Committee, October 2007, http://www.ofes.fusion.doe.gov/FESAC Planning Report.pdf

[Najmabadi 2006] F. Najmabadi, et al., Fusion Engin. Design **80** (2006) 3.

5. FUSION ENERGY SCIENCE IN DIII-D

The most enduring contribution of the DIII-D research program will be the scientific knowledge gained. This knowledge will provide new insights into complicated physical phenomena of matter in the high temperature state and form the basis for the design and successful operation of next-generation fusion devices. DIII-D is well positioned to contribute to this knowledge base with a flexible set of control tools, an extensive operating space, and a comprehensive diagnostic set capable of providing both profile and highly localized information on plasma phenomena. DIII-D's strong connections to the theoretical and experimental communities both in the U.S. and internationally will enable the program to adapt quickly to the latest developments in fusion research worldwide.

5.1. INTRODUCTION

The research planned for 2009–2013 covers a broad range of fusion science topical areas (transport, stability, energetic particles, heating and current drive, and boundary) with particular emphasis placed on addressing those issues that have critical impact on the prospects of fusion energy and in those areas which DIII-D has unique capabilities or high leverage. The research elements of this plan are chosen to be consistent with three basic goals of the Fusion Science program on DIII-D:

- Discover and characterize new physics phenomena.
- Seek out transformational breakthroughs.
- Test and validate theoretical models of important physical processes.

Key research themes in the topical science areas are given in Table 5-1. Exciting advances are expected over the next five years. In the transport area, the new capabilities of DIII-D will allow us to isolate the effects of one type of turbulence (*e.g.*, trapped electron mode) while suppressing the effects of other modes (*e.g.*, ion/electron temperature gradient modes), providing a unique opportunity to test turbulence simulation models in unprecedented detail. Stability physics is moving beyond validation of the linear theory towards tackling the nonlinear picture. This is a crucial step in predicting the effects of sawteeth, NTMs, ELMs, and error fields on ITER. Also critical to ITER's success is energetic particle research, which will rigorously test theoretical models and simulation codes and develop the means to control their effect. In the heating and current drive area, the emphasis will focus on validating models of NBCD and the bootstrap current, which will ultimately play an important supportive role in our AT program. In the area of boundary physics, research will focus on to understanding the delicate balance between transport, sinks, and sources in the SOL/edge region which affects the divertor function and determines the plasma parameters at the separatrix.

It is anticipated that the research themes in Table 5-1 will change as the research program unfolds over the next few years as a result of new experimental discoveries and theoretical developments. In addition, we anticipate that by the end of this five-year period, a new frontier in fusion science will be emerging in which detailed experimental measurements are used to validate the predictions of sophisticated simulation codes that embody state-of-the-art theoretical descriptions of basic plasma phenomena. Once validated, these simulation codes will serve as a key resource in utilizing the knowledge gained from the fusion science research program.

Table 5-1
Fusion Science Research Objectives for 2009–2013

Topical Science Area	Research Theme	Research Objectives
Transport	Turbulence Characterization	• Improve characterization of turbulence and zonal flows • Quantify importance of nonlinear dynamics • Validate nonlinear turbulence simulations
	Momentum Transport	• Identify mechanism for intrinsic rotation • Assess role of turbulence in momentum transport • Determine sources of momentum sinks
	Electron Energy Transport	• Detailed comparisons of fluctuation measurements with gryo-kinetic simulations • Assess mechanisms for high electron transport rates
	Particle Transport	• Measure main ion transport characteristics • Identify mechanism responsible for density peaking • Determine role of various micro-instabilities
	Transport Barrier Physics	• Assess relative role of ExB shear, magnetic shear, and Shafranov shift in internal barrier formation • Identify L-H transition trigger mechanism
Stability	Neoclassical Tearing Modes	• Evaluate ECCD modulation for improved suppression • Assess mechanisms determining seeding threshold
	Resistive Wall Modes	• Determine mechanisms for stabilization at low rotation • Demonstrate feedback stabilization at high β and low rotation
	ELMs	• Assess tools for ELM suppression • Develop improved understanding of ELM crash
	Error Fields	• Improve understanding of plasma response to error field
	Sawteeth	• Examine role of shape and fast ions in sawtooth crash dynamics
	Extended MHD	• Improve understanding of nonlinear coupling of MHD • Assess role of rotation in MHD stability
Energetic Particles	Model Validation	• Characterize linear wave properties • Evaluate role of nonlinearity in mode saturation
	Transport	• Assess fast-ion transport (both classical and anomalous)
Heating and Current Drive	ECH/ECCD	• Evaluate far off-axis ECCD and quasi-linear effects
	FWCD/ICRF Heating	• Test methods to minimize FW absorption on beam ions • Identify mechanisms causing parasitic damping in edge
	Neutral Beam	• Evaluate dependence of NBCD on classical and anomalous effects
	Bootstrap	• Validate theory over range of plasma conditions
Boundary	SOL Transport	• Determine mechanism(s) responsible for radial transport and poloidal flows • Evaluate correlation of SOL transport with divertor heat flux footprint
	Hydrogenic Retention	• Test methods to minimize deuterium uptake in walls • Evaluate techniques for in situ removal of co-deposited deuterium
	Plasma Surface Interactions	• Document key PMI processes for erosion and redeposition in divertor and midplane region • Identify and characterize sources of dust

The ability to control and diagnose plasma properties is a key enabling feature of DIII-D. The plasma control system (PCS) on DIII-D has long been able to dynamically control global parameters such as the plasma shape, density, and β. Newer capabilities allow the PCS to dynamically control local values of the current density, toroidal rotation, and electron/ion temperatures. These control capabilities allow scientists to isolate the effects of plasma parameters, thereby enabling the elucidation of the important physical processes. Diagnostic innovation will continue to be a high priority for the DIII-D program. Such innovation supports our mission to discover and characterize new plasma phenomena. New measurements also lead naturally to new physics ideas, some of which will become transformational breakthroughs. State-of-the-art measurements of plasma profiles are needed to continuously advance our fundamental science understanding, mainly by testing the best-available theoretical model.

The fusion science research plan will be presented here according to topical science area. Note that the research described here has significant synergy with similar research topics in other programmatic areas. For example, NTMs, ELMs, and pedestal physics are part of the ITER research area (Chapter 2), while transport and noninductive currents are important to the steady-state high performance area (Chapter 3); these topics are also studied in the fusion science area. Since experiment/theory comparisons are an important component of advancing fusion science, a close working relationship will be sought between this area and the Integrated Modeling area (Chapter 6). Many of the diagnostic systems discussed in Chapter 8 are integral to the fusion science mission. Finally, the Fusion Science area is the primary point of engagement for universities and students.

5.2. Transport and Turbulence Physics

The long-term goal of fusion energy science is a predictive understanding of magnetized, high temperature plasmas embodied in a suite of well-tested modeling codes. The transport effects are particularly difficult to calculate accurately because they are the result, in most cases, of turbulent processes. To date, even under the best conditions, only the ion thermal transport has been reduced to the turbulence-free, purely collisional, neoclassical value. The difficulty of completely characterizing turbulent processes is illustrated by the example of fluid turbulence, which remains an active area of research in fluid dynamics 150 years after the fundamental, Navier-Stokes equations were first formulated. In spite of the difficulty, advances in computational capability over the last decade now make it feasible to plan an attack on the transport problem in magnetized plasmas which can lead, over the next five to ten years, to the ability to accurately calculate turbulent transport effects. Optimism in this area is based on the substantial theoretical progress that has been made in the last five years and the expectation of a continuing rapid advance in computer capabilities.

The transport and turbulence research discussed in this five year plan is well aligned with several of the topical scientific questions identified by the FESAC Priorities Panel [FESAC 2005]:

T3. How can external control and plasma self-organization be used to improve fusion performance?

T4. How does turbulence cause heat, particles and momentum to escape from the plasma?

T5. How are electromagnetic field and mass flows generated in the plasma?

An increased emphasis on transport in the DIII-D and the U.S. program is quite appropriate now that the ITER program is going forward. The development of an integrated set of modeling codes that are well validated by experiment over the next decade will be extremely valuable in the development of experimental plans for ITER and in maximizing the scientific benefit garnered from participation in ITER.

The work proposed for the next five-year period is best understood by considering the transport landscape shown in Fig. 5-1. As illustrated, transport in the plasma core is believed to be dominated by three turbulence mechanisms. These are the ion temperature gradient (ITG) mode, the trapped electron mode (TEM), and the electron temperature gradient (ETG) mode. As indicated in the figure, these turbulence modes span a substantial range of spatial scales and contribute different amounts to the various transport channels. The ion thermal transport is fairly well understood and is primarily driven by ITG modes. The electron thermal transport, the particle transport, and the momentum transport are less well understood.

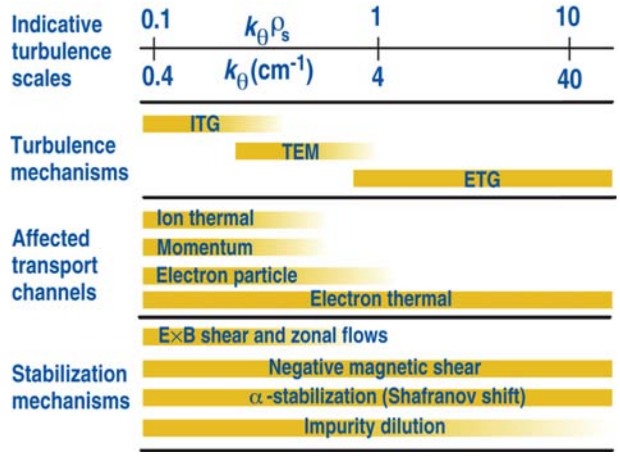

Fig. 5-1. An outline summary of drift wave turbulence scales, with corresponding turbulence mechanisms, affected transport channels, and stabilization mechanisms.

Typically, these turbulence mechanisms lead to radial transport rates that are much larger than expected from collisional diffusion. However, the various modes are stabilized by a variety of mechanisms as shown in Fig. 5-1. In some cases, transport can be reduced significantly, leading to the formation of transport barriers. The DIII-D program has been responsible for the pioneering work on stabilization of microturbulence by sheared ExB flow. A comprehensive review of this work has recently been published [Burrell 2005a]. This mechanism is responsible for the reduction of transport across the entire cross-section in our high performance plasmas [Lazarus 1996, Lazarus 1997, Greenfield 1997] and plays a dominant role in the formation of external (H-mode) transport barriers and internal transport barriers. The ExB shear effects include both the quasi-steady electric fields associated with the plasma equilibrium and the oscillating, zonal flows that are generated by the plasma turbulence itself [Diamond 2005]. Although ExB shear is a very effective stabilization mechanism, it is limited to the longer wavelength turbulence, which explains why internal transport barriers are often observed in the ion thermal channel. Shafranov shift stabilization and negative magnetic shear ($\hat{S}$) are stabilization mechanisms that often work together, as shown in Fig. 5-2. Both large positive magnetic shear and negative magnetic shear are stabilizing to microturbulence [Kinsey 2006]. But as the pressure gradient is increased, the negative

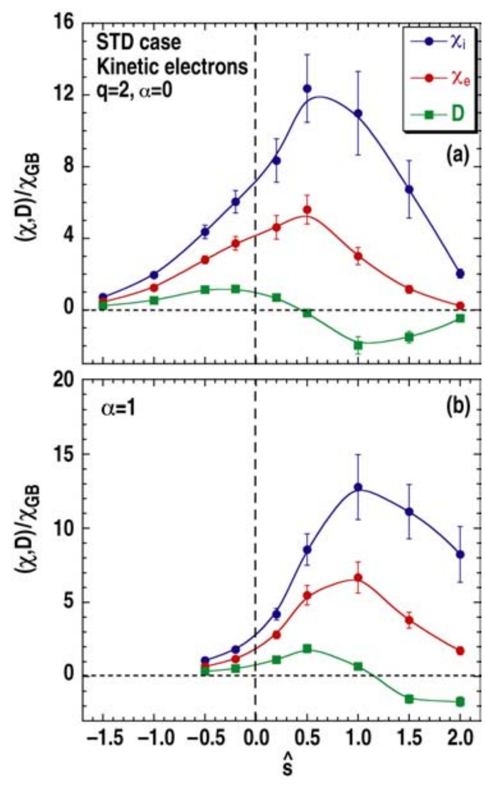

Fig. 5-2. GYRO calculation of time-averaged ion energy (blue circles), electron energy (red circles) and ion particle (green squares) diffusivities as a function of magnetic shear $\hat{S}$ for two different values of $\alpha = -q^2 R_0\, d\beta/dr$. Note that transport decreases as $\hat{S}$ drops below about 1/2 and decreases as alpha increases.

shear is more stabilizing and stabilization in the positive shear region is weakened. This mechanism is responsible for the internal transport barriers in all transport channels (ion thermal, momentum, particle, electron thermal) observed in DIII-D strong NCS discharges [Stallard 1999]. Both large q_{min} and large negative shear are favorable. The advantages of this stabilization mechanism are that it is effective for all turbulent scale lengths and projects favorably to higher magnetic field burning plasma regimes.

The overall level of transport in fusion plasmas is set by the interaction between the turbulence drives and the various stabilization mechanisms which ultimately leads to a nonlinearly saturated, turbulent state. Understanding those interactions (at all relevant scale lengths), testing the transport models, and simulations of those interactions and understanding this nonlinearly saturated state is the long term goal of transport research.

The challenge of predicting transport is sufficiently complex that no one machine or research group can tackle the whole problem by itself. Now that simulations and transport models have matured, intensive interactions of DIII-D experimentalists with the international community of theorist and modelers are needed on a continuing basis. In addition, the ensemble of tokamaks in the U.S. and internationally together can provide a wider range of tests than any one device can, leading to more definitive tests. For example, experiments testing the effects of aspect ratio are possible by combining data from DIII-D and NSTX. Experiments testing the dependence on relative gyroradius (ρ_*) can cover a significantly broader range when data from C-Mod, DIII-D and JET are combined.

New diagnostics and diagnostic upgrades are also a priority in the 2009–2013 period. As is shown in Table 5-2, some of the diagnostic improvements are upgrades of existing systems. Adding more spatial views for the BES system and the fast ion diagnostic system and expanding the capability of the various microwave diagnostic system is straightforward. For others, the basic concepts exist, but the development required is much more extensive. These include ion temperature and toroidal velocity fluctuation measurements; main ion (deuteron) measurements of density, rotation and ion temperature using charge exchange spectroscopy; and local measurements of turbulence-driven particle flux using BES. As also indicated in Table 5-2, there remain transport measurements that would be extremely fruitful if we could figure out some way to do them. These include local measurements of turbulence-driven angular momentum flux and heat flux. If we had these, they would greatly facilitate tests of the simulation and modeling codes.

5.2.1. Fundamental Turbulence

Motivation and Status: Plasma turbulence and the associated turbulent-driven cross-field transport of energy, particles, and momentum remains one of the central scientific challenges confronting the fusion energy sciences. This turbulence, believed to be dominated by temperature- and density-gradient-driven drift wave turbulence, causes transport that is typically orders of magnitude above the irreducible neoclassical minimum. Because turbulence can be reduced or even quenched by equilibrium and time-varying radially sheared ExB flows, it interacts with and affects pressure and rotation profiles in a complex, highly nonlinear process. A comprehensive understanding of plasma turbulence is a crucial component of plasma transport, and a long-term goal of this research is the development of a predictive capability for future burning plasma experiments.

Table 5-2
Diagnostic Improvements

Scientific Objective	Physics Measurement	Diagnostic Technique
Near Term Upgrades:		
Understanding role of turbulence in transport	Density fluctuations and turbulent eddy velocities for zonal flow studies	Increase BES from 32 to 64 spatial points
	Electron temperature fluctuations	Broader spatial coverage and increased sensitivity for correlation ECE system
	Turbulence velocity	Broader spatial coverage for Doppler reflectometry
	High-k turbulence	Continued development of scattering and PCI
Improved profile measurements for more stringent tests of transport models	Internal magnetic field structure	Increased spatial coverage and sensitivity for B-Stark system
	Full radius plasma rotation measurement for study of rotation structure	CER measurements at locations with $R<R_{axis}$
	Fast ion profile	Increased spatial coverage for fast ion D-alpha system
	Edge current density profile for stability, L-to-H transition, and pedestal studies	Increased sensitivity for lithium beam polarimetry system and development of edge MSE analysis to separate E_{rad} and j(r)
	Edge electron temperature and density profile for L-to-H transition and pedestal studies	Improved spatial resolution for edge Thomson scattering
New measurements using extensions of present techniques:		
Understanding role of turbulence in transport	Ion temperature and toroidal velocity fluctuations for ITG studies	High frequency CER
	Local measurement of turbulence-driven particle flux	Advanced BES analysis; ECE and microwave imaging
	Internal magnetic fluctuation measurements to determine if magnetic fluctuations drive transport	Polarimetry, crossed polarization scattering
Improved profile measurements for more stringent tests of transport models	Main ion (deuteron) density and rotation measurements for improved understanding of rotation	CER measurements of D_α spectrum
	Edge T_e measurements with <= 0.1 ms time resolution for L-to-H transition measurements	High frequency Thomson scattering; X-ray spectroscopy
New measurements which require new techniques:		
Understanding role of turbulence in transport	Local measurements of turbulence-driven heat and angular momentum flux	N/A
	2D neutral deuterium density measurements	N/A

In the previous five years, the DIII-D program developed the tools needed for intensive, definitive comparison of transport theory and experiment. We have developed and/or significantly upgraded a whole suite of turbulence diagnostics. These include a dramatic improvement in the signal to noise in the beam emissions spectroscopy (BES) measurements [Fig. 5-3(a)] [McKee 2006a], allowing core measurements of density [Shafer 2006] and low-frequency velocity fluctuations [Schlossberg 2006] even in reduced-turbulence H-mode; development of high-k turbulence diagnostics for investigation of fluctuation scales associated with electron temperature gradient modes [Fig. 5-3(b)] [Rhodes 2006]; and the very recent advances in measurements of electron temperature fluctuations [Schmitz 2007]. These have produced important new physics results on zonal flows [Gupta 2006, McKee 2006b, Holland 2007], leading to the elucidation of their role in core transport barrier formation [Austin 2006, Waltz 2006]; demonstration that high-k turbulence exists independent of the low-k turbulence [Rhodes 2007a] and demonstration of a clear reduction in electron temperature fluctuations in the H-mode plasmas [Schmitz 2007]. In addition, we have equipped the machine with new transport control tools such as simultaneous co plus counter neutral beam injection and long pulse electron cyclotron heating systems [Wade 2006]. The new NBI capability allowed control of the core ExB shear so that its effect on transport can be studied [Politzer 2006] and facilitated angular momentum transport studies by separating torque and power input [Solomon 2007]. Finally, the advent of a new computer cluster greatly improved the throughput of simulations with gyrokinetic codes such as GYRO; to run these is the sine qua non of theory-experiment comparison.

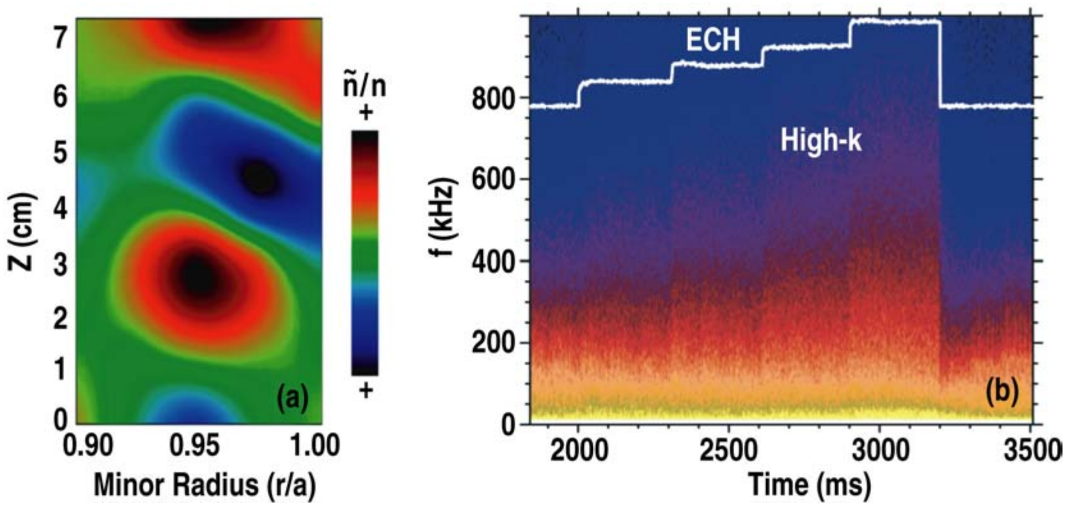

Fig. 5-3. (a) Visualization of edge turbulence by the BES diagnostic. (b) Increase in high-k turbulence during ECH measured by microwave backscattering.

Research Plans for 2009–2013: The DIII-D research program has made significant advancements in understanding turbulence and is poised to make substantial progress towards increasing the understanding turbulent processes across the wide variety of accessible plasma regimes. The primary objectives of the proposed DIII-D turbulence research program are to:

- Continue to develop advanced fluctuation diagnostic systems.
- Fully characterize plasma turbulence across plasma conditions, fields, and wavenumbers.

- Identify the underlying modes responsible for turbulence in various regimes, *e.g.*, ITG, trapped ion, trapped electron, and ETG modes.
- More fully identify and characterize zonal flows (including geodesic acoustic modes).
- Measure and quantify the nonlinear dynamics of turbulence.
- Identify the existence and importance of turbulence spreading.
- Validate nonlinear turbulence simulations for the core and edge regions.
- Possibly learn to control turbulence and resulting transport through advanced tools and increased understanding.

These studies will be aided by the extensive profile diagnostic capabilities on DIII-D, the numerous control tools, the flexibility of different operating regimes and scenarios, and a strong research focus on basic plasma physics studies.

The major emphasis in the 2009–2013 period will be the validation of sophisticated nonlinear 3D simulations, such as GYRO, GTC, and GEM and associated transport models (GLF23, TGLF). The nonlinear simulations have advanced to the point where all of the relevant physics is believed to be included, and simulations of realistic plasmas with actual geometry and measured profiles can be performed. The goal is to quantitatively compare fluctuation characteristics between simulations and measurements. These characteristics will first include fluctuation spectra, amplitudes, radial and poloidal correlation length, and wavenumber spectra across much of the minor radius of the plasma, and to do these for multiple fluctuating fields (*e.g.*, density, temperature and velocity).

In order to perform these comparisons, synthetic diagnostics that model the performance of actual diagnostics is required. These synthetic diagnostics relate the output of simulations, which typically have high spatial resolution, and no added noise sources, to measurements via application of transfer functions, thus simulating actual measurements, including finite spatial and time resolution. In addition to these first order spectral characteristics it will be very important to compare higher-order nonlinear characteristics of the turbulence, such as bispectral properties, internal energy transfer, and zonal flow generation. These parameters are typically more difficult to measure due to their increased sensitivity to noise in the measurements and reliance on multipoint measurements, but will provide a more credible comparison with simulations since they more directly probe the nonlinear physics inherent to turbulence in a magnetically confined plasma. While some work has already been done in this area for the BES system, this is an area of emphasis in the next five year plan. This is discussed further in Section 6.2.2.1.

Turbulence characterization and simulation comparison will rely on systematic dedicated experiments to measure the important quantities and vary critical parameters. Single-point comparisons under a single condition have typically not been especially useful for understanding turbulence properties. It is often helpful to vary a key parameter so that both absolute characteristics as well as trends can be compared. Generally, relatively simple plasmas (low to moderate β, standard shapes, quasi-stationary), are especially fruitful for turbulence studies. To perform well-characterized experiments and obtain useful measurements, it is typically useful to vary only one parameter at a time. Dimensionless scaling experiments provide a particularly successful method for performing experimental scans in a systematic fashion that will be useful for simulation comparison. Transport relevant dimensionless parameters include ρ_* ($=\rho_I/a$), T_e/T_I, q_{95}, ν_* (collisionality), M (Mach number), β, κ (elongation), δ (triangularity),

$\eta_{I,e}$ (ratio of density to temperature scale lengths). To perform these experimental scans, one dimensionless parameter will be varied while other dimensionless quantities are held constant, requiring careful control over density, temperature, and other profiles. These dimensionless scans are particularly useful for studying specific aspects of turbulence physics. For example, varying the Mach number changes rotation and therefore the radial electric field, and thus radially sheared ExB flow, and understanding the relation between shear flow and turbulence levels is extremely important; varying ρ_* has demonstrated that radial correlation lengths and other turbulence parameters scale in a manner consistent with the underlying gyrokinetic equations, important to understanding extrapolation of turbulent transport to large devices; T_e/T_I is predicted to affect the drive and stabilization of ITG modes; $\eta_{I,e}$ is expected to change the relative drive of ITG and TEM modes, and ν_* may affect the damping of zonal flows.

Diagnostic and Hardware Requirements: Fluctuations arising from plasma turbulence cover a wide range in parameter space and adequately diagnosing them has required the development and deployment of several fluctuation diagnostics at DIII-D. Several fluctuating fields are currently measured over a wide range of frequencies and spatial locations, including density, electron temperature, poloidal velocity and potential (in the edge). The wavenumbers covered by the various diagnostics ranges from long-wavelength range ($k_\perp\rho_i < 1$), thought to dominate much of turbulent transport, through the electron wavenumber range ($k_\perp\rho_i < k < k_\perp\rho_e$), thought to contribute most directly to electron thermal transport. The currently available fluctuation diagnostics on DIII-D are listed in Table 5-3 (see Diagnostics Section 8 for further details). Furthermore, new diagnostics are planned or under development to measure ion temperature, toroidal velocity, and magnetic field fluctuations. Arguably, the fluctuation diagnostic set on DIII-D is the most comprehensive in the world fusion program, and continued new developments and expansions will further enhance capabilities and performance.

Table 5-3
Currently Available Fluctuation Diagnostics and their Basic Measurement Capabilities

Diagnostic	Measures	Wavenumber Range
BES	Multipoint (2D) density fluctuations, v_θ	Low-k
CO_2 interferometer	Line-integrated n_e fluctuations	Low-k
Correlation ECE	T_e fluctuations	Low-k
Correlation reflectometer	Radial correlations lengths at multiple radii	
Doppler reflectometer	Density fluctuations, v_θ	Low-k to medium-k
FIR scattering	Density fluctuations	Low-k to high-K
PCI	Density fluctuations	High-k
Reciprocating Langmuir probes	n_e T_e, potential fluctuations in edge/SOL	

In addition to the development of new diagnostic systems, it is also necessary to develop advanced analysis techniques to more fully exploit the array of multi-dimensional, multi-field, and multi-wavenumber measurements. Such techniques will be necessary to unravel higher order nonlinear dynamics of the turbulence and will include such methods as bispectral analysis to examine three-wave and/or three-frequency phase coherence, time-delay-estimation for 1D time-varying flows, velocimetry for 2D flows, and biorthogonal decomposition analysis to discern growth rates and nonlinear coupling coefficients between different fluctuating fields. We note that most of the fluctuation diagnostic systems under development and deployed at DIII-D have been developed through successful collaborations with a number of universities. As such, this program element provides a fertile research field for graduate students and postdoctoral researchers entering fusion research.

Recently implemented control systems on DIII-D, described in Table 5-4, significantly enhance the experimental capabilities for studying turbulence and its dependence on plasma parameters. These tools allow for well-characterized systematic experiments to examine turbulence as a function of the key parameters that affect turbulence and turbulent transport.

Table 5-4
Control Tools for Turbulence Studies

Plasma Quantity to be Controlled	Tool to be Used
Magnetic shear	ECCD, FWCD (co or counter)
Toroidal rotation/flow shear	NBI (co or counter injection)
Electron-to-ion temperature ratio	ECH, FW
Nonaxisymmetric (stochastic) fields	Internal and external coils

This world-class combination of advanced plasma control tools, a comprehensive set of fluctuation diagnostics, advanced nonlinear simulations, and systematic experiments will greatly aid a dramatic increase in the understanding of turbulent-driven plasma transport, aid the development of a predictive capability for next-generation burning plasma experiments, and possibly lead to development of control tools for turbulent transport.

5.2.2. Momentum Transport

Motivation and Status: Rotation is widely acknowledged as playing a central role in the performance of fusion plasmas. It has been shown to be beneficial in the stabilization of deleterious MHD such as resistive wall modes and neoclassical tearing modes, and can improve the overall confinement of the plasma through increased ExB shearing. Despite the importance of rotation to fusion, momentum transport remains arguably the least well understood of the transport channels at present. Consequently, the ability to make predictions of rotation in ITER and other next step devices is questionable, and it is difficult to make an accurate assessment of whether the rotation will be sufficient, for example, to stabilize resistive wall modes or whether external feedback control is required. Furthermore, the scaling rules and various other database studies developed over the last 20+ years to extrapolate performance metrics to future devices come almost exclusively from plasmas with levels of rotation much greater than currently expected for ITER. Hence, obtaining predictive understanding of rotation and momentum transport is of significant importance.

Recent enhancements to the auxiliary heating systems on DIII-D allow us to do low torque (or even torque-free) heating experiments, as seen in Table 5-5, making it possible to study momentum transport more systematically. In particular, following the reversal of one of the beamlines, it has become possible to investigate momentum transport at significant stored energy but low net injected torque using balanced NBI [Luce 2006]. Experiments have typically revealed a nonlinear response of the angular momentum to the applied torque, indicating a torque dependence of the momentum confinement. In some cases, this leads to an improvement in the momentum confinement at low torque (Fig. 5-4), although in other plasmas (*e.g.*, hybrid scenario), the opposite effect is observed [Solomon 2007]. It is speculated that differing levels of ExB shear may account for the observed difference, but it is clear that we have a long way to go to understanding momentum transport when even simple concepts like how the rotation scales with applied torque is poorly understood.

Table 5-5
Sources and Sinks of Momentum on DIII-D

Auxiliary system	Primarily Heats	Torque injection
ECH	Electrons	Essentially zero
FW	Electrons	Essentially zero
NBI	Ions	Variable (co/balanced/counter)
Coils (internal or external)	N/A	Resonant/nonresonant breaking

Neoclassical predictions for rotation in tokamaks have to date not been very successful in either the toroidal or poloidal directions. Experimentally, the local toroidal momentum diffusivity χ_ϕ is found to be several orders of magnitude greater than the neoclassical prediction. Theory has suggested how the momentum diffusivity can be tied to the ion thermal diffusivity χ_i due to micro-turbulence, but even in cases where the low-k turbulence is seemingly suppressed and the ion thermal transport behaves neoclassically, the momentum transport remains anomalous. Moreover, even the long-standing observation that $\chi_\phi \sim \chi_i$ is now being questioned, with recent results on DIII-D showing that this relationship becomes less valid as the rotation is reduced [Solomon 2007]. Other studies on JT-60U [Yoshida 2007] and JET [Tala 2007] also suggest that the transport of momentum and energy may not be so closely coupled. In the poloidal direction, neoclassical theory has been shown to be inadequate to describe the poloidal rotation, either in magnitude or even direction. This has serious consequences in terms of the ability to go from typical measurements of the impurity (carbon) rotation to the main ion (deuterium) rotation, which is usually done through radial force balance, using the neoclassically calculated poloidal rotation.

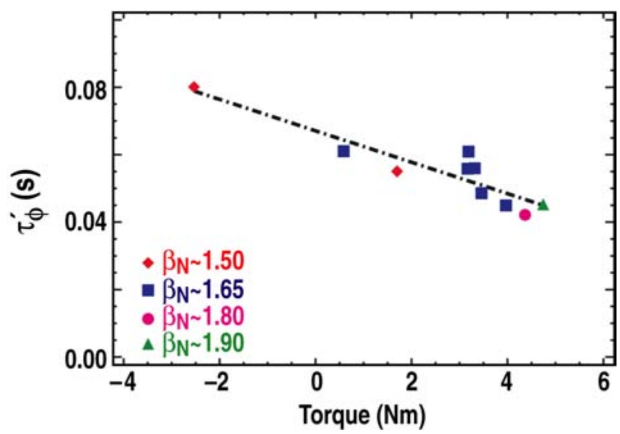

Fig. 5-4. Incremental momentum confinement time versus torque, where the intrinsic rotation has been subtracted from the actual rotation prior to τ'_ϕ being computed [Solomon 2007].

Research Plans for 2009–2013: Obtaining predictive knowledge of rotation will require advances in understanding in several key topics, as seen in Table 5-6. A relatively recent discovery in tokamaks is the existence of an “intrinsic rotation”, observed in the absence of any applied auxiliary torque (*e.g.*, [Rice

1998, deGrassie 2007]). An example of intrinsic rotation is shown in Fig. 5-5 for ECH H-mode plasmas on DIII-D. Scaling laws are being developed to predict the intrinsic rotation level, but without understanding the physics mechanism behind this intrinsic rotation, there will remain significant uncertainty in the magnitude of intrinsic rotation in ITER. The investigation of intrinsic rotation on DIII-D in the next five years will focus on identifying the source/generation of torque to the plasma, linking this with theory, and establishing how this source scales with plasma conditions. Also we must bridge the gap between zero and finite torque input. Experiments to date have indicated that the applied neutral beam torque adds to the anomalous torque associated with the intrinsic rotation, but this is not yet conclusive and needs to be answered. This can best be done on DIII-D, where neutral beam torque can be varied at constant power/beta. The obvious goal of these studies is to make some estimate of the intrinsic rotation level to be expected in ITER.

Table 5-6
Momentum Topics to be Investigated in Next Five Years on DIII-D

Topic	Key Physics	Issues
Momentum sources	Intrinsic rotation	• What is the cause? • How does it scale? • Does external torque simply add to intrinsic torque?
	Alfven eigenmode activity	• How does the redistribution of beam ions by AEs affect the torque deposition profile?
Momentum sinks	Error fields	• How much does the rotation change because of resonant/nonresonant breaking? • Can we measure the neoclassical toroidal viscosity?
Momentum transport	Main ion rotation	• Can we develop global confinement scaling relations? • Can we separate diffusive and pinch terms? • Is measured χ_ϕ in agreement with turbulence simulations?

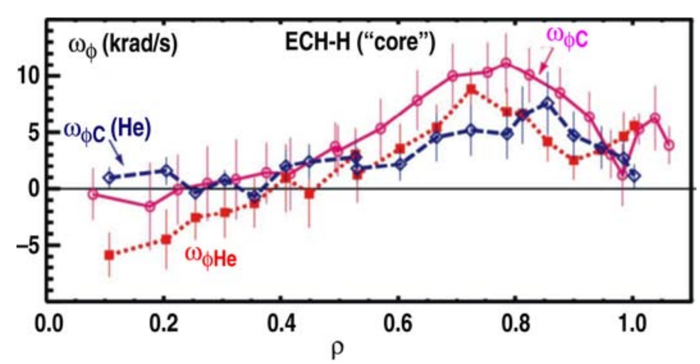

Fig. 5-5. Intrinsic rotation in ECH H-mode [deGrassie 2007].

Other sources of momentum input, such as that provided by neutral beam injection, also need to be validated in future experiments on DIII-D. A significant fraction of DIII-D plasmas are run at high plasma performance and Alfven eigenmode (AE) activity is commonly present. AEs are thought to be capable of redistributing the fast ion profiles, and with this the torque deposition profiles. Quantitative understanding of rotation depends critically on understanding our sources, and there should be strong interest and collaboration on this aspect with the energetic particles group. This issue also overlaps with the interests of the heating and current drive group, as the redistribution of beam ions can broaden/reduce the NBCD.

The flipside of this is to obtain a better understanding of momentum sinks in the plasma. It is well known that error fields can affect the rotation (*e.g.*, resonant/nonresonant braking). However, only limited success has been made in actually computing the torque associated with these error fields (the neoclassical toroidal viscosity [Cole 2007]) and quantitatively computing the expected change in rotation. This is also of importance in a broader study of stability and is relevant to ELM suppression techniques such as resonant magnetic perturbations (RMP).

Once the momentum sources and sinks are better understood, we will be ready to better characterize the momentum transport. First, global scaling relations for the momentum confinement time should be developed, similar to the work done for scaling of the energy confinement time τ_E. Second, we need to extend our momentum transport studies to consider in more detail local transport effects. Since the main (deuterium) ions have a different rotation rate than the impurity (carbon) ions, especially for low torque injection, we need to study all ion species. To better characterization of the momentum transport, we need to study the role of both diffusivity and pinch terms; momentum perturbation experiments are a key way of making such assessments. We should continue to investigate whether there is any fundamental relationship between χ_ϕ and χ_i, particularly under varying rotation conditions. Ultimately, as our description of momentum transport advances in tandem with our understanding of sources and sinks, then the existing theories need to be tested more extensively. We need to pursue comparisons with gyrokinetic codes like GYRO to see whether the momentum fluxes are consistent with experimental observations and can be characterized by comparable local transport quantities. A complete verification of the a rotation model should be the ultimate goal of the momentum transport research program, and following from that we should gain an understanding of how we can manipulate the rotation to maximize fusion performance (control rotation shear, maximize intrinsic rotation, etc).

Hardware and Diagnostic Requirements: Improvements in diagnostic capability are also needed. There is a clear need for the capability to make rotation measurements of the main ion species (deuterium). The charge exchange spectrum of deuterium is difficult to analyze due to the various aspects, including the bright cold edge light, emission from halo neutrals, Stark splitting of the charge exchange coming from the three beam energy components, *etc*. In principle, all of these can be handled with a model of the charge exchange spectrum of sufficient complexity; the simultaneous measurement of the ion temperature using a carbon line will remove one unknown quantity from the fit. In light of the difficulty/uncertainty in converting impurity carbon rotation into a deuterium toroidal rotation, the development of a main ion charge exchange measurement should be considered a high priority. Continued study of intrinsic rotation could be greatly improved by the addition of an alternate method of measuring rotation without using neutral beam injection as required for CER. One possibility worth considering would be an X-ray crystal spectrometer similar to that used on C-Mod. Finally, the turbulence fluctuation diagnostics have been shown to give useful information of the rotation of various modes. A list of the diagnostics for momentum transport studies in the 2009–2013 period on DIII-D is given in Table 5-7.

Table 5-7
Diagnostics for Momentum Transport Studies

Type	Diagnostic	Measure
Spectroscopy	CER	• Deuterium ion density and rotation • Impurity (helium, carbon, *etc.*) ion density and rotation
	X-ray crystal	High-Z impurity rotation
Fluctuations	BES	Advection of turbulence
	Doppler reflectometry	Rotation of turbulent modes (if k value is known)

5.2.3. Electron Energy Transport

Motivation and Status: Although significant progress has recently been made, the electron energy transport in tokamaks is still not as well understood as ion energy transport. In burning plasmas the electrons will be heated by the alpha particles generated by fusion reactions whereas a large fraction of the ion heating will be due to electron-ion collisional transfer. Therefore, electron energy transport will be an important factor in reactor performance. In the upcoming proposal DIII-D will expand on its current accomplishments in this area. As seen in Fig. 5-2, electron energy transport can be driven by ITG modes, TEM, and ETG modes. To elucidate understanding in this critical area, improved and unambiguous measurements across the wavenumber spectrum are needed, particularly in the intermediate to high wavenumbers, to relate turbulence to simultaneous measurements of the transport properties.

An example of recent results from DIII-D is shown in Fig. 5-6, where the responses to 1.4 MW of ECH (heating centered near r/a~0.6) are shown of spatially localized high-k ETG-scale density fluctuation measurements ($k \sim 35\ cm^{-1}$, $k\rho_s \sim 6$–8) and electron and ion energy fluxes [Rhodes 2007b]. There is a large increase in the electron energy flux for r/a>0.6 associated with the local deposition of ECH. The ion energy flux also increases across the radii during ECH owing to collisional heat exchange with the electrons. Not shown are FIR measurements of low through intermediate-k density turbulence showing an increase in ñ with ECH, which potentially contributes to the increase in energy flux with ECH. It is intriguing that the high-k ñ decreases and increases in regions where the electron energy flux shows the same response.

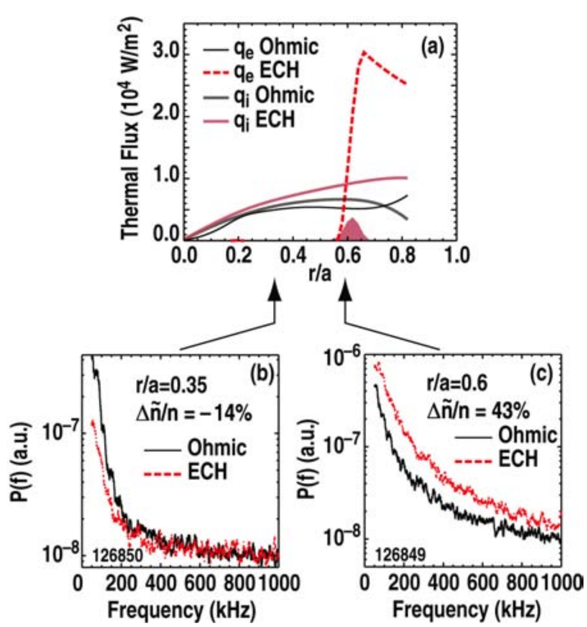

Fig. 5-6. (a) Ion and electron energy fluxes for Ohmic and Ohmic plus ECH (1.4 MW). The ECH heating location is indicated by the shaded region. (b) and (c) show the localized high-k density fluctuation levels during the Ohmic and ECH times. The percentage change at each location is indicated in the figure. The spatial resolution of these high-k measurements is ~±0.1 in r/a [Rhodes 2007b].

Some of the important questions that will be addressed regarding electron energy transport over the 2009–2013 period are:

- Is there evidence of streamers associated with transport from ETG modes?
- What is the role of magnetic shear in the electron ITB?
- For ion ITBs, why is the ion transport greatly reduced while the electron transport remains at anomalous levels?
- How significant is magnetic flutter transport, especially for high β plasmas?
- Is some electron transport caused by nonfluctuation mechanisms, such as micro-tearing modes?

Hardware and Diagnostic Requirements: The tools that will be available on DIII-D to carry out this task are listed in Table 5-8. Higher power ECH will allow larger scans of the electron temperature gradient and higher electron energy fluxes, resulting in more accurate determination of the electron transport. A more extensive use of transient transport (*i.e.*, ECH modulation) experiments will allow us to further distinguish between the effects of diffusion and thermal pinches. A valuable improvement in our study of the turbulent modes responsible for electron transport will come from spatially localized measurements of low-k (currently localized measurements obtained with BES and reflectometry), intermediate-k, and high-k (microwave backscatter) fluctuations. During this proposal period it is planned to add multiple, simultaneous channels of localized intermediate wavenumber ñ measurements (*i.e.*, TEM wavenumber range). In addition, the high-k system will be upgraded to measure poloidal wavenumbers that will complement the current radial wavenumber measurements. Finally, recent electron *temperature* fluctuation measurements (from correlation ECE, or CECE) will be expanded to multiple simultaneous channels. This will build on the current success of the CECE system and add the capability of simultaneously measuring two different turbulent fields at multiple radii. To make detailed comparisons between the fluctuation measurements and the transport simulation codes, synthetic diagnostics need to be build into the codes. The use of the current diagnostics, higher power ECH, and transport simulation codes, combined with upgraded and new diagnostics, will allow continued progress in this important area.

Table 5-8
Tools for Studying Electron Energy Transport

Tool	Purpose	Improvements
ECH	Easily vary electron energy flux and ∇T_e	Increase power to 12 MW
BES	Measure low-k density fluctuations	Additional channels
Reflectometry	Measure low-k density fluctuations	
Correlation ECE	Measure low-k T_e fluctuations	Simultaneous measurements in multiple locations
PCI	Measure high-k density fluctuations	Move chords to view center of plasma
TBD	Measure intermediate-k fluctuations	New diagnostic capable of localized measurements
Microwave backscattering	Measure high-k density fluctuations	Measure poloidal wavenumbers
Transport simulation codes (*e.g.*, GYRO, TGLF)	Simulate turbulence and electron transport	Add synthetic diagnostics

High power ECH heating, localized turbulence measurements from small-k to large-k, and the turbulence simulation and transport codes (TGLF, GYRO) constitute a powerful tool set on DIII-D for answering these important questions regarding electron energy transport.

5.2.4. Particle Transport

Motivation and Status: Particle transport will strongly affect the fusion performance of burning plasma experiments, primarily through the control of helium ash and other impurities, but also through the peaking of the density profile. Our physics understanding of particle transport is not as advanced as for heat transport; therefore, the application of additional resources to this topic should yield a rapid advance in our knowledge. Examples of previous work on particle transport on DIII-D are the dependence of helium transport on dimensionless parameters [Petty 2005a] and changes in the density peaking with T_i/T_e [Casper 2006].

The radial transport of the main (hydrogenic) ions is a relatively unexplored area of physics in fusion plasmas. The hydrogenic ions are little studied because the spectroscopic techniques that work well for the heavier impurity ions are much more difficult. Since the impurity and main ion density profiles are linked in a complicated manner, our study of particle transport would greatly benefit from the simultaneous study of all plasma species.

Research Plans for 2009–2013: Experiments over the 2009–2013 time frame on DIII-D will measure the dependence of particle transport for various plasma species (main ions, electrons, impurity ions) on:

- Dimensionless parameters (ρ_*, β, ν_*, Z_i, *etc*.)
- Microturbulence modes (ITG mode, TEM, ETG mode, *etc*.)
- Turbulence stabilization mechanisms (ExB flow shear, Shafranov shift, zonal flows, *etc*.)
- Stiffness (*i.e*., test of critical gradient models).

The research plan mainly involves core particle transport because the density of the H-mode pedestal is greatly affected by nondiffusive processes such as ELMs and nonaxisymmetric magnetic fields; however, see Section 5.6.2 for a discussion of research plans to study particle transport in the plasma boundary and SOL.

The hydrogenic ion measurements will be complemented by studies of the electron particle transport; these two species must have similar transport rates to maintain plasma quasi-neutrality. The electron transport can be measured using modulated gas puffs of deuterium or shallow pellets [Baker 2000]. In this case, the evolution of the electron density profile is determined from the density profile reflectometers, Thomson scattering, and line-integrated density interferometers. The dependence of the electron density profile (*i.e*., density peaking) as a function of dimensionless parameters such as ρ_*, ν_*, T_e/T_i, *etc*., will be determined and compared to transport simulations. As a direct result of our research program into particle transport, a fundamental understanding of density peaking in tokamaks will be acquired.

The study of impurity ion transport is more straightforward than main ion transport because of the relative ease of spectroscopic measurements. The transport of helium ions has direct implications for ash build up in burning plasma devices, while the accumulation of heavy impurities in the plasma core is a concern for vessels with all-metal plasma facing components. The helium density evolution can be measured spectroscopically following a helium gas puff [Wade 1995], allowing both the diffusive and convective components of helium ion transport to be determined. The transport of heavier impurity ions can also be determined from spectroscopic techniques, primarily by measuring the exponential decay of the impurity density following a gas puff (this requires an external particle sink such as a cryopump). To study the transport of metallic ions, a new laser "blow off" diagnostic can be installed on DIII-D. This diagnostic injects particles into the plasma using a ~1 J laser pulse to evaporate a spot off a thin film of metal deposited on a glass plate. Tungsten is a good candidate for study because it is one of the leading choices for a plasma facing material.

Testing our theoretical understanding of particle transport is an important part of the research plan. The timing is perfect on this subject because sophisticated nonlinear transport simulation codes like GYRO [Candy 2003] are just now becoming ready to predict the particle transport. Theory based models using reduced gyro-Landau-fluid equations, such as the TGLF transport model, are also available for comparison with the measured particle transport. These tests can be made more stringent by applying the principles of similarity and scale invariance in designing particle transport experiments. The first step in this regard is to test whether the principle of similarity holds for particle diffusion and convection, which can be accomplished by comparing two tokamaks of different physical size but the same values of the dimensionless parameters.

Hardware and Diagnostic Requirements: A novel proposal for this five year plan is to determine the main ion transport from the Balmer-alpha light. When the injected neutrals charge exchange with hydrogenic ions, the excited states of the neutral atoms promptly radiate, yielding a somewhat complicated spectrum that contains information about the thermal population, the energetic population, and the Stark splitting of the Doppler-shifted neutral-beam line; the spectrum also contains a large amount of emission from the cold population of neutrals in the plasma edge. With diligent effort, information about the thermal population can be obtained from this spectrum, including the main ion density and rotation [Svensson 2001]. This direct measurement of the (thermal) hydrogenic density profile can be used to determine the main ion diffusion and convection following a gas puff (or shallow pellet). While the existing chords and spectrometers viewing the neutral beam lines on DIII-D are adequate for beginning these studies, it is planned that additional chords will be added that are optimized for these main ion measurements.

A second novel proposal is to study the transport of electrons and ions with several times the thermal velocity using detailed measurements of the noninductive current density. The close association between the current drive profile and the current-carrying particle density allows a separate measurement of super-thermal particle transport. DIII-D is ideally suited for this method because of the excellent motional Stark effect (MSE) diagnostic [Rice 1997]. The transport of energetic electrons can be determined by comparing the electron cyclotron current drive profile with predictions from bounce-averaged Fokker-Planck codes. The transport of energetic deuterium ions can be studied by comparing the measured neutral beam current drive profile with Monte-Carlo calculations. For both electron and ion current drive, the source profiles can be verified experimentally, which adds to the confidence of the particle transport measurement. New diagnostics that can be added to DIII-D for this purpose include solid-state neutral-particle detectors, fusion product detectors, an improved soft x-ray array, and a hard x-ray camera.

5.2.5. Transport Barrier Physics

Transport barrier formation involves the physics of the various turbulence suppression mechanisms: ExB shear decorrelation and magnetic shear and Shafranov shift effect on the turbulence growth rate. For edge transport barriers, the physics appears to be dominated by the effect of ExB shear while for core transport barriers both the electric and magnetic shear effects play a role.

Understanding transport barriers requires consideration of two separate but related questions. First, since we have observed improved confinement states which last essentially forever when measured on turbulence time scales, what is the mechanism that allows the plasma to remain in the reduced-turbulence, improved confinement state where the gradients (which provide the free energy drive to the turbulence) are actually increased? Second, what is the trigger mechanism that starts the process allowing the plasma to bootstrap itself into the improved confinement state?

The general answer to the question of sustained operation with reduced turbulence is that the equilibrium relations between pressure, rotation, magnetic field and electric field allow solutions with significantly reduced turbulence growth rates or with turbulence fully quenched. This is most easily seen in the case of ExB shear, where increased pressure gradient and rotation in the improved confinement state lead to increased equilibrium ExB shear which maintains the turbulence suppression. Much of the essential physics here has already been captured in modeling codes like GLF23.

Investigation the trigger mechanism(s) is an area of active research. One focus of that work is on the effect of zonal flows on the turbulence. Zonal flows are stable flows that are driven by the turbulence and provide an energy sink for that turbulence [Diamond 2005]. Accordingly, they are an essential part of the overall turbulence picture. They can be responsible for predator-prey type oscillations in the plasma condition. During these cycles, background equilibrium changes in the plasma can occur which then lock the plasma into the improved state. These issues are discussed further in the next two sections.

5.2.5.1. Core Transport Barriers. Understanding the improved confinement state requires investigating the relative roles of ExB shear decorrelation, magnetic shear, and Shafranov shift in the formation of core barriers. In the past, confronting this question was difficult, since increasing the beam power increased the plasma rotation, which affects ExB shear, and also increased the plasma β, which affects the Shafranov shift. With the advent of co plus counter beam injection on DIII-D, we now have the tools to separate these effects, since we can independently vary rotation and input power. In addition, as we obtain more ECH power, we have greater capability to shape the current profile, thus altering the magnetic shear. These new tools will be used in a series of experiments to determine the relative roles of ExB shear and magnetic shear.

The question of a trigger mechanism for core transport barriers is complex. At sufficiently high power levels, we have evidence of barrier formation on relatively long time scales (10s to 100s of milliseconds), consistent with equilibrium effects only [Greenfield 1999]. On the other hand, at lower input powers, we have seen clear evidence of trigger events which happen just before q_{min} reaches low order rational values (*e.g.*, q_{min} = 2, 5/2, 3 ...) [Greenfield 1999, Austin 2006]. These observations are similar to those made on JET, for example [Joffrin 2003]. As can be seen in Fig. 5-7, these latter events are associated with localized, transient changes in the turbulent eddy velocity; this is consistent with the theoretical model [Waltz 2006] that zonal flows grow up at or slightly before the time when q_{min} reaches the low order rational values.

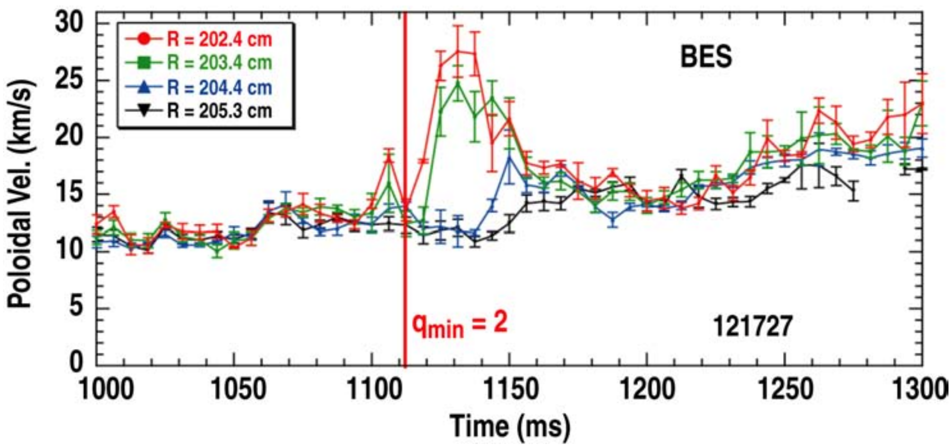

Fig. 5-7. Turbulent eddy velocity determined from the BES density fluctuation measurements using multipoint correlation analysis measured at the points indicated in the figure. Notice that the shear in the eddy velocity starts increasing slightly before the time when q_{min} =2 and increases even more after that time [Austin 2006].

The key question for the core barrier trigger associated with q_{min} is whether the zonal flow effect can explain the experimental observation that the confinement improvement starts 10 to 20 ms prior to the time when q_{min} reaches the low order rational value. Answering this question will require more detailed GYRO simulations as well as detailed experiments exploiting the turbulence velocity signature as a diagnostic of the zonal flow. As part of this work, we will use our detailed data to test other hypothesis about the core barrier formation [Joffrin 2003].

5.2.5.2. Edge Transport Barriers. The plasma parameters in the edge are an important boundary condition for the interior (core) profiles. For example, the physics of the H-mode edge pedestal is an integral part of the overall investigation of the sustained, improved confinement state. Since we are pursuing this latter topic in the context of developing and testing predictive models, the pedestal work is covered in Section 6.2.3. The associated work on pedestal stability and ELM control is in Section 2.2. This section covers the question of the trigger for the L-to-H transition and how it determines the H-mode power threshold, which is the main fusion science question to be confronted in the edge transport barrier area. The value of the H-mode power threshold is of importance to the operation of ITER since it cannot achieve its goals in a L-mode edge plasma. While the most crucial question is how much heating capability must ITER have to access H-modes in deuterium/tritium plasmas, it is also valuable to know what will be the H-mode power threshold for pure hydrogen plasmas because ITER will not introduce deuterium or tritium for the first few years of experiments.

In order to investigate the L-to-H trigger more fully in the future, we plan a concentrated investigation of how the edge turbulence varies when we change the machine control parameters that can influence the H-mode power threshold (listed in Table 5-9). There is an extensive literature on the factors that can influence the power threshold for the L-to-H transition, and recent experiments on DIII-D have added the co plus counter beam mix to the list [Luce 2006]. It is clear from the large number of factors that influence the transition that simple scalings cannot capture the true physics that is taking place. DIII-D is an ideal device to study the physics of the L-to-H trigger because of the flexibility to alter all of the parameters listed in Table 5-9 and our excellent suite of edge diagnostics.

Table 5-9
Parameters that can Control the H-mode Power Threshold

- Plasma density
- Ion ∇B drift direction relative to X-point
- Plasma cleanliness/fueling
- Hydrogenic species
- RMPs (nonaxisymmetric magnetic fields)
- Plasma shape
- I_P ramp rate
- Sawteeth trigger
- Co/counter beam mix
- Rotation

These results will be compared both to simple, analytic theories of Reynolds drive of zonal flows and, as they mature, to full gyrokinetic calculations of the plasma edge. One hypothesis that we are investigating involves the effect of turbulence generated zonal flows on the transition [Moyer 2001, Tynan 2001]. As can be seen by the bicoherence analysis in Fig. 5-8, three-wave coupling consistent with zonal flows occurs near the time of the L-to-H transition. Since the velocimetry analysis of the BES data now allows measurement of the turbulent eddy velocity in the plasma, we are in a better position to test some of these ideas.

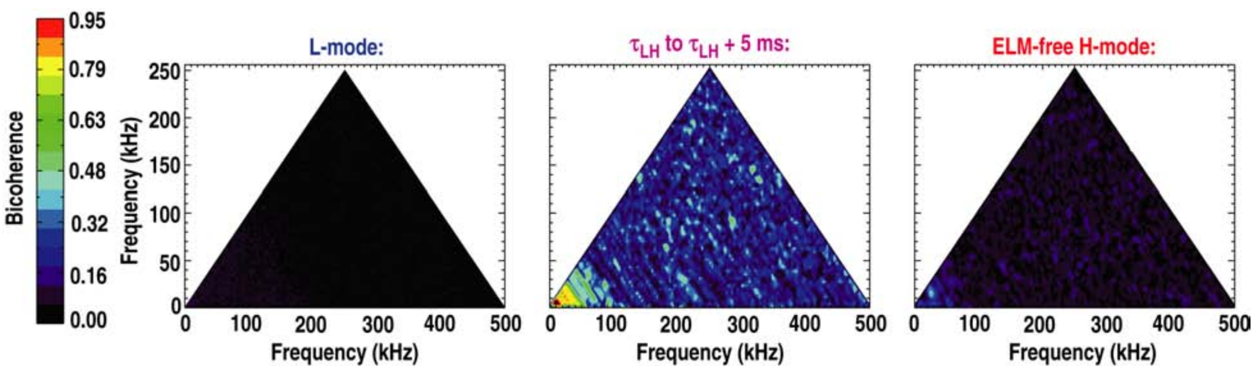

Fig. 5-8. Auto-bicoherence across the time of the L-to-H transition using density fluctuations measured by the BES system. During the L-to-H transition, there is strong coupling as lower (<100 kHz) frequencies and some residual coupling is observed in H-mode at those frequencies. There is no sign above the noise level of bicoherence in L-mode. This is consistent with zonal flows driven by Reynolds stress.

5.3. MHD STABILITY

The goal of stability research on DIII-D is to establish the scientific basis to predict and control macroscopic instabilities. This research addresses the following questions from the FESAC Priorities Report:

T2. What limits the maximum pressure than can be achieved in laboratory plasmas?

T3. How can external control and plasma self-organization be used to improve fusion performance?

The status of MHD stability research on DIII-D and a list of future challenges in given in Table 5-10. MHD stability is important to a number of research topics on DIII-D; therefore, research in this area is covered in several different sections of this five-year research proposal, as seen in Table 5-11. A central theme in stability research over the 2009–2013 period is understanding the nonlinear evolution of MHD phenomena such as sawteeth, NTMs, ELMs, and the plasma response to error fields. Strong use will be made of upgrades to the DIII-D heating and current drive capabilities in evaluating stability issues over the next five years.

Table 5-10
Status and Future Challenges of MHD Stability on DIII-D

Current Status	Future Challenges
• RWM stability observed at very low plasma rotation. • Active RWM feedback control provides dynamic error field correction and recovery from transient perturbations.	• Validate physics of torque balance and rotational stabilization, including kinetic effects. • Demonstrate feedback stabilization at low plasma rotation; develop optimal control.
• NTM stabilization by localized ECCD demonstrated, using feedback control and real-time tracking of rational surface. • Multi-machine scaling shows onset β proportional to ρ*.	• Validate EC power requirement for 2/1 stabilization: CD width, alignment, modulation. • Extend physics understanding of NTM seeding and threshold physics, including role of plasma rotation.
• Massive gas injection mitigates wall heat loading and halo currents in disruptions. • Efficient assimilation of injected gas requires rapid delivery during thermal quench.	• Validate physics of gas jet mitigation, gas delivery and mixing, and role of MHD activity. • Validate physics of runaway electron generation, transport, and suppression. • Develop means to reach Rosenbluth density for runaway avalanche suppression.
• Central plasma stability and thermal transport are closely linked in sawtooth instability. • Modification of sawtooth period by localized ECCD demonstrated in L-mode plasmas.	• Benchmark Porcelli model and detailed 3D models (NIMROD) for sawtooth stability. • Demonstrate sawtooth modification in H-mode plasmas, with reduction of NTM seeding
• ELM stability shown to be in good agreement with peeling-ballooning mode stability limits. • ELM suppression by resonant magnetic perturbation correlated with edge island overlap.	• Clarify physics of heat and particle transport during RMP ELM suppression. • Investigate role of rotation in QH-mode and small ELM regimes. • Extend parameter regimes for RMP suppression, QH-mode, and small ELM operation. • Optimize pellet size and rate for ELM pacing.
• Plasma response must be accounted for in error field correction at both high and low β. • Error field correction improved by I-coil, feedback control.	• Validate ideal MHD model of plasma response in error fields at high and low β. • Extend feedback-controlled error field correction to moderate-beta plasmas. • Investigate the need for n>1 error correction
• Nonlinear modeling (NIMROD) applied to NTM onset. • Atomic physics coupled to NIMROD for modeling of disruption mitigation	• Develop and validate more complete, realistic models, including nonlinear, 3D physics, kinetic effects, and heat transport. • Apply models to ELM suppression, sawtooth stability, MHD mode coupling, and interaction of plasma rotation with MHD stability. • Develop nonaxisymmetric equilibrium modeling
• Separate stabilization of RWM, NTM, ELM, … demonstrated in dedicated experiments	• Integrated stability control for avoidance or suppression of instabilities. • Reliable, disruption-free operation at high beta. • Reliable disruption mitigation when needed

Table 5-11
Location of MHD Stability Research in this Proposal

Section	Topic
2.2	ELM control
2.3	NTM stabilization
2.4	RWM control
2.7	Disruption characterization and mitigation
3.5	Instability avoidance and control for advanced scenarios
5.3	Physics of sawteeth, NTMs, ELMs, error fields, extended MHD
6.2.2.2	MHD stability and control for integrated modeling
6.2.3.2	ELM stability and control for integrated modeling

The various diagnostics and modeling tools needed to carry out the MHD stability research on DIII-D is described in the subsections on each individual topic.

5.3.1. Sawteeth

Motivation and Status: Sawteeth can have several deleterious effects of fusion plasma performance. Besides the effects the sawtooth crash has on the central confinement, it can also serve as a trigger for NTMs. DIII-D studies in recent years have focused on the impact of shape on the sawtooth dynamics and examining methods for modifying the stability of sawteeth [Lazarus 2007]. These experiments have shown that shape has a significant impact on the sawtooth dynamics. The central ion energy confinement is observed to vary considerably as plasma triangularity (δ) varies from 0 to 0.4. However, the most radical result of DIII-D experiments to date has been the severe degradation of electron energy confinement as δ is reduced below 0.2. These experiments also showed that the plasma does not allow a change in $\langle\kappa_n\rangle$ in the central region, where $\langle\kappa_n\rangle = (\partial/\partial\psi \langle B^2+2\mu_0 P\rangle)/B^2$ is the normal curvature. Instead, the attempt to change the magnetic well results in a change in P' so that a condition $\langle\kappa_n\rangle \approx 0$ persists. At the same time, the effects predominately appear in the electron channel while the ions maintain a pressure gradient and the interchange criterion is violated.

Sawtooth stability has been explored using both ECH and ICRF heating. Initial experiments in L-mode plasmas [Pinsker 2003] showed that the time between sawteeth was a strong function of the location of ECH relative to the q=1 surface, as seen in Fig. 5-9. The production of more rapid, smaller sawteeth is a potential way to reduce the seed islands below a destabilizing value for NTMs. A second approach is to use ICRF heating at high harmonics to accelerate the deuterium beam ions to higher energy and density inside the q=1 surface. This has a strong stabilizing effect of the sawteeth, and previous L-mode experiments on DIII-D using ~1 MW of ICRF heating succeeded in producing "monster" sawteeth [Heidbrink 1999].

Research Plans for 2009–2013: Experiments in the next five-year period will continue to explore the role of plasma shape on the sawtooth character. The strong increase in central electron transport as the triangularity is reduced will be examined in more detail by documenting the plasma turbulence and the profile evolution during the sawtooth cycle. Particular emphasis will be placed on obtaining good profile data that is suitable for time-dependent transport analysis. In addition, DIII-D's unique plasma shaping capability will be utilized to explore the electron response to higher order moments of the plasma shape.

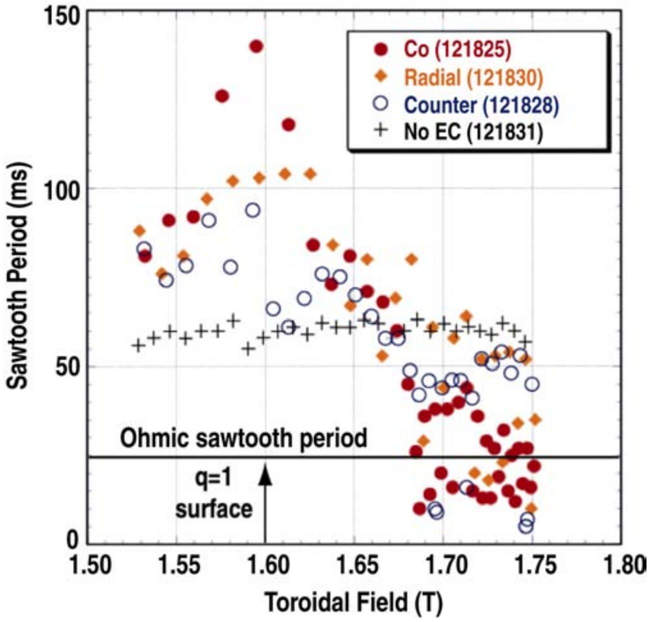

Fig. 5-9. Dependence of sawtooth period on the ECH location relative to q=1 surface for L-mode plasmas.

The sawtooth stability studies with localized ECH near the q=1 surface will be extended to high beta H-mode plasmas. Detailed studies will be carried out to develop a better understanding for the physical mechanism causing the change in sawtooth stability as the ECH deposition location is varied. In addition, the upgraded ICRF power should allow stabilization of sawteeth through the generation of a energetic particle tail. Such stabilization has the potential for use in producing H-mode plasmas with high internal inductance that maintain q<1 indefinitely.

With both ECH and ICRF heating, the Porcelli model [Porcelli 1991] will be tested utilizing DIII-D's diagnostic capability to accurately measure both the current profile as well as the thermal and fast ion pressure profiles. While this model seems to have gained wide acceptance worldwide, its ability to describe DIII-D observations is still an open issue. Further work is needed to tests this model.

5.3.2. NTM Physics (Seeding and Threshold)

Motivation and Status: Neoclassical tearing modes (NTMs) are the principal stability limit on performance in the standard sawteething ELMy H-mode [Hender 2007]. The m/n=3/2 and 2/1 islands, for example, destabilized and sustained by the helically perturbed bootstrap current, can reduce confinement and thus β. Potentially worse is the drag on plasma rotation by eddy currents induced in the resistive vacuum vessel wall, which can bring the plasma rotation to a stop with concomitant loss of H-mode and possibly lead to disruption [La Haye 2006a]. NTM stability depends on a combination of the classical tearing stability parameter Δ', assumed negative and stabilizing, the small island stabilizing effects due to either transport and/or to polarization currents, and the seeding by other MHD events such as sawteeth, fishbones, or ELMs [La Haye 2006b]. As Δ' is sensitive to the details of the current density and q profiles, it is always problematic to get an accurate enough equilibrium reconstruction to believably calculate Δ'.

Research Plans for 2009–2013: The proposal for the next five years on DIII-D endeavors to answer questions raised by recent experiments regarding NTM seeding and threshold, A list of these research issues is given in Table 5-12. The first topic is tearing stability. It was observed on DIII-D that reducing the plasma rotation using a mix of co and counter beams made the tearing stability (without ECCD) worse in multiple regimes, *i.e.*, sawteething H-mode, the hybrid scenario, and the advanced tokamak [La Haye 2007]. Slower plasma rotation at rational surfaces may remove the stabilizing effect of flow shear that occurs in tokamaks, making NTMs less classically stable [Coelho 2007]. The dependence of this effect will be investigated. In addition, Δ' may also depend on whether q_{min}>1 or not [Brennan 2007]. Keeping q_{min}>1 transiently does increase the β stable to 3/2 modes by not having the sawteeth and concomitant seeding [La Haye 2000]. This line of research is especially important for the hybrid scenario with q_0~1, and accurately determining whether $q_0<1$ or $q_0>1$ may be the key to understanding the 2/1 tearing β limit near an ideal kink "pole" in Δ′.

Table 5-12
NTM Topics to be Investigated in Next Five Years on DIII-D

Topic	Key Physics	Issues
Tearing stability (Δ′)	Flow shear stabilization	How does Δ′ depend on flow shear?
		How does this scale with magnetic shear and Alfven time?
		What roles do the signs of the relative terms play?
	Poles	Can β limit for 2/1 mode be raised in hybrid scenario by increasing q_0 more above 1?
NTM threshold	Marginal island width	Does it scale the same for 2/1 mode as measured for 3/2 mode?
		Does it depend on rotation?
		Which physics dominates, transport or polarization?
	Flows around islands	What is the magnitude and sign of the threshold polarization current?
NTM seeding	Sawteeth and ELMs	Can sawtooth destabilization with ECH avoid triggering NTMs in high β H-modes?
		Is the NTM stable β limit higher in plasmas with RMP control of ELMs?
		Is the NTM stable β limit yet higher with combined sawtooth/ELM control?

The second topic is NTM threshold physics. Previous experiments on DIII-D showed that the empirical small island width w_{marg} for removal of the 3/2 NTM by either β ramp down (without ECCD) or by ECCD (at constant β) is ~ twice the ion banana width in existing tokamaks with "high" rotation [La Haye 2006a]. This research needs to be extended to 2/1 NTMs as well as "low" rotation cases. Using the ability of balanced NBI to slow the island rotation speed, we expect to make measurements of the flow in and around a rotating island during the 2009–2013 period for the first time anywhere in a tokamak. This is a key element of the magnitude and particularly the sign of the "threshold" polarization current. Ideally we want to measure all of the perturbed plasma quantities in the vicinity of the island for comparison with nonlinear MHD codes such as NIMROD. An example of a profile measurement of the helically perturbed current density using MSE data for a slowly rotating "quasi-stationary" 2/1 mode is given in Fig. 5-10. In the future we want to extent such measurements to rotating NTM islands.

The third topic is seeding of NTMs, particularly from sawteeth and ELMs. Figure 5-9 showed that ECH near the q=1 surface can obtain frequent, low amplitude sawteeth. Future experiments need to determine whether this kind of sawtooth control can be used to sustain ELMy H-modes at high β. Successful n=3 edge RMPs have eliminated ELMs in the sawteething H-mode at moderate β [Evans 2004], suggesting another source of NTM seeding control/removal to raise the NTM-stable β. We will also examine whether the NTM stable β limit is yet higher with both sawteeth and ELMs removed as seeds of NTMs. This may eliminate the need for lower efficiency off-axis ECCD control at multiple NTM rational surfaces. Stability of high β operational limits will be contrasted between seed control and direct off-axis ECCD.

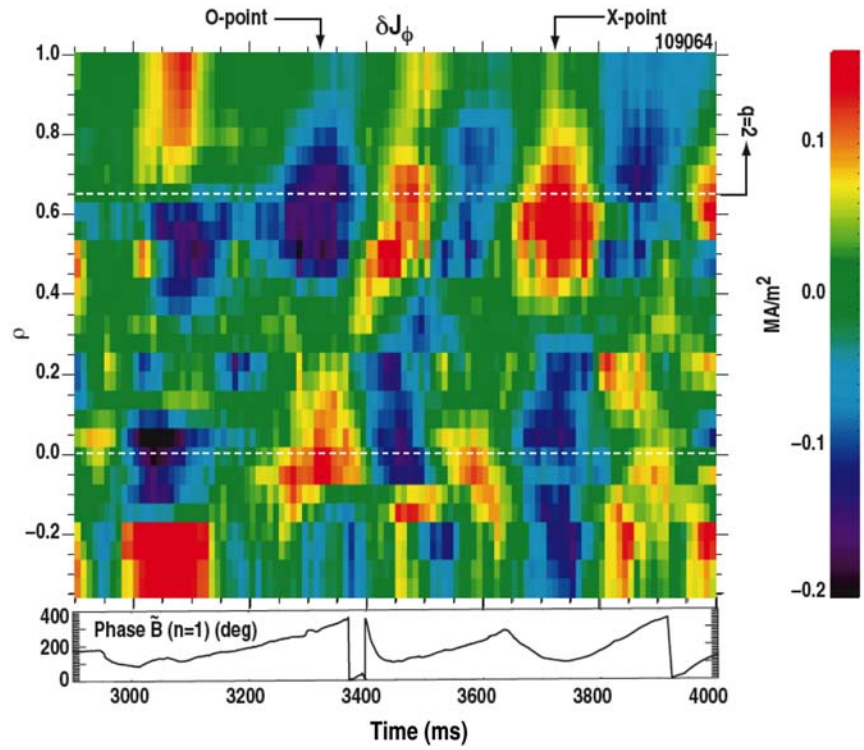

Fig. 5-10. Helical perturbations to the toroidal current density for a “quasi-stationary” 2/1 mode.

Hardware and Diagnostic Requirements: With its state-of-the art and unique tools, the DIII-D operation in 2009–2013 is well-positioned to address the open issues regarding NTM physics, including what is the exact value of Δ' (particularly at low rotation), what the small island threshold physics is due to, and how elimination of the major seeds can raise the NTM stable β limit. The major tools and diagnostics needed for this work are:

1. Counter-beam MSE view combined with co-beam views, giving greater accuracy in determining the E_{rad} correction to the MSE data and in locating rational surfaces. This will lead to more accurate determination of q_0 and the equilibrium Δ' analysis.
2. Balanced NBI (5 co and 2 counter beams, or vice versa with reversed I_P) allows reducing the plasma rotation to near zero to study the effects of reduced flow shear on NTM stability. The lower island rotation in tandem with better time resolved CER measurement of plasma rotation (and T_i) can address the issues of flow in an island and the polarization current. The new two-toroidal view ECE diagnostic for T_e, the increase in the number of BES channels for density fluctuations, and perhaps the MSE diagnostic (if the rotation speed is slow enough) will allow a complete suite of perturbed quantities in a tokamak island to be measured.
3. ECH for sawteeth control (destabilization or stabilization). The DIII-D PCS can fine tune the toroidal field value for sawteeth control by making real-time measurement of the sawtooth period and feeding back on B_T with ECH aimed at q=1 to keep the period short as β is increased.
4. RMP control of ELMs for eliminating this major seeding event of 2/1 NTMs.

5.3.3. Edge Stability

The edge stability questions to be studied in the 2009–2013 period on DIII-D deal mostly with ELM issues. The status and future challenges for edge stability research are given in Table 5-13.

Table 5-13
Status and Challenges of Edge Stability Research on DIII-D

Topic	Status	Challenges
QH-mode	Candidate for saturated peeling-ballooning mode	Develop full model for saturation mechanism
ELM energy loss	Filaments observed, structure consistent with peeling-ballooning model	Measure nonlinear evolution of ELM cycle at high space/time resolution Nonlinear modeling using NIMROD and BOUT codes
Impact of ELMs on material surfaces	20% of ELM heat load to first wall, filaments concentrate heat load	Document the spatial distribution of ELM heat loads Identify mechanism for propagation of ELM heat to the main chamber walls
Interaction between ELMs and NTMs	Coupling between ELMs and NTMs observed but not fully understood	Determine whether NTM triggering by ELMs is Δ' effect or seed island effect Find mechanism by which ELMs broaden current profile in hybrid scenario

5.3.3.1. Extension of Edge Stability Studies to the Nonlinear Regime. Work of the General Atomics theory group and the DIII-D experimental group has established the peeling-ballooning mode as the instability that sets the threshold condition for the ELM [Snyder 2005]. This understanding is one of the key components in developing a predictive capability for the kinetic pressure obtainable in the H-mode transport barrier, which in turn is expected to strongly affect the performance of reactor scale tokamaks. In the next five years this collaboration should continue with the goals of develop an understanding of the nonlinear phase of the instability and the ELM energy loss mechanism.

There is experimental evidence of the fast formation and ejection of plasma filaments during the ELM energy loss phase. The filaments have spatial structures similar to the structure of peeling-ballooning mode linear eigenmode. Theoretical work at GA and elsewhere indicates that such filaments might be expected in the nonlinear phase of the peeling-ballooning instability. The importance of these filaments in the ELM energy loss mechanism, however, remains unclear. There is also evidence for loss of the H-mode transport barrier during the ELM, which may account for some part of the ELM energy loss.

On DIII-D, studies of the spatial structure and temporal evolution of the nonlinear phase of the ELM instability over a range of ELM sizes coupled with computations with the nonlinear resistive MHD code NIMROD, and the nonlinear Braginski equation code BOUT, could provide insight into the ELM energy loss mechanism. Study of the nonlinear phase of the ELM instability requires diagnostics with high time resolution (10 microsecond resolution is required to observe the structure of the expanding filaments), high radial spatial resolution (~0.5 cm), but with large poloidal and toroidal coverage as the structures are extended along a field line. Reduction in signal to noise, increase in number of digitizer samples and improved mode structure analysis software is needed for magnetics and soft x-ray diagnostics. A gas puff imaging system could also improve understanding on ELM dynamics on DIII-D as it has on other tokamaks. An example of a CIII image of an ELM event on DIII-D, seen in Fig. 5-11, shows a filamentary structure similar to the simulated image from the ELITE code for an n=18 peeling-ballooning mode. The UCSD fast camera should also be more fully exploited in understanding ELM dynamics. Simulation of diagnostic signals needs to be built into NIMROD and BOUT.

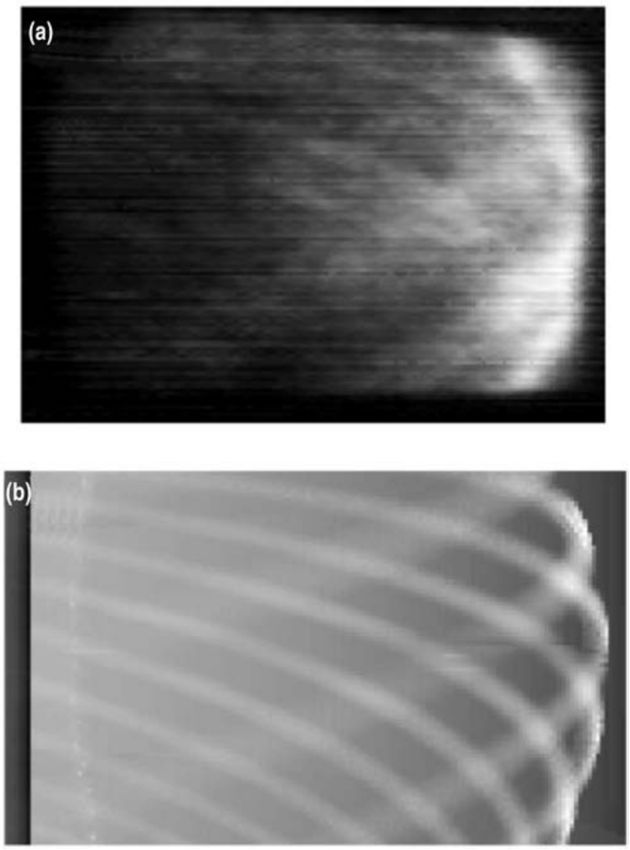

Fig. 5-11. (a) CIII image during ELM event. (b) Mode structure calculated by ELITE code for an n=18 peeling-ballooning mode [Snyder 2005].

Of particular importance is the study of the effect of rotational shear on ELM dynamics. The ELM-free QH-mode [Burrell 2005] may represent a regime in which access to the low n branch of the peeling-ballooning mode coupled with strong rotational shear in the edge leads to saturation of the instability. One of our goals during the 2009–2013 period is the development of a full model for the saturation mechanism. This would involve coupling of a model for the transport effects of the mode with a nonlinear edge stability code that includes the effect of rotational shear and wall drag. Rotation has also been shown to have a strong effect on the magnitude of the ELM energy loss on JT60-U. The addition of a counter neutral beam source on DIII-D provides a facility for the study of rotational effects on ELM dynamics and QH-mode access and characteristics. Rotational shear effects are included in the NIMROD code allowing detailed comparison with experimental results.

5.3.3.2. ELM Crash and Energy Loss. Another important feature of the ELM energy loss mechanism is the poloidal and toroidal distribution of the energy loss. Although most of the ELM energy loss is deposited in the divertor, up to about 20% of the energy loss appears to go to the main chamber walls. Even this relatively small fraction of ELM energy loss to the vessel wall is of concern since these surfaces are designed to withstand much lower heat loads than the divertor. Evidence from DIII-D and other machines suggests that the ELM impact to the main chamber wall may come in the form of filament structures which would further concentrate the heat loads. An effort should be made in the next five years to document the spatial distribution of ELM heat loads and to identify the mechanism for propagation of ELM heat to the main chamber walls and connect this mechanism to the ELM instability process. Improvements in diagnostics mentioned above would aid in this work, also additional IR camera views. Previous experiments looking at reciprocating probe and BES measurements have provided insight into the formation and ejection of filament structures during ELMs and these experiments should be continued in the 2009–2013 time frame, and comparisons to nonlinear ELM simulations with the BOUT or NIMROD codes should be made.

5.3.3.3. Interaction of ELMs and NTMs. ELMs can trigger neoclassical tearing modes in the plasma core resulting in loss of core confinement or even a plasma disruption. Possible mechanisms for this effect are increased pressure gradient at the NTM mode rational surface as the ELM cold pulse propagates inward, or creation or expansion of the NTM seed island through magnetic coupling to modes associated with the ELM. It is important to understand this effect in more detail in order to predict if it will be a problem in future tokamaks. Experimental results on DIII-D compared with NIMROD calculations could provide insight into this important effect. In addition, analysis of MSE data in hybrid plasmas indicates that an interaction between ELMs and the 3/2 NTM may redistribute (broaden) the current profile [Petty 2007]. As this may be playing an important role in maintaining the q_0>1 in hybrid plasmas, this coupling needs to be better understood.

5.3.4. Nonaxisymmetric Error Fields and Plasma Effects

Motivation and Status: The tokamak magnetic field is conceptually toroidally axisymmetric and two-dimensional. However, departures from axisymmetry in real machine magnets and nearby magnetic objects are inevitable, and they introduce weak three-dimensional, nonaxisymmetric magnetic field components, usually called "error fields". In the absence of plasma rotation, the resonant helical Fourier harmonics (those whose phase fronts $(m\theta - n\phi) = constant$ align with the magnetic field, *i.e.*, $m/n = q$, where ϕ is the toroidal angle and θ is the poloidal angle in a straight-line magnetic coordinate system) make driven magnetic islands on rational-q surfaces that reduce confinement. Fortunately tokamak plasmas rotate, either naturally or from neutral beam injected angular momentum. If the plasma rotates with sufficient toroidal speed to shield the radial component harmonics $\tilde{B}_{m,n}$ by the v_ϕ x $\tilde{B}_{m,n}$ driven parallel currents, then the residual islands remain small and the toroidal magnetic surfaces are nearly perfect. However, this shielding is slightly dissipative in finitely resistive plasma, and there is a braking torque. Additional braking torque applied by nonresonant error components, which are not shielded, can be significant if the nonresonant components are large. When the resonant braking exceeds a critical magnitude, the plasma slows nonlinearly and rapidly, until at low enough speed the plasma shielding fails and a large driven island locked to the error field appears. The appearance of such a "locked mode" is accompanied by notably degraded plasma confinement and even disruption.

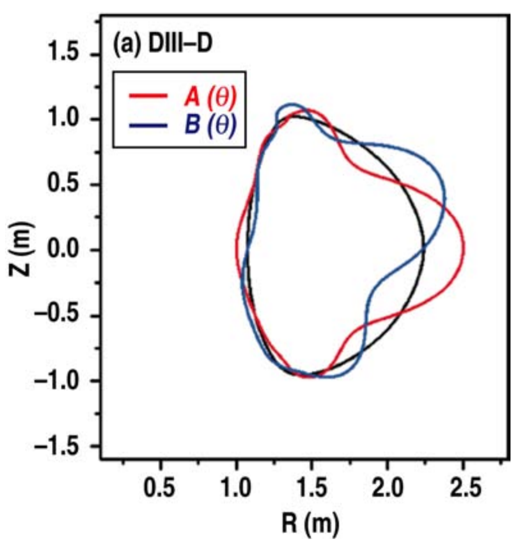

Fig. 5-12. Distributions of the external error fields on the plasma boundary. {[Park 2007] Reprinted Fig. 7a with permission from J.-K. Park et al., Phys. Rev. Lett. **99**, 195003 (2007). Copyright 2007 by the American Physical Society.}

Error fields are undesirable even when locked modes are not a problem. They distort plasma surfaces, and when the direct displacements are amplified several-fold by weakly damped ideal MHD modes, as seen in Fig. 5-12, they may be the cause of unexplained inconsistencies in the mapping of experimental diagnostic data onto axisymmetric reconstructed magnetic surfaces. Decreased plasma rotation, even without locking, degrades plasma confinement and stability. For all these reasons it is important to reduce magnetic errors.

Research Plans for 2009–2013: The long-term goal of error field research at DIII–D is to develop a predictive theory of the plasma consequences of error fields and how to efficiently reduce them by realistic, imperfect external correction fields. A list of the codes, tools, and diagnostics needed for this work is given in Table 5-14. Recent pioneering numerical results with the new IPEC perturbed 3D equilibrium code [Park 2007] shows that the nonaxisymmetric plasma currents can amplify some components of the external perturbing field by ~10 times — a truly major new effect. A new equilibrium fitting code for weakly 3D plasmas, EFIT3D, is presently being developed for DIII-D use. EFIT3D should yield realistically deformed magnetic surfaces in response to known and/or hypothesized errors and the self-consistent 3D plasma currents. This will enable more precise mapping of diagnostic information from all around the torus onto the 3D magnetic surfaces, which will be especially important in understanding the high-gradient edge pedestal region. The computed 3D equilibria will guide and check the development of analytic theories of plasma amplification. The code will help support the science of dynamic and static error correction. Equilibria with realistic internal 3D fields will be the starting point of calculations of resonant and nonresonant braking during the good confinement phase, before the plasma descends into a locked mode. New codes to calculate braking quantitatively will have to be developed.

Table 5-14
List of Codes, Tools, and Diagnostics for Error Field Research on DIII-D

Category	Item	Purpose
Codes	IPEC	Perturbed 3D equilibrium code
	EFIT3D	3D equilibrium code
External Control	Redesign B-coil current feed at 30 deg	Reduce large source of high-n error fields
	I-coils and C-coils	Can correct low-n error fields (may need additional power supplies)
	Inner wall coils	Improved spatial distribution of radial field for canceling error field at boundary
Diagnostics	Additional external magnetic measurements at different poloidal and toroidal locations	Needed input for EFIT3D
	ECE at two toroidal locations, toroidal SXR	Directly measure asymmetries in flux surfaces

While the advances in theoretical and computational tools are important, experiments remain essential to providing quantitative guidance for practical error field correction. Theoretically motivated corrections failed frequently in the past, but this situation is expected to improve as experiments can be analyzed with the aid of 3D equilibrium fitting codes and quantitative braking calculations. The relevance of the DIII–D error field experimental program to new and future superconducting tokamaks would benefit by a reduction of the existing large, high-n error from the B-coil current feed at 30°, which is peculiar to DIII–D. A concept test experiment in 2007, temporarily using the nearby I-coil IL30 loop, showed that its correction would also reduce mode locking in DIII–D. Absent the 30º feed error, DIII–D would be dominated by (a) n=1, high-|m| error harmonics from poloidal field coil misalignments, and (b) n = 1 and 2 errors from toroidal field coil geometric irregularities. Reduction of the high-n should make error correction by the existing I-coils and C-coils more effective, yielding both better operation and simpler interpretation of experiments.

Theoretical studies with the IPEC code [Park 2007] suggest that the total nonaxisymmetric field in the plasma depends mainly on how the external fields couple to the least stable MHD mode(s). If true, this effect will simplify practical error correction. Good quantitative tests of this and future theories may require more general currents in the I- and C-coils than have been possible in the past. Additional power supplies might be required for these coils. In addition, error correction should improve as the spatial distribution of the radial component of the correcting field more closely matches and cancels the error field on the boundary surface, which is the goal of the new inner wall coils.

Hardware and Diagnostic Requirements: EFIT3D requires new magnetic diagnostics of the radial and poloidal magnetic field in poloidal arrays at a minimum of four different toroidal locations (see Diagnostics Section 8). Other diagnostics are desirable to check and constrain EFIT3D fits. A second ECE diagnostic, located at a judiciously chosen second toroidal location, could help determine the widths of nonrotating magnetic islands, and they might also detect and quantify amplified ideally displaced closed magnetic surfaces. Upgraded soft x-ray arrays, with densely packed, carefully characterized view chords at multiple toroidal locations would complement and extend the second ECE diagnostic.

5.3.5. Extended MHD

Motivation and Status: Research in MHD over the next five years will see a shift in paradigm as nonideal and nonlinear effects – collectively referred to as ‘Extended MHD’ – become the focal point rather than linear ideal MHD. Over the past two decades the experimental and analysis tools required to compare the linear ideal theory with the experimental observations have matured to the point where mode structures and growth rates, in addition to the stability limits, can be compared and discrepancies documented. While linear ideal theory provides a good starting point for describing the stability and related phenomena in DIII-D, the documented discrepancies show the importance of nonideal and nonlinear effects. Given this success and the availability of new and upgraded diagnostics and state-of-the-art computational tools, we are now in a position to study these extended MHD effects in quantitative detail.

There are many traditional ‘Extended MHD’ phenomena – sawteeth, NTMs, RWMs, ELMs, and AE modes – that will be addressed by DIII-D in the next grant period. Table 5-15 shows that many lines of research are common to these phenomena. In general, the role of plasma rotation is an important issue. Nonlinear evolution is also a largely unresolved issue in general. An understanding of the nonlinear conditions involved in ELMs may yield an explanation of the EHO and other essentially ELM-free regimes. A related issue is the role of dissipation and other nonideal effects in stability analyses. In general, nonideal effects provide a dissipation mechanism and therefore tend to be stabilizing. Prominent examples are resistivity and resistive walls. However, nonideal effects can provide additional instability pathways leading to new modes. Thus, plasma resistivity leads to tearing modes, wall resistivity to the RWM, and fast ions to new fast ion driven instabilities such as fishbones. Over the 2009–2013 time frame we will take advantage of the new diagnostic capabilities and new modeling capabilities to study the fundamental physical mechanisms that lead to new instability paths – reconnection for example – in more detail than has previously been done. The role of 3D effects is another area of general interest. Tools being developed to calculate the plasma response to nonaxisymmetric fields include the NMA code (for the RWM), and the DCON and GATO codes (for more general cases). The NIMROD code additionally can provide a full nonlinear response if run to saturation, though this is computationally demanding.

Table 5-15
Common Lines of Research in Extended MHD

Topic	Issues
Plasma rotation	Need to untangle effects of bulk fluid rotation and the separate species two-fluid effects Need to untangle interactions with static error fields and the individual modes or islands
Nonlinear evolution	Need to explain bursting behavior of sawteeth, fishbones, AEs, ELMs, *etc*. Theoretical analysis involves full-scale coupling of the stability with transport and slower evolution
Nonideal effects	Explore trade off between stabilization of ideal modes and destabilization of new modes Determine the physical mechanisms that can lead to new instability paths
3D effects	Need to evaluate plasma response to static error fields and nonlinearly saturated modes

The analysis tools available for this computationally intensive research include, first and foremost, equilibrium analysis provided through the EFIT code. Additionally, TEQ and TOQ provide flexibility to study variations around the reconstructed discharge equilibria. An array of standard stability codes, including GATO, DCON, MARS, and NOVA for low to intermediate toroidal modes, ELITE for intermediate to high n, and BALOO, MBC, and CAMINO for high n provides detailed predictions of the ideal linear MHD stability. Nonideal effects are included in MARS (resistive effects, rotation, and some kinetic effects), MBC (resistive and some kinetic effects), NOVA (kinetic effects in an extension NOVA-K), and ELITE (rotation effects). Additionally, the PEST3 linear resistive MHD code has been adapted for study of DIII-D discharge equilibria. Full extended MHD analysis can also be done through the M3D and NIMROD codes, which have been extensively used to study a variety of MHD phenomena in DIII-D, including sawteeth, NTMs, ELMs, and also disruptions.

Hardware and Diagnostic Requirements: DIII-D has arguably the best diagnostic set of any fusion experiment. In the case of equilibrium diagnostics, this is definitely the case and our experience is that the accurate and detailed reconstruction of the discharge equilibria is crucial to obtaining the detailed comparisons with theory necessary for reproducing the linear ideal case and extracting the extended MHD effects. The diagnostics are described elsewhere but a key component is the extensive external magnetic arrays and the internal magnetic field pitch angle measurements used to obtain the current profile. In addition, an extensive and highly flexible array of fluctuation diagnostics, including ECE, MSE, BES, reflectometer, and SXR, all of which have recently been upgraded, provide unprecedented information on the MHD activity in DIII-D. The addition of fast ion loss diagnostics, in conjunction with various analysis tools described below, will also add to our ability to provide data on the fast ion population – a key ingredient in several of the extended MHD phenomena of interest to both DIII-D and ITER.

5.4. ENERGETIC PARTICLES

The study of the interaction of energetic particles with plasma is simultaneously a topic of great interest for plasma science and one of immense importance for the practical realization of fusion energy. In plasma science, the nonlinear dynamics of fast-particle driven instabilities occur in a novel regime where both wave-wave and wave-particle nonlinearities are important. Moreover, instabilities that are driven by energetic-particle populations are common in stars and planetary atmospheres. For fusion energy, the successful confinement of charged fusion products is essential in a fusion reactor.

The DIII-D Energetic Particle research program strives to validate theoretical models of fast-ion instabilities and to develop the means to control their effects. This research addresses the following questions from the FESAC Priorities Report:

T12. How do energetic particles interact with plasma?

The emphasis in the next five years will be on model validation. The DIII-D program is positioned to lead the international effort to provide high-quality data that rigorously tests theoretical models and simulation codes. In collaboration with theoretical efforts worldwide, our goal is to develop validated codes that can predict fast-ion instabilities and their consequences in ITER. In control, our goal in the 2009–2013 time frame is to lay the groundwork for an ambitious program in the subsequent five year period.

Several instabilities are important in tokamaks at different frequencies, as seen in Fig. 5-13. Attention will focus on the Alfven instabilities that are the most relevant for ITER, such as the toroidicity-induced Alfven eigenmode (TAE) and reversed shear Alfven eigenmode (RSAE). Note that the newly discovered beta-induced acoustic Alfven eigenmode (BAAE), which exists at much lower frequencies, is not included in Fig. 5-13. The tools developed for Alfven mode studies will also prove useful for studies of the sawtooth instability and of the interaction of neoclassical tearing modes with fast ions.

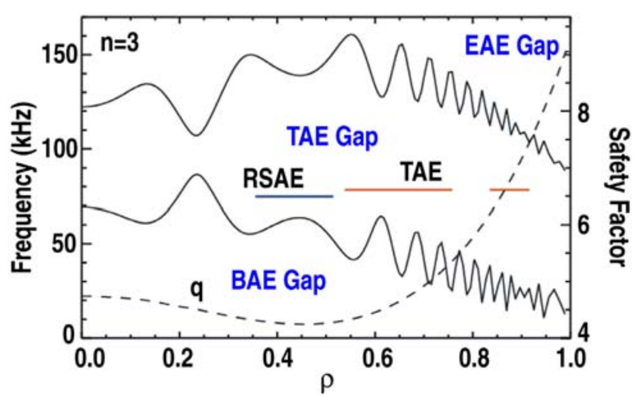

Fig. 5-13. Diagram of “gaps” that allow various AEs to exist, as a function of radius and frequency.

5.4.1. Validation Plan

The DIII-D program will supply data that will test specific energetic particle (EP) physics models. The process of validation must address *all* levels of the models, including the most fundamental ones. The physics models build upon a hierarchical construction (sometimes called the “primacy hierarchy”) that begins with fundamental constituents and results in fast-ion transport predictions. The elements of this hierarchy are listed in Table 5-16. At the most fundamental level are the linear wave properties of the instabilities under study. These include the polarization, frequency, spatial structure, and stability threshold. Next are the mechanisms that determine the ultimate mode amplitude, which include both wave-wave and wave-particle interactions. The wave electric and magnetic fields modify the EP distribution function (DF) and cause fast-ion transport, which is something that can be observed experimentally. If these phenomena are understood for a particular case, then the next level of validation testing is parametric scaling. The final validation stage is agreement with trends across multiple devices.

Table 5-16
Primary Hierarchy for Validation of EP Turbulence and Transport Predictions

Primacy Hierarchy	Observables	Agent/Mechanism	New Diagnostic
Linear wave properties	Polarization, structure, frequency, threshold	EP spatial gradient, velocity anisotropy	CO_2 array, B components, polarimetry
Nonlinear saturation	Spectral intensity, bispectra, zonal flows/fields	Wave-wave, wave-particle interaction	BES, Edge loss, NPA, Neutrons
Transport	Distribution function and transport	Cross-phase, relaxation	FIDA
Scaling trend	Similarity experiments	Dimensionless scaling	
Statistics	ITPA database	Inter-machine	

The current status of experimental research on DIII-D in regard to the primary hierarchy is that the linear wave properties are in excellent agreement with theory [Van Zeeland 2006] for some conditions. Studies of the mechanisms of nonlinear saturation are just beginning. In EP transport, present theories are unable to explain the observations [Heidbrink 2007]. Cross-machine comparisons have begun between DIII-D and NSTX, and between DIII-D and JET.

Diagnostic Requirements. A combination of existing and new diagnostics will provide a complete data set that will thoroughly constrain the theoretical models. Reliable validation requires excellent diagnostics in three distinct categories: wave fluctuations, fast ions, and background plasma. A list of the existing and future DIII-D diagnostics relevant to these categories is given in Table 5-17. In recent years, the DIII-D team has made tremendous progress in the diagnosis of wave fluctuations. A sensitive 40-channel ECE radiometer measures local temperature fluctuations at the electron volt level. Electron density fluctuations are measured by many diagnostics: CO_2 interferometers, BES, reflectometers, far-infrared scattering, and phase-contrast imaging. Three different types of magnetic coils measure magnetic fluctuations. Three enhancements of the current diagnostic set will be made in the next few years. First, we currently have no direct measurements of mode polarization. The installation of magnetic loop pairs that measure toroidal and vertical field components will distinguish between shear Alfven waves and modes with a significant compressional component, a technique recently utilized on NSTX [Fredrickson 2007]. New polarimetry diagnostics will help distinguish electrostatic instabilities from electromagnetic modes. Second, toroidal mode numbers are currently obtained from Mirnov coil arrays but core-localized Alfven modes are often barely detectable on the magnetics. An array of toroidally displaced CO_2 interferometers will remedy this deficiency. Third, the upgraded BES diagnostic is invaluable for local measurements of the density fluctuations and poloidal mode numbers but is currently limited to a small spatial volume (3 by 3 cm). Additional channels will extend the radial coverage to the entire outer half of the plasma.

Table 5-17
Diagnostics for Validation of EP Physics

Category	Existing Diagnostics	New Diagnostics
Wave Fluctuations	ECE, CO_2, BES, FIR, PCI, reflectometers, external magnetics	Magnetic loop pairs, polarimetry, upgraded BES
Fast Ions	FIDA, neutron counters, equilibrium fast ion pressure (MSE+ECE), fast ion loss detectors (foils)	Enhanced FIDA, fast ion loss detectors (scintillators), NPA, faster neutron counters
Background Plasma	CER, TS, ECE, CO_2, MSE	

Because of its multi-dimensional nature, diagnosis of the EP distribution function is particularly challenging. The distribution function depends on two velocity coordinates (energy and pitch) and is not a flux function in coordinate space. The current diagnostic set consists of a pair of dedicated fast-ion D-alpha (FIDA) channels [Heidbrink 2004, Luo 2007], the volume-integrated neutron rate, the fast-ion pressure from analysis of the equilibrium, and a pair of foils at the vessel wall that serve as beam-ion loss detectors. Three new systems and one enhancement are planned. A new 8-channel FIDA instrument will measure the fast-ion density profile on every beam-heated discharge and will provide additional information about the velocity distribution on selected discharges. The second system will measure fast-ion losses to the wall at numerous poloidal locations with sufficient temporal resolution to resolve fluctuations at the TAE frequency. The third system will use neutral particle analysis to complement the existing diagnostics (which all effectively average over velocity space) by probing narrow regions of velocity space. The fourth improvement is to increase the bandwidth of the existing neutron scintillators through improved acquisition electronics.

DIII-D is already equipped with diagnostics that measure the plasma parameters that impact EP physics. In particular, accurate determination of the q profile from MSE data is essential.

The final column of Table 5-16 shows the principal contribution of each of the new diagnostics to the validation program. The CO_2 array, polarimetry, and magnetic polarization measurements are necessary for correct identification of the underlying instability amongst the many different types of Alfven eigenmodes. In addition to their utility for mode identification, the extended BES array and the ECE radiometer are needed for the search for wave-wave interactions and any possible zonal flows associated with the wave couplings. The NPA provides localized measurements in velocity space in order to determine which class of particles exchanges energy with the waves. The high bandwidth edge-loss detectors and neutron scintillator signals are needed to search for wave-particle couplings, as well as for direct evidence of fast-ion transport. FIDA is the principal diagnostic for measurement of fast-ion transport by the instabilities.

Analysis and Experimental Conditions. An essential element in the validation program is the development of synthetic diagnostics that account for experimental resolution and uncertainties. An example of a recent comparison of the measured mode structure with a theoretical prediction is shown in Fig. 5-14. Excellent agreement is obtained only after the theoretical prediction is processed to include the spatial resolution of the ECE diagnostic [Van Zeeland 2006]. Similarly, to compare fast-ion measurements with theory it is necessary to use the predicted fast-ion distribution function in a simulation code [Heidbrink 2004] that predicts the FIDA data. The DIII-D team will develop the synthetic diagnostics necessary for meaningful comparisons of EP data with theory.

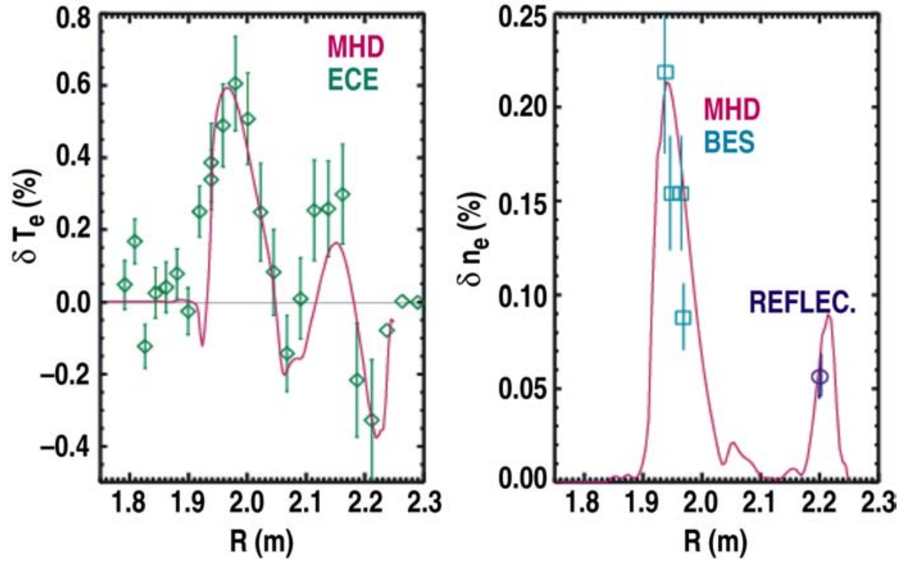

Fig. 5-14. Electron temperature and density fluctuations predicted for a *n*=3 RSAE by the NOVA code (solid) with ECE, BES, and reflectometer measurements (symbols) [Van Zeeland 2006] .

New analysis techniques are also needed. As EP physics progresses from linear mode identification to the complexities of nonlinear dynamics, many of the tools that are commonly used in thermal transport studies, such as bicoherence techniques [Crocker 2006], must be adapted and employed.

The latter stages of validation require variations in experimental parameters. The flexibility of the DIII-D facility accommodates a wide range of experiments that test the underlying dependencies. Readily altered parameters of importance in EP theory include the following.

- Fast-ion distribution function, which is controlled by the co/counter neutral beam mixture and by perpendicular acceleration of the fast ions through fast-wave heating.
- Plasma shaping, which alters the damping of Alfven eigenmodes in linear theory.
- The q profile, which strongly influences the Alfven gap structure and the resulting eigenmodes.
- Field strength, which alters the ratio of fast-ion speed to the Alfven speed.
- Helical field perturbations, which may alter mode coupling and EP transport.

A major emphasis in the experimental program will be the creation of plasma conditions with differing nonlinear dynamics. For example, conditions with many high-*n* modes [Nazikian 2006, Kramer 2006] are more likely to exhibit wave-wave couplings than conditions with a few coherent low-*n* modes. This flexibility is invaluable for comparative experiments between facilities. For example, DIII-D is able to closely match all NSTX parameters except for major radius, which differs by a factor-of-two between the devices [Heidbrink 2003]. With recent advances in EP diagnostics on both devices, new comparative experiments between the devices will provide a rigorous test of the theoretical models. Of particular interest is a striking difference in nonlinear dynamics: fast-particle instabilities in NSTX often "chirp" rapidly in frequency, while this behavior is rare in DIII-D.

Collaborations. National and international cooperation are essential to the success of this validation program. DIII-D experimentalists already collaborate with leading analytical theorists at UT Austin and UC Irvine. Experimental results are compared with simulation codes developed at Princeton and Frascati. Plans have already been formulated to validate the new codes developed through the SciDAC Energetic Particle initiative. DIII-D personnel lead the EP group in the U.S. Burning Plasma Organization and will coordinate national validation efforts. Internationally, personnel who represent the U.S. in the ITPA will distribute outstanding DIII-D cases to the international EP modeling community.

5.4.2. Control Tools

A long-term goal of EP research is to exploit fast-ion driven instabilities to improve plasma performance, which is referred to as "phase-space engineering". Two ambitious examples of the phase-space engineering that will be explored during the next five years on DIII-D are:

1. Fisch's idea to use a pair of waves to channel alpha-particle energy directly into fuel ions [Fisch 1994].
2. Wong's idea to use TAE transport to form a shear layer that improves thermal transport [Wong 2005].

At a less ambitious level, one can imagine using waves at modest power levels to control the nonlinear dynamics of unstable modes, *e.g.*, to replace large unpredictable bursting events with steady EP transport. Fig. 5-15 shows an example where the application of ECH at the q_{min} surface suppressed RSAE activity on DIII-D [Van Zeeland 2008].

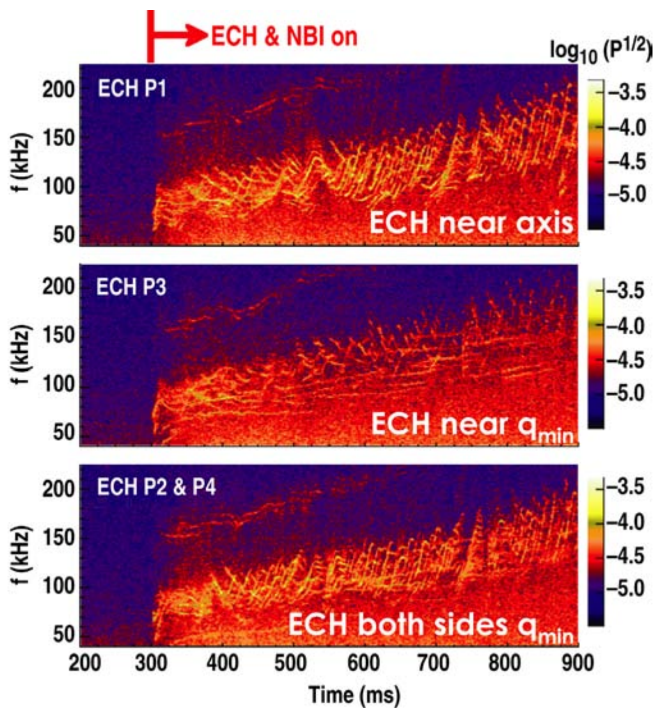

Fig. 5-15. AE activity measured with a CO_2 interferometer. For (b), the RSAE mode is suppressed during ECH near the q_{min} surface [Van Zeeland 2008].

Our objective in the 2009–2013 period is more modest: we plan to test the effect of existing control tools on several EP instabilities. These data will also provide parametric tests of the physics models (the fourth column in the validation hierarchy of Table 5-16). Available control tools, which are summarized in Table 5-18, include the following:

1. **ECH.** This has been shown to have a strong effect on AE stability. Changes in local magnetic shear through ECCD is another potential control tool that will be explored.
2. **ICH.** Application of perpendicular heating to electrons that were driving an interchange instability altered the nonlinear dynamics in a dipole experiment [Maslovsky 2003]. By altering

the effective collision frequency of fast ions that are trapped in instability wave fields, ICH might alter the nonlinear dynamics of instabilities in DIII-D. The interaction between fast waves and injected deuterium ions is strongest for the 60 MHz system (fourth harmonic, 2 MW source power), but interaction using the 90 MHz system (sixth harmonic, 4 MW source power) is also possible.

Table 5-18
Control Tools Available for EP Studies on DIII-D

Control Tool	Parameter Being Controlled	Purpose
ECH	Electron distribution function, local magnetic shear	Stabilize AE modes (such as RSAE)
ICH	Ion distribution function	Alter nonlinear dynamics by altering effective collision frequency of fast ions
I-Coils	Helical magnetic field perturbations	Induce orbit stochasticity for energetic particles
Co/counter NBI, magnetic braking	Shear in toroidal rotation	Distinguish between flow shear stabilization and changes in fast-ion distribution function
On-axis/off-axis NBCD, ECCD, FWCD	Safety factor profile (including magnetic shear)	Alter AE gap structure, AT studies

3. **I-Coils.** In the presence of sufficiently large helical magnetic field perturbations, resonances between drift orbit harmonics and the field perturbations can induce intrinsic orbit stochasticity for energetic particles [Mynick 1993]. One can also imagine resonances between energetic particles that are trapped in the instability wave field and static perturbations that selectively enhance the transport of energetic particles. By modifying the fast-ion transport, the nonlinear dynamics could be affected. Tests of these ideas using existing hardware will be developed and tested.
4. **Rotation sheared toroidal rotation** may alter Alfven eigenmode stability and impact nonlinear wave-wave coupling. In recent experiments, changes in rotation correlate with fast-ion transport. Carefully designed experiments that utilize magnetic braking and variations in co/counter beam torque will distinguish between changes in instability behavior associated with rotation and changes caused by modification of the fast-ion distribution function.
5. **Current drive** varies the safety factor profile, and with it the magnetic shear. While this is useful for altering AE gap structure, it also allows the energetic particle studies to be extended to advanced tokamak (AT) regimes with q_{min}>2 and negative magnetic shear. This is critical for evaluating the importance of AE modes on DEMO.

Armed with the results and understanding from these studies, in the latter portion of the five-year period we will propose hardware modifications such as a dedicated antenna for use in the 2014–2018 time frame.

In summary, DIII-D is already an outstanding experimental facility for the study of EP instabilities with the best fluctuation diagnostics in the world, excellent MSE, competitive fast-ion diagnostics and great flexibility in plasma conditions and the fast-ion distribution function. Moreover, the U.S. brings strong expertise to EP physics in experiment, theory, and computation. With additional diagnostics and a focused experimental effort, the DIII-D team can lead the international effort to develop validated codes that can predict fast-ion instabilities and their consequences in ITER.

5.5. HEATING AND CURRENT DRIVE PHYSICS (WAVE/PARTICLE)

Current drive and heating studies aim at developing comprehensive, predictive models for neutral beam current drive (NBCD), electron cyclotron current drive (ECCD), and fast wave current drive (FWCD). In addition, research on the self-generated bootstrap current is in this topical area. This research addresses the following questions from the FESAC Priorities Report:

T11. How do electromagnetic waves interact with plasma?

Experiments in this area support the use of our heating and current drive tools in many other areas of the DIII-D research program. The important diagnostics for studying heating and current drive on DIII-D are listed in Table 5-19. The MSE diagnostic is crucial for current drive studies, and the LIB diagnostic is needed for measurements of the edge bootstrap current. Our quantitative measurements of the driven current profiles are greatly facilitated by our ability to change the current drive direction from co to counter while observing the slight changes in the magnetic field pitch angles; this is a relatively new capability of the NBI system.

Table 5-19
Important Diagnostics for Heating and Current Drive Studies

Topic	System	Quantities to Measure	Diagnostics
Heating profile	ECH, FW, NBI	Temperature and density profiles	ECE, CER, FIDA, TS, reflectometers, neutrons
Current drive profile	ECCD, FWCD, NBCD, bootstrap	Poloidal magnetic flux profile	MSE (co and counter views), LIB, external magnetics

5.5.1. Science Issues for ECH and ECCD

The basic physics behind electron cyclotron heating (ECH) and electron cyclotron current drive (ECCD) are comparatively well understood through development of “first principles” computer codes that have been extensively validated against experiment over a wide range of experimental conditions [Petty 2001], as presented in Fig. 5-16. In all cases, ECH work on DIII-D builds on the unique strengths of the facility of having a high power ECH system with fully steerable antennas and universal polarization capabilities and an MSE system with good resolution and the ability to view both counter-injected and co-injected neutral beams for unique determination of radial electric fields and for improved radial resolution.

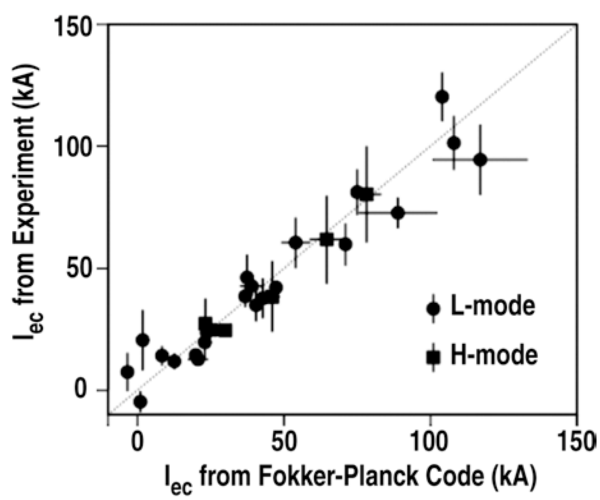

Fig. 5-16. Comparison of measured ECCD with theoretical value calculated using the quasi-linear bounce-averaged CQL3D Fokker-Planck code [Petty 2001].

Computer codes like TORAY-GA and GENRAY use linear models to evaluate wave damping and current drive, with current drive being a particularly sensitive test of the absorption model. Fokker-Planck codes, like CQL3D, are more general than the linear models and include effects due to moderately high power density, momentum conservation in electron-electron collisions, a collision operator that does not use the high velocity limit, and synergistic effects with the residual parallel electric field. Extensive experiments on DIII-D have shown that under DIII-D conditions the Fokker-Planck approach is a slightly better fit to the data, but under most conditions the linear approach is satisfactory as well as more convenient.

Work on outstanding physics issues for ECH/ECCD which will be done in the 2009–2013 time frame include:

- **Far off-axis ECCD** Comparisons between experiment and theory so far have been with $r/a \leq 0.5$, but applications like current profile control and MHD stabilization require current drive at larger r/a. There is no reason the theory should not still apply, but this must be checked for this strong trapping regime.
- **Quasi-linear effects** Calculations show that central ECCD at the high temperatures and densities of ITER is more efficient than the linear calculations suggest, due primarily to the electron-electron momentum conservation in collisions. When the ECH system on DIII-D reaches its full power of 12 MW, the difference between linear and quasi-linear calculations of ECCD can be directly tested at a collisionality and power density scaled to ITER conditions. This will improve the confidence in projections to ITER and extend the range of conditions under which the models have been validated.
- **Effect of islands on ECCD** A key application of ECCD on ITER is the control and suppression of neoclassical tearing modes (NTMs), which are otherwise expected to place a low limit on plasma performance. This control process is sensitive to the peak current density driven by EC waves rather than the total driven current. Three processes are theoretically expected to be important and will be evaluated:
  1. Even in a MHD quiescent plasma, diffusion of energetic electrons through normal transport processes is expected to broaden the ECCD profile, thereby diminishing its peak value. Some measurements of this have been made in DIII-D and other tokamaks, but detailed measurements require full operation of all chords of the MSE diagnostic, ideally with higher spatial resolution.
  2. A detailed evaluation of ECCD in the presence of NTM islands has not been made anywhere. Present evaluation techniques require axisymmetry, so extension to weakly nonaxisymmetric conditions must be developed. High frequency operation of the MSE diagnostic, at the mode frequency of ~10 kHz, also will be very helpful, as well as increased spatial resolution.
  3. Theoretical issues have been raised about ECCD on rational surfaces since there the EC wave may interact with a much smaller number of particles, generating a flattening of the local electron distribution function and reduced absorption of the wave. Again, increased resolution of the MSE system will improve these measurements.

5.5.2. Science Issues for FWCD and ICRF Heating

The primary role for the DIII-D Fast Wave (FW) system in this five-year period is to provide additional central electron heating power (without a core particle source) for the AT program. Raising the electron temperature increases the bootstrap current fraction, increases the current driven by EC waves, and makes the Te/Ti ratio more reactor relevant. By performing this electron heating role with co-current-drive and counter-current-drive antenna phasing on the three FW antenna arrays (0 deg, 180 deg, and 285/300 deg), and by adjusting the power levels in the three systems, the central non-inductive current density can be fine-tuned to vary the seed current or to modify the evolution of the central current density. The FW system will also be used in support of various wave-particle science experiments, i.e., those directly related to the physics of FW coupling, heating and current drive and as a tool for the study of other Fusion Science topics, as described below.

5.5.2.1. FW Antennas. The basic physics of Fast Wave (FW) heating and current drive in the Ion Cyclotron Range of Frequencies (ICRF) is quite different from that of ECH, for two fundamental reasons. The absorption mechanisms for the fast wave in the parameter range accessible to DIII-D are comparatively weak, so that instead of complete absorption of the waves on the first traversal of the power through the plasma (the usual ECH situation), multiple passes through the plasma are required to damp coupled FW power in the plasma core. If the single-pass absorption in the core is comparable to losses per bounce from the plasma edge, the power lost to edge absorption can be a significant fraction of the coupled power. Increasing the electron temperature, with either the FW or EC systems, improves the first-pass direct electron absorption of the FW and increases the current drive efficiency of both FWCD and ECCD.

The other fundamental difference between the EC and FW physics is that the EC wave that propagates up to the damping location in the plasma core also propagates in vacuum. Hence the wave launcher can be distant from the plasma surface and the wave coupling is insensitive to the plasma/wall distance, edge density profiles, ELMs, *etc*. In the FW case, the waves that propagate in the bulk of the plasma (without experiencing any *upper* density limit, unlike the EC waves) are evanescent in vacuum. The wave launcher must therefore be located close to the plasma surface in order to excite waves in the plasma by tunneling through the low density region at the plasma edge. Therefore, FW coupling is intrinsically sensitive to the plasma conditions in the immediate vicinity of the launcher and in particular is degraded by large antenna/plasma gaps.

The DIII-D FW system has reached the point of development where the limit to the power that can be coupled to standard H-mode discharges is set by breakdown in the wave launchers, not by the transmitter capability (up to 6 MW) or by the transmission lines. To substantially increase the range of experiments in which the FW systems are useful, the antennas must be substantially improved. The antenna at 285/300 deg, which is the oldest antenna (designed in 1989), is not actively cooled, so it cannot be used at full power for more than 2 seconds. We propose to replace this antenna first, with an actively cooled wave launcher of significantly lower impedance, in order to be consistent with the longer pulse requirements in this period and to substantially increase the power that can be coupled to H-mode plasmas with outer gaps of 8–10 cm. Such a structure is relevant to ITER, since ITER requires FW couplers to operate with outer gaps in the 15–20 cm range. The understanding of the physics of antenna breakdown and other limitations must be improved. Owing to the high power density environment near the antennas on DIII-D, in-situ diagnostics to study the breakdown processes are not planned (although improved camera views of the antennas are desired). Rather, studies of antenna breakdown physics done in dedicated bench-top facilities elsewhere (ORNL, MIT) will complement the study of antenna performance on DIII-D. Ideas involving localized modification of the plasma near the wave launcher to improve the coupling without decreasing the outer gap will be evaluated as part of these studies.

5.5.2.2. FW Absorption and Current Drive. Studies of FW Current Drive (FWCD) performed in the 1990s on DIII-D showed that the current drive efficiency was in agreement with theory under conditions where the single-pass absorption by direct electron damping dominated other absorption mechanisms (ion cyclotron harmonic absorption on neutral beam ions, edge losses) [Petty 2005b]. An example of the successful experiment/theory comparison is shown in Fig. 5-17. It was also shown that under conditions where the tunneling region at the plasma edge disappears due to rapid small ELMs fueling the edge, the edge losses dominate the core absorption and the current drive efficiency is strongly reduced.

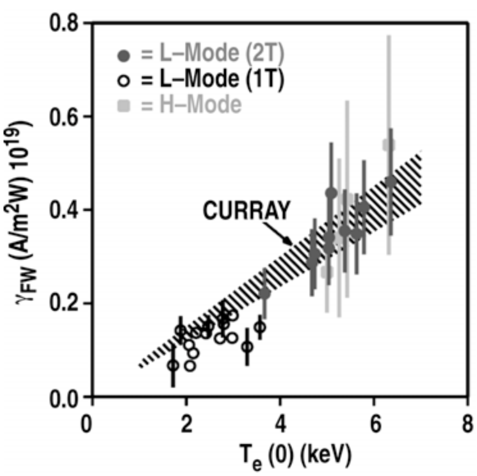

Fig. 5-17. Figure of merit for FWCD as a function of central electron temperature on DIII-D [Petty 2005b].

Table 5-20 lists the scientific issues to be explored in the 2009–2013 time frame, which include high harmonic damping, rf edge losses, and current drive. DIII-D FW experiments since 1998 have concentrated on quantifying the absorption by neutral beam ions at the third through eighth cyclotron harmonics. The subject is relevant to the damping of FW power on energetic alpha particles in a DT plasma, as well as to the use of FWCD in DIII-D plasmas with substantial fast ion populations. UC Irvine collaborators have developed an innovative diagnostic technique (Fast Ion D-Alpha charge exchange spectroscopy) that has facilitated the study of rf/energetic ion interactions on DIII-D. The DIII-D results on harmonic absorption have stimulated substantial efforts in the modeling community (see Section 6.2.1.3), with comparisons made to codes such as CURRAY, ORBIT-RF, CQL3D, and AORSA. We propose to extend these experiments and to continue to work closely with the modeling community (RF SciDAC) with the goal of developing a quantitative model of FW absorption in the presence of energetic ion populations and high electron temperatures, so that the partitioning of power between ion acceleration and direct electron absorption can be successfully predicted. The acceleration of injected beam ions by FW harmonic absorption may also be useful to the DIII-D program for controlling sawtooth behavior [Heidbrink 1999].

Table 5-20
Scientific Issues for FW Studies on DIII-D

Issue	Key Diagnostics	Applications
High harmonic FW damping	FIDA, neutron counters, equilibrium fast ion pressure (MSE+ECE)	Sawtooth stabilization, TAE mode control
RF edge loss	Variety of edge diagnostics	Avoid impurity generation
FWCD	MSE	Full noninductive current drive, q_0 control

The coupling of high FW power into DIII-D plasmas under conditions with substantial edge losses in conjunction with the edge diagnostic suite on DIII-D may permit direct quantitative measurements of the edge loss mechanisms. This goal has not yet been achieved on any tokamak; evidence for losses in rectified sheaths is fairly indirect so far.

The completion of the high power EC system on DIII-D will also allow the FW system to reach the goal for which it was originally designed: FWCD studies in moderate density plasmas with reactor-scale electron temperatures. Early design studies showed the possibility of driving MA-level currents in ~10 keV plasmas with high beta using the combination of neutral beam current drive, ECCD and FWCD. Control of the central safety factor can also be accomplished with FWCD.

5.5.3. Science Issues for Neutral Beam Current Drive

Although the neutral beams are an important source of heating and external noninductive current drive for DIII-D, our understanding of neutral beam physics has not been validated to the same degree as for electron cyclotron waves or fast waves. This is due to (1) the relative ease of measuring the plasma response to direct electron heating compared to ion heating, and (2) the long-standing ability to switch between co- and counter-current drive between shots for electron cyclotron waves and fast waves. Previous experiments on DIII-D have measured the NBCD profile and compared it to theoretical calculations for a few cases [Politzer 2005], but no systematic investigation into the physics of NBCD has been undertaken. The original paper that describes how to use the Maxwell equations and Ohm's law to experimentally determine the noninductive current profile shows that the total noninductive current agrees (to roughly 1σ) with the sum of the calculated NBCD and bootstrap current [Forest 1994]. Evidence for the influence of tearing modes on NBCD in DIII-D was also reported [Forest 1997].

Since NBCD plays an important role in controlling the current profile for Advanced Tokamak scenarios in DIII-D, and possibly in future burning plasma experiments as well, we need to experimentally validate the essential physics contained in our theoretical models of NBCD. The effects to be experimentally tested fall into two categories: classical and anomalous. Over the next five years, neutral beam physics experiments will study the dependence of the NBCD on:

- Classical effects
  - Electron temperature
  - Plasma impurities
  - Injection energy
  - Deposition location (*i.e.*, off-axis)
- Anomalous effects
  - Turbulent transport
  - Tearing modes
  - Energetic particles modes
  - Simulations of alpha channeling

Changing the geometry of a DIII-D beamline to give off-axis deposition rather than on-axis deposition will give DIII-D more flexibility in sustaining Advanced Tokamak scenarios with high q_{min}. However, testing the dependence of NBCD on deposition location is important given reports from Axisymmetric

Divertor Experiment Upgrade (ASDEX-U) that the current drive efficiency for off-axis NBCD can be well below the theoretical value [Günter 2005]. New theoretical predictions [Murakami 2007], shown in Fig. 5-18, predict that the off-axis NBCD is sensitive to the direction of the twist of the magnetic field lines, which may possibly explain the ASDEX-U results. Initial investigations of off-axis NBCD can be made with the existing beam geometry by moving the plasma axis up or down by ~0.4 m with a reduced minor radius. The DIII-D MSE system is essential for these studies; by comparing co/counter NBI, localized driven currents as small as 1% of the total plasma current can be measured accurately [Petty 2002].

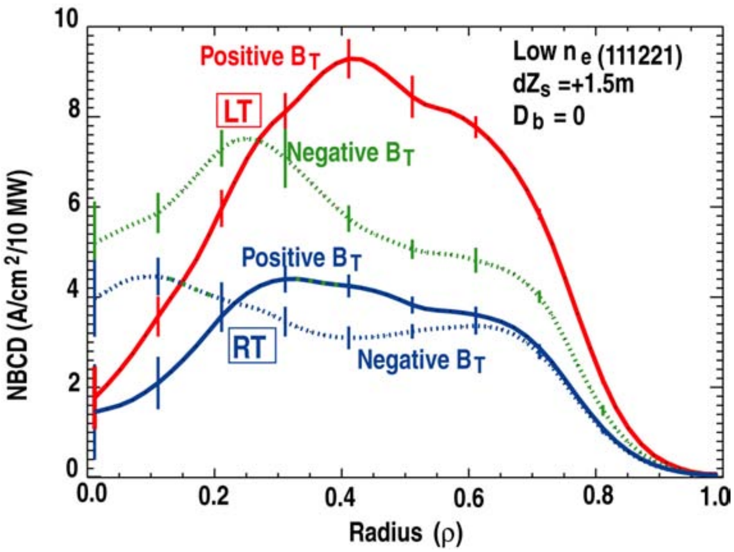

Fig. 5-18. NBCD profiles for the left and right beams with both positive and negative B_T directions [Murakami 2007].

The anomalous NBCD effects to be studied center around the effect of plasma fluctuations on fast ion orbits. This issue is related to the question of the orbits of alpha particles in a burning plasma and the determination of the resulting heating profile. Previous experiments in DIII-D have seen clear evidence that energetic particle instabilities can broaden, or even make hollow, the NBCD profile [Wong 2004]. In addition, experiments to control the evolution of the current profile during the current ramp up phase of Advanced Tokamak discharges find that the NBCD profile is much broader than predicted [Ferron 2006], which may also be associated with energetic particle instabilities. Experiments in DIII-D over the 2009–2013 time frame need to achieve a thorough understanding of the effect of plasma fluctuations on the fast ion density profile and resulting NBCD profile, including the development of a theoretical model that can explain the fast ion transport. This research can be an important contribution towards achieving alpha channeling in future burning plasma experiments.

Several new tools for studying neutral beam physics will be available on DIII-D in the next five years, as seen in Table 5-21, making this an opportune time to reinvigorate our efforts on this subject. One newly added tool is the rotated 210 beamline, so that two of the eight neutral beams now inject counter to the (normal) plasma current direction. Changing the geometry of one of the beamlines to allow off-axis deposition will allow the theoretical models to be tested for different beam orbits. The new MSE views associated with the counter NBI also improves our ability to accurately measure the poloidal magnetic flux evolution. An array of interferometers to measure Faraday rotation, and NPAs to probe a narrow region of velocity space, could allow us to measure changes in the NBCD on fast time scales, which is important for studying the effects of specific MHD events. An upgraded fast ion D-alpha (FIDA) diagnostic with additional radial channels will allow more accurate measurements of the fast ion density profile. Better time resolution for the FIDA signals would allow us to study the effects of specific MHD events. Other upgrades to the fluctuation diagnostics will better allow us to characterize the mechanisms by which fast ions may be transported anomalously. The experimental data will be compared with theoretical models of NBCD, such as those contained in TRANSP and CRONOS, which include ion orbit and transport effects.

Table 5-21
New Tools and Diagnostics for Neutral Beam Physics

Issue	Tool/Diagnostic	Purpose
Current drive physics	Co/Counter NBI	Allows more accurate measurement of NBCD profile by comparing co- and counter-injection beams because systematic uncertainties in currents (like bootstrap) and plasma profiles (like Z_{eff}) will cancel out in the analysis
	On-axis/Off-axis NBI	Directly test theoretical models for off-axis NBCD, and increase sensitivity to transport effects on current drive profile
	Co and Counter MSE views	Better discrimination between E_{rad} and pitch angle measurements
	Faraday rotation interferometers	Changes on very fast time scales (*i.e.*, MHD)
Fast ion physics	Upgraded FIDA	More accurate measurements of the ion density profile for comparison with models
	NPAs	Probe a narrow region of velocity space on a fast time scale
	Upgraded fluctuation diagnostics	Characterize mechanisms by which fast ions are anomalously transported

5.5.4. Science Issues for Bootstrap Current

The basic theory of the bootstrap current is an integral part of neoclassical transport theory. This current is, in essence, a diamagnetic current associated with the toroidal projection of the finite width of the banana orbits of trapped particles. These transfer momentum to passing particles, amplifying the pressure gradient driven toroidal current. Past experiments have shown agreement between the calculated and measured total bootstrap currents in tokamak plasmas. However, there are few local measurements of the bootstrap current density with sufficient precision to test neoclassical predictions. An example of one such comparison is shown in Fig. 5-19. Working in plasmas with high overall bootstrap fraction, and using the ability provided by the DIII-D ECH system to locally modify the electron pressure gradient, it should be possible to measure the local bootstrap current with sufficient precision to test the theory over a larger range of conditions.

Bootstrap studies in DIII-D are expected to concentrate in three main areas during FY09–FY13:

1. AT improvement and alignment studies
2. Behavior of the pedestal bootstrap behavior and its relationship to stability
3. Tests of theory in the regimes where it is most suspect.

The bootstrap current is a key feature of AT plasmas because of the necessity of steady-state operation. The bootstrap current is also a significant factor in the physics of other important tokamak plasma phenomena. For example it is a key parameter in determining the total current density in the pedestal region, and therefore in setting the characteristics of ELMs. Also it is the feature that allows the existence of neoclassical tearing modes, which would be stable but for the modification of the bootstrap current by the island structure. Understanding and verifying the physics of the bootstrap current thus is a significant part of the overall effort to produce advanced tokamak plasma scenarios.

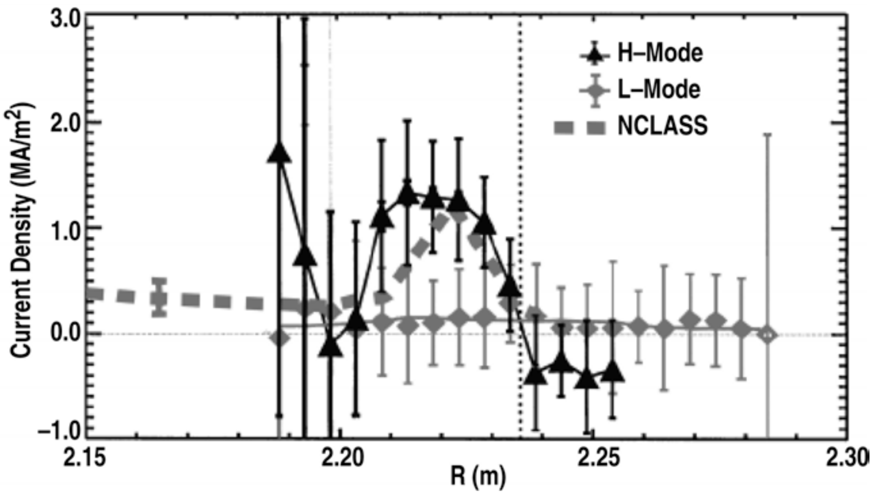

Fig. 5-19. Edge current density determined from LIB pitch angles, compared to the neoclassically predicted value for H-modes [Thomas 2004].

One region of the tokamak plasma where the bootstrap current is modified from standard neoclassical theory is in the steep gradient region of the H-mode pedestal. Here the assumption that the drift orbit width is small compared to the pressure gradient scale length breaks down, at least for ions. Thus the electron terms in the bootstrap calculation should still be valid, but the ion contributions need to be modified. The calculation of currents in this region is further complicated by the presence of a loss region in the velocity space of the ions – certain orbits are lost to the wall – leading to a distortion of the ion velocity distribution. In this region and in the SOL just outside the separatrix, the calculation of the bootstrap current is thus strongly coupled to the calculation of plasma flows, and to the viscous transfer of momentum across flux surfaces. Both theory and creative experiments are needed to elucidate these processes. The LIB diagnostic and upgraded MSE system now allow us to make detailed localized measurements of the poloidal magnetic field across the plasma. Using the appropriate kinetic magnetic equilibria, these measurements give information on the distribution of pressure driven (Pfirsh-Shlüter and bootstrap) currents. We plan to extend the use of the LIB diagnostic for edge studies by improvements to beam performance and detector upgrades, as well as a possible upgrade to perform main ion measurements using this device (DOE Diagnostic Grant Application). These localized measurements will allow us to test the effects of collisionality and pedestal impurity content, and temperature on the various terms.

Further work is needed on theory and experiment in other regions of the plasma where the standard neoclassical theory breaks down. Near the magnetic axis, the radial extent of the orbits of trapped particles is no longer small compared to the minor radius, and in fact there are the so-called potato orbits, which are trapped particle orbits that encircle the axis. The presence of potato orbits modifies the neoclassical theory and allows a finite bootstrap current at the magnetic axis. Verification of this effect would enhance our ability to project tokamak physics to high bootstrap fraction reactor scenarios. Perturbative tests using improved ECH power will allow us to isolate the bootstrap current by studying the modulation (relaxation) of the current density at different locations.

5.6. BOUNDARY PHYSICS

Controlling the interaction of heat and particles emanating from the core plasma with material surfaces has long been identified as a critical issue in the successful demonstration of fusion energy production [ITER 1999]. Because of the extremely large variations in plasma properties across this interface region, developing a first principles understanding of all the processes that determine the degree of heat and particle control one can reasonably achieve is extremely challenging, both from an experimental and theoretical point of view. In response to this challenge, the DIII-D program has identified two major goals for boundary research during the period 2009–2013:

1. Develop the capability to predict and control divertor heat flux in conditions consistent with Advanced Tokamak operation
2. Provide the physics basis for carbon as a plasma facing component in ITER

To accomplish the first goal, it is envisioned that two parallel research paths will be utilized with coordination of the activities increasing as the five-year period progresses. The first path will focus on providing an improved physics basis for the *prediction* of divertor heat flux, with particular emphasis placed on those issues that will have the largest impact on the heat flux profile. Examples include the characterization of SOL turbulence and its impact of heat flux widths, the origin of SOL flows and their effect on particle control, and the dynamics of divertor detachment. The research program for this line of research is discussed later in this section. The second path will emphasize the testing and development of advanced heat flux *control* techniques consistent with Advanced Tokamak operation. The major elements of this program are discussed in Section 3.7. In support of the second goal, DIII-D will take advantage of its all-carbon first wall and unique material sampling capabilities to quantify hydrogenic retention with co-deposited carbon and develop means to mitigate such retention. The plans in this area are discussed extensively in Section 2.8. In addition to these high-level program goals, it is envisioned that the DIII-D program will continue to carry out important research across a broad range of topics, albeit at a lower priority. Of particular note are studies aimed at improving our understanding of plasma-wall interactions utilizing divertor and main chamber material sampling diagnostics.

An overview of the boundary physics plan for the period 2009–2013 is presented in Table 5-22. Included in the table are program elements discussed in more detail in Sections 2.8 and 3.7. Also included are the goals of these studies, the important physics challenges, including the physics based scaling to be addressed as a well as points of difficulty that need to be overcome. Proposed new hardware and diagnostic tools that will help address these issues are also summarized. The proposed research is aimed at the following issues identified by the FESAC Priorities Report [FESAC 2005]:

T5. How are electromagnetic fields and mass flows generated in plasmas?

T10. How can a 100 million degree burning plasma be interfaced to its room temperature surroundings?

T13. How does the challenging fusion environment affect plasma chamber systems?

Table 5-22
An Overview of the Boundary Issues to be Addressed During the Five Year Period 2009–2013

Objective	Physics Challenges	New Tools
Predicting Divertor Heat Flux		
Correlate scaling of heat flux width λ_q in SOL and divertor (Section 5.6.1)	1. Relation of λ_q in near SOL to λ_q in divertor 2. Relation of λ_q in near SOL to turbulent transport 3. Variation with operating modes	1. IR + Visible TV periscopes 2. Gas Puff Imaging 3. In tile calorimetry 4. Divertor Bolometry 5. Midplane MACH + turbulence probes, HFS and LFS. 6. T_i from gridded energy analyzer
Improved understanding of turbulent radial transport (Section 5.6.2)	1. Relation of parametric scaling to underlying turbulence drives. 2. Role of divertor detachment	1. Midplane MACH + turbulence probes 2. Gas Puff Imaging 3. IR + Visible TV periscopes
Identify mechanisms responsible for SOL flows (Section 5.6.2)	1. Poloidally resolved measurements of turbulent transport and flow 2. Role of cross field drifts	1. Gas Puff Imaging 2. Midplane MACH + turbulence probes
Controlling Divertor Heat Flux		
Develop techniques for heat flux control in Advanced Tokamak regime (Section 3.7)	1. Heat flux dispersal in a low density edge plasma 2. Compatibility of edge solution with core requirements	1. IR + Visible TV periscopes 2. Divertor flux expansion coils 3. Inner wall RMP coils
Basis for Carbon in ITER		
Test whether operation at high wall temperatures can reduce global hydrogenic uptake (Section 2.8)	1. Choice of optimum temperature to balance competing surface processes 2. Note—many engineering issues to address	1. IR spectroscopy of residual gas analysis 2. In-situ analysis of tile coatings 3. Water/air heating systems
Evaluate efficacy of O_2bake in removing co-deposited deuterium (Section 2.8)	1. Choice of target bake temperature and O_2 partial pressure to achieve adequate removal while preserving the integrity of in-vessel components	1. IR spectroscopy of residual gas analysis 2. In-situ analysis of tile coatings
Plasma Material Interactions		
Identify and characterize processes that cause unacceptable erosion in main chamber (Sections 2.8, 5.6.3)	1. Low net erosion rates require long exposure times 2. Impulsive events (Blobs, ELMs, disruptions) may dominate 3. Large toroidal asymmetries likely	1. Advanced Midplane Probe/Movable Limiter/Surface Station 2. IR + Visible TV periscopes 3. B_4C and/or SiC coatings on main chamber tiles
Identify primary dust sources and dust transport mechanisms	1. Improved efficiency of observation of dust from 30 nm to 10 micron radius. 2. Relation of video images of glowing dust to dust size	1. IR + Visible TV periscopes 2. More fast video cameras and improved video analysis
Develop detailed documentation of the key PMI processes in a graphite dominated divertor (Sections 2.8, 5.6.3)	1. Improved high resolution spectroscopy for neutral and molecular line/band shape studies 2. Accurate molecular and atomic photon yield coefficients 3. In-situ surface analysis	1. New MDS camera 2. Improved Porous Plus Injector 3. In-situ surface analysis

5.6.1. Scaling of Heat Flux in SOL and Divertor

Motivation and Status: The possibility of destructively high heat loads on the divertor surfaces of high-powered future tokamaks is a major concern. To predict the heat loading for such devices with confidence, one needs to understand the physics behind the heat flux distribution at the divertor targets, as characterized by the power scrape-off width λ_q at its divertor targets [ITER 1999]. Power deposition at the divertor targets is determined by the magnetic geometry, the competition between radial transport and perpendicular transport along the scrape-off layer field lines, and energy dissipation in the SOL by radiation or other dissipative processes. Unfortunately, the extrapolation of λ_q from present-day tokamaks to future devices is presently uncertain, largely because the underlying physics of the perpendicular transport is not well understood. As a result, the survivability of a DEMO-class tokamak divertor cannot be assured. The updated ITER Physics Basis [ITER 2007] has recognized the uncertainty in extrapolating λ_q to larger, more powerful tokamaks and the necessity in resolving this issue in a definitive way: "All that can be strongly concluded (from the present scalings for λ_q) is that there is a need for improved experimental measurements and a theory-oriented approach for making extrapolations for the target heat flux in ITER."

There is a strong impetus, then, to improve the understanding of the physics underlying the behavior of λ_q for higher power tokamaks. While many tokamaks around the world such as ASDEX-U [Herrmann 2002], JET [Eich 2005], and JT-60U [Loarte 1999] are contributing this research, the DIII-D program is well positioned to address issues related to λ_q. As we discuss below, DIII-D has an impressive capability to simultaneously achieve the shaping, heating power, density control, and control over inimical MHD activity to successfully achieve and then maintain the operating regimes and collisionalities of interest in studying λ_q. The strong boundary diagnostic systems, bolstered by the new initiatives discussed in this document, will provide key data necessary for benchmarking the edge turbulence codes such as BOUT, TEMPEST, and XGC, and the 2D transport models including UEDGE and SOLPS5.

Research Plans for 2009–2013. The scaling of λ_q with parameters, such as plasma current and input power, can be very useful from a tokamak engineering and design standpoint. As such, data from these scaling studies provide a valuable steppingstone for extrapolating divertor heat flux behavior to next generation, highly powered tokamaks. However, the underlying physics of what governs λ_q behavior is very difficult, if not impossible, to extract using this approach. Thus, in our power width study, we will also focus on specific experiments that directly address λ_q behavior on a more fundamental level.

Experiments that examine the link between turbulent transport in the SOL and λ_q provide an appropriate place to begin this study. In particular, we determine if there is a correlation between increased turbulence in the SOL plasma and increased λ_q. These results will be compared with the predictions of the BOUT code, which has been used to compute edge and SOL transport. DIII-D has both the plasma shaping capability and the diagnostics to make the key measurements for this type of experiment and subsequent analysis. In the region of the divertor, dissipative processes can play a key role as the power from the plasma flows along the field lines to the divertor. Heat flux dissipation by radiation and charge exchange can provide a significant reduction in the peak power loading on divertor components.

Identifying the processes responsible for the observed λ_q behavior will allow prediction of λ_q based on the extrapolated physics identified from this "first principles" study. Comparison of these predictions to λ_q measured on other tokamak, *e.g.* C-Mod and NSTX, allows a verification of the model. Once a consistent picture is produced, the uncertainty that presently exists in predicting λ_q for the ITER device and post-ITER devices would be significantly reduced. A second very important consequence is this: Knowledge of what drives the cross-field transport in the SOL also presents the possibility of actively manipulating cross-field transport in a way that would increase λ_q.

Hardware and Diagnostic Requirements: The DIII-D vessel is favorably configured to conduct λ_q-scaling experiments. A large flat surface on top of the lower outer baffle extends 40 cm radially outward. Key diagnostics that are needed for successful analysis of λ_q, such as infrared thermography to determine heat loading and D_α measurements to determine recycling activity, have an unobstructed view of the *entire* scrape-off region adjacent to the outer divertor strike point. Other measurements at or near the divertor strike point also benefit from this broad, flat "test bed" by making the interpretation of the data straightforward and unambiguous. These would include radial arrays of thermocouples covering several toroidal locations, providing an independent way of determining heat deposition on the baffle shelf roof. Radial profiles of the electron density and temperature along the baffle roof will be determined by Langmuir probe measurements. Electron density and temperature measurements in the divertor volume are made by an already existing Thomson Scattering diagnostic.

DIII-D has the capability to achieve operating modes of interest to ITER during λ_q studies. DIII-D routinely operates in confinement and stability regimes that are relevant to future tokamak operation, including "hybrid" and *AT* H-mode scenarios (*e.g.*, $H_{89P} \geq 2$ and $\beta_N > 3$). Moreover, large variations in scrape-off layer and divertor densities (and collisionalities) are possible in DIII-D by using three cryopumps, each of which can be operated independently of each other. Combined with plasma heating power up to 30 MW, λ_q behavior can be studied over a wide range in collisionality.

Unique to DIII-D will be the capability to study λ_q behavior in high-energy confinement regimes under resonant magnetic perturbations (RMP) and divertor poloidal flux expansion operating modes, discussed in detail in Section 3.6. Both operating modes are of interest because they show promise in being able to reduce the peak heat flux at the divertor targets without significantly perturbing the core plasma. The RMP approach is related to ITER's attempt at active ELM suppression during H-mode operation. We believe that ELM suppression with the upgraded RMP coil array proposed in the Five Year Proposal (Section 2.2) will also provide peak heat flux reduction at the divertor targets "for free" by increasing the effective wetted area in the divertor. Because we expect that the behaviors of both particle recycling and heat flux at the divertor targets during active ELM-suppression are inherently three-dimensional, the openness of the lower outer divertor test bed is ideally suited for unobstructed viewing of the recycling and heating patterns in both radial and toroidal directions. Thus, coupled with the 3D IR and Visible TV periscopes planned for in the Five Year Proposal, DIII-D will have the full capability to study systematic changes in the heat flux profiles at the divertor target(s) when the RMP coil is activated under different operating scenarios. DIII-D also plans to have the capability to study how the heat flux profile at the divertor target is affected by changing the poloidal flux expansion at the outer lower divertor target (Section 3.6).

DIII-D has the capability to expand its operating window by controlling deleterious MHD during high power operation. It is also important to note that not all of the available heating power can necessarily be used to study λ_q. Operation at high power input can result in plasma susceptibility to MHD-related instabilities, degrading confinement and leading to possible plasma disruption. However, DIII-D has demonstrated the capability to suppress or limit the growth of several classes of plasma instabilities that could otherwise limit the power level to well under the available heating power. DIII-D's expertise in MHD feedback control for high power plasmas is important because it can expand the window for high power operation and can open up operating regimes that might not otherwise be reachable.

5.6.2. SOL Flows and Turbulence Driven Transport

Motivation and Status: Experiments in recent years worldwide have identified intermittent, "blob"-like turbulent structures in the scrape-off-layer (SOL) as a potential mechanism for cross-field transport in this region [Zweben 2002, Boedo 2003, Zweben 2004]. Over the same timescale, separate experiments have measured anomalously high particle flows in the SOL [LaBombard 2004, Erents 2004, Asakura 2004]. Early results from DIII-D using fast-stroke Langmuir probes showed the importance of ExB poloidal flows in the divertor, which can carry up to 50% of the particle inventory in the divertor [Boedo 2000]. In addition, toroidal flows measured by these same probes are anomalously high in the plasma crown. Recent measurements of the poloidal distribution of edge plasma turbulence, concomitantly with results from Alcator C-Mod, have uncovered significant poloidal asymmetries in the turbulence level. These results taken together suggests a potential link between SOL turbulence and SOL flow with the ballooning character of the turbulent radial transport creating strong pressure asymmetries that cause high parallel flows in the SOL/edge. Representative measurements from DIII-D of the SOL turbulence and resulting transport at the outboard midplane, where turbulence is expected to be large, are shown in Fig. 5-20 [Rudakov 2002]. Experiments where methane is puffed in a symmetric manner into DIII-D and then the ionized byproducts followed via 2D imaging, provide additional support to the probe measurements of the high Mach flow in the SOL [Groth 2007]. Such poloidal asymmetries are seen in the nonlinear fluid model BOUT in the saturated turbulent transport. Interpretative simulations of these results support the asymmetric transport paradigm within reasonable assumptions.

Research Plans for 2009–2013: SOL turbulence studies will focus on identifying and characterizing the mechanisms responsible for SOL transport with particular emphasis placed on comparing experimental results with theoretical predictions of SOL turbulence codes such as BOUT and TEMPEST. Additional efforts will examine the scaling of the turbulence driven radial flux of particles, heat and momentum with both global and local parameters. In the area of SOL flows, DIII-D research will aim to characterize the poloidal asymmetries of flows and their primary drive mechanisms, including neutral ionization, local plasma pressure perturbations, and poloidal drift terms. The connection between SOL turbulence and SOL flows will also be explored taking advantage of the capability to measure radial profiles at multiple poloidal locations.

Hardware and Diagnostic Requirements: The increased level of complexity required to address these issues will benefit from an increase in the coverage of 2D diagnostics. However, there are several measurement capabilities that must be developed to make significant progress. Momentum transfer, especially the anomalous channel, needs to be measured since it drives spontaneous rotation that is thought to be important in the L-H transition. Progress can be enhanced by the addition of gas puff imaging to our existing plunging probe and beam emission spectroscopy. Core and SOL ion fueling resulting from ionization of neutrals plays a significant role in SOL plasma flow. The radial and poloidal distributions of neutral density and ionization source term are not well measured; thus a neutral diagnostic should be emphasized. Unfortunately there are no known diagnostic techniques for the effective 2D imaging of the neutral density over the required dynamic range, roughly three orders of magnitude. Presently we use D_α recycling light and edge neutral pressure measurements to benchmark 2D Monte Carlo models of neutral transport. Clearly this is a measurement waiting for an advancement in instrumentation.

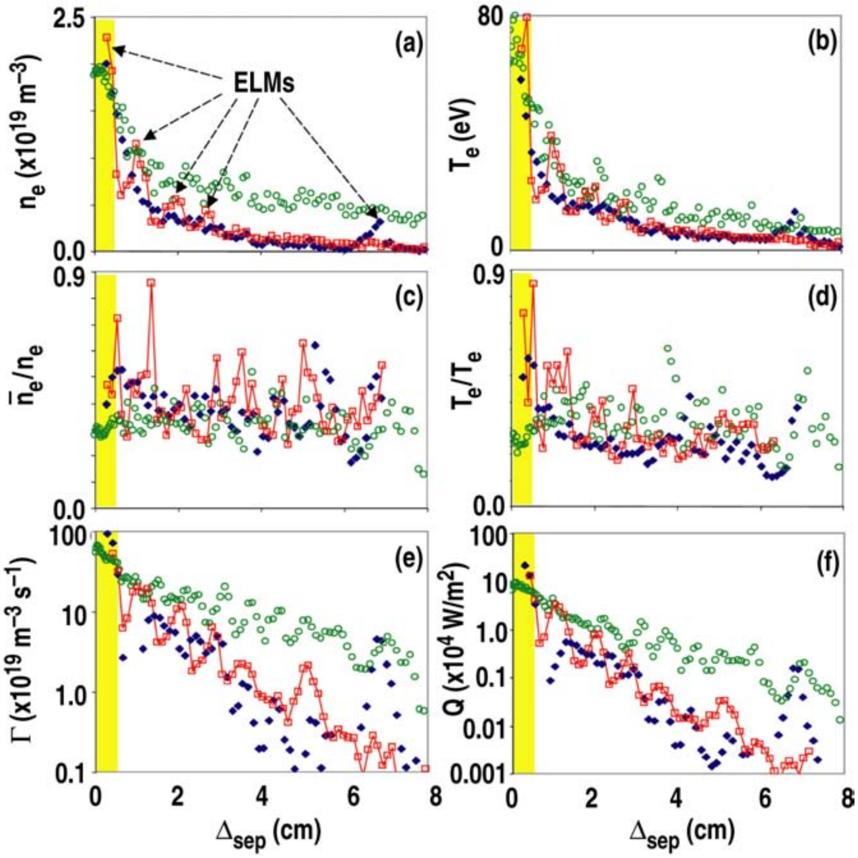

Fig. 5-20. Radial profiles of time-averaged (over 1 ms) electron density (a), electron temperature (b), their relative fluctuation levels (c,d), turbulent particle (e) and heat (f) fluxes in L-mode (open circles) slowly ELMing H-mode (solid diamonds) and rapidly ELMing H-mode (open squares + line). Shaded areas represent estimated error in defining the probe position versus the separatrix. Note semi-log scale in (e,f). [Rudakov 2002]

Additionally, little is known with regard to the ion channel in the SOL although it is clear that $T_i > T_e$ in many, if not most DIII-D plasma regimes. A measure of the ion temperature in the divertor and SOL would be very valuable. We propose to respond to these needs with the following diagnostic initiatives:

- **Ion temperature** By measuring the ion temperature in the SOL, one addresses the question of heat transport on the ion channel. This diagnostic is also useful to analyze the ion channel contribution to transient events such as ELMs. The proposed diagnostic is a retarding field analyzer that can be located in the divertor floor and in the x-point and midplane fast stroke probes..

- **Turbulence** Deploy a 2D "array" of probes at multiple poloidal and toroidal angles to get better measurements of poloidal and toroidal asymmetries and wavenumbers. Deploy gas puff imaging to measure 2-D distributions of turbulence and poloidally extended variations and structures.
- **Neutrals** Increased coverage of edge neutral pressure measurements. Deploy innovative neutral measurement schemes as they become available.
- **Flow** A new pop-up mach probe on the centerpost is proposed to document inner SOL flow. This probe should also be equipped to measure the local electric field, as a measure of the $E \times B$ flows and local turbulence driven radial transport. The addition of a new high-resolution spectrometer with tangential views in the upper divertor to measure carbon ion flows is also proposed.
- **Synthetic/virtual diagnostics** Synthetic Langmuir probes and fast camera imaging within the simulation codes for a direct comparison to the data from these diagnostics.

5.6.3. Plasma-Wall Interaction and Dust Studies

Motivation and Status: Controlling plasma-material interaction (PMI) is a critical issue for future fusion devices [Federici 2001]. While plasma interaction with the main chamber wall should be minimized to the extent possible in order to prevent damage to the first wall elements and core plasma contamination with impurities, such interactions will invariably occur. The unique capability to insert removable material samples in DIII-D using the Divertor Material Evaluation System (DiMES) [Wong 1992] has provided key insights into the erosion and redeposition of several candidate materials for plasma facing components. Using this system, the erosion rates of carbon and tungsten have been measured over a wide range of plasma conditions ranging from L-mode to detached H-mode operation [Whyte 2005]. The DiMES apparatus has also been used to deploy a novel Porous Plug Injector that can simulate chemical sputtering in the divertor, allowing determination of key atomic rates for divertor modeling [McLean 2007].

In an idealized picture of divertor tokamak operation, plasma particles cross the last closed flux surface (LCFS) into the SOL and flow along the open field lines into the divertor volume, where most of the plasma–material interactions occur. However, if the SOL width is comparable to the distance between LCFS and the wall, significant plasma interaction with the main wall elements may also occur. Experiments on DIII-D and elsewhere have shown that such material interactions do occur in the main chamber. Because these surfaces are much closer to the core plasma, the impurities generated by these interactions have a significantly higher probability of entering the core plasma than impurities generated in the divertor region. Quantifying the amount of impurity generation at the midplane relative to the divertor has been extremely limited to date to lack of adequate diagnostics.

Improved diagnostic capabilities (e.g., fast imaging camera, Mie scattering) have enabled initial DIII-D experiments assessing dust generation and transport [Bray 2006]. These studies indicate that while dust is indeed present in DIII-D, the level of dust is generally very low during normal plasma operations. However, off-normal events such as disruptions are observed to generate large levels of dust. In a similar manner, dust is more readily observable in discharges with large type I ELMs than similar discharges without ELMs. Images with the fast camera indicate that dust is transported along field lines and easily reaches the midplane from a localized source at the divertor. These observations are consistent with DustT modeling.

Diagnostic Plans for 2009–2013: Proposed PMI research in support of ITER, discussed in Section 2.8, is focused on erosion and redeposition/codeposition studies. In this section, we will discuss the diagnostic initiatives related to the PMI studies. These initiatives fall into two categories:

1. Expansion of existing first wall diagnostic systems to better define the plasma-wall boundary condition
2. New diagnostic initiatives that have the potential to provide a quantum leap in the efficiency of PFM studies on DIII-D.

During this period, we will maintain and upgrade the existing systems and will add some new capabilities for PMI studies. These include spectroscopy systems (MDS, filterscopes, *etc*.), cameras (tangential TVs, DiMES TV) and fixed Langmuir probes mounted in the divertor tiles. The lower divertor additionally features the Divertor Material Evaluation System (DiMES), allowing exposure of material samples with in-situ temperature control. We also plan on installing fast cameras operating in both visible and IR parts of the spectrum.

Diagnostics for PMI studies at the main chamber wall are at present limited and will be significantly improved. Until recently, only a fast mid-plane reciprocating probe was available for studies of the plasma interaction with the outboard chamber wall. In 2006 two filterscope channels and three camera views of the outboard wall were added. A fast framing camera proved to be particularly useful for diagnostics of the PMI caused by transient phenomena such as blobs and ELMs. However, additional fast camera views are needed to resolve the complicated interaction patterns that are neither poloidally nor toroidally symmetric. During the period of 2009 through 2013 we will provide additional fast visible and IR camera views of the outer wall and centerpost. We will also install arrays of fixed Langmuir probes (including some calorimeter probes) at the outboard wall, bumper limiters, divertor baffles and the centerpost to monitor the incident plasma fluxes.

To provide significantly improved characterization of the plasma interaction with the outer main chamber wall, we propose to install an advanced main chamber surface station that will incorporate a fast stroking probe, an instrumented materials testing station that is extractable between shots, and a movable bumper limiter, also well instrumented for the measurement of plasma heat and particle flux. This module can be used for tests of new advanced PFMs and reactor-relevant main chamber PFCs. As a first step in this direction, the Midplane Material Evaluation Station (MiMES) was installed in 2007. A schematic drawing of the system installed on DIII-D is shown in Fig. 5-21. It allows exposures of material samples and diagnostic mirrors at the outboard wall. However, in its present version MiMES does not allow use of instrumented samples and/or in-situ temperature control. The new advanced system will most likely occupy an equatorial plan port, but other outer wall ports may be satisfactory. The fast stroke probe will occupy the central region of the port, and will retain its capability to have an adjustable scan extendable to inside the typical last closed flux surface. Surrounding the probe will be a movable bumper limiter that will also serve as an instrumented sample exposure station. The limiter motion will be slow, its location will be adjusted only between shots. Instrumentation capability will be added to MiMES and will be upgraded for the probe to allow advanced functions, *e.g*., ion mass and energy analysis. Both the probe head and MiMES will be extractable behind a vacuum interlock for exchange of probes and exposure samples.

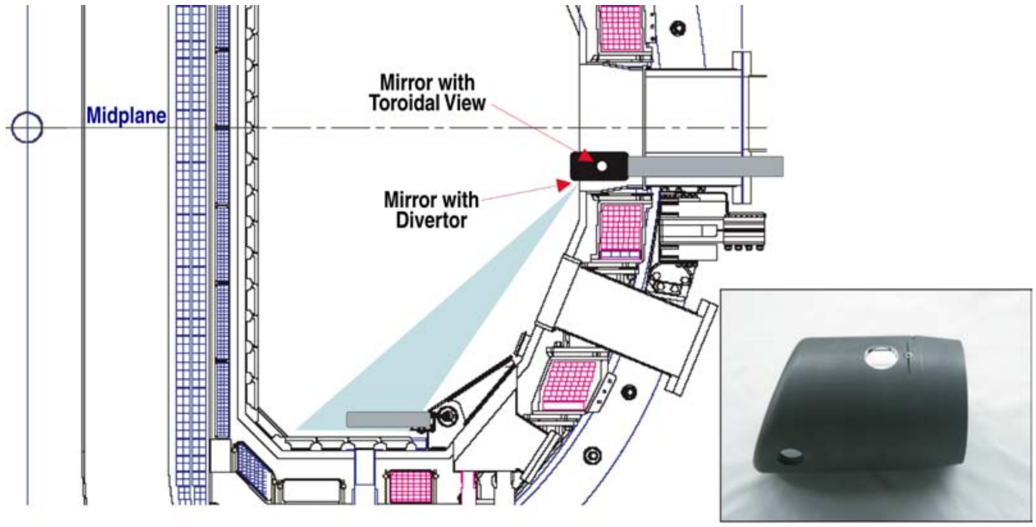

Fig. 5-21. A schematic of the MiMES probe and a photo of the MiMES shroud containing prototype ITER mirrors that were installed and exposed during the 2007 campaign.

For the upcoming period we will be looking into installing diagnostics allowing in-situ real-time or in-between shot measurements of erosion/deposition. As discussed in Section 2.8, present studies of erosion and redeposition rely entirely on the ex-situ sample analysis. The candidate techniques (for particular measurements) include colorimetry (deposition thickness), in-situ ellipsometry (deposition thickness), laser induced breakdown spectroscopy (surface composition), ARRIBA (deposition thickness, composition), micro-profilometry (erosion/deposition), and Quartz Microbalances. A related proposed diagnostic uses Cavity Ringdown Spectroscopy to measure the hydrocarbon content in the divertor pumping plenum. A possibility of developing an articulated arm manipulator to install/remove material samples and PMI diagnostics inside the vessel between shots will be investigated. The development of an articulating arm for use in PMI research has the potential to make an enormous impact on the efficiency of first wall research on DIII-D and on future devices such as EAST and KSTAR.

Even though DIII-D will continue to focus on graphite as a primary plasma-facing material (Section 2.8), a possibility of installing individual tiles or sections of tiles made of other ITER/reactor relevant materials in the divertor and/or at the main wall will remain open. Such tests can provide valuable information on PMI processes in mixed material environment including carbon.

A major upgrade to the measurement of surface heat flux using a fast, high resolution IRTV combined with a tangentially viewing periscope is the most critical of the boundary diagnostic initiatives we are proposing. A set of three of these systems would provide almost full coverage of the heat flux to the plasma facing surfaces at time resolution sufficient to resolve ELMs. It would also greatly expand our knowledge of the 3D transport of energy to the walls due to the resonant magnetic perturbations used for ELM suppression.

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6. DEVELOPMENT AND VALIDATION OF INTEGRATED MODELS

The goal of integrated modeling is to develop and validate the capability to understand and predict the behavior of tokamak discharges. Such a model encapsulates what we really know about the plasma physics of tokamak discharges. The validated integrated model is an enduring contribution of the DIII-D program that can be applied to present issues and future toroidal fusion devices.

This goal of having a validated model of tokamak behavior is much broader than the DIII-D program. Nearly all fusion laboratories have relevant work ongoing, and there are inter-institutional organizations like the Fusion Simulation Project and SciDAC and SWIM that have similar goals. The more specific goals of the integrated modeling to be carried out within the DIII-D program and its close collaborations in the next five years emphasize advances on pedestal transport and stability; plasma boundary and scrapeoff layer; and energetic particle instabilities. These topics are cited in many studies as the key uncertainties in proceeding to a burning plasma experiment. They are also topics for which the flexibility of the DIII-D discharge parameters and heating systems and the outstanding diagnostic set provide a unique capability for comprehensive and conclusive research.

Development and validation of an integrated predictive model of tokamak behavior is important because such a model can be used to interpret and understand present experiments, to guide and optimize future experiments, and to support the design of new devices like ITER. Integrated models are based on simpler component models which should be validated through comparisons with experiment. Some of the component models which are presently being validated are shown in Table 6-1. This table also identifies some of the key needs for development or validation of the component models in the topical areas of equilibrium, sources and actuators, transport, MHD stability and control, pedestal, and scrape-off layer (SOL). Some of these component models are being assembled into more universal tools for integrated modeling through work in the DIII-D program and in other laboratories. Some areas, like equilibrium and core sources, are close to the level of development needed, while other areas, like the SOL, are too complicated for inclusion in an integrated model at the present level of understanding. Validation of an integrated model is a very difficult but necessary process which may start during the contract period, as the model advances. Extension of the modeling from the plasma core, which is the present status, to include the edge pedestal is expected to be started through implementation of the TGLF transport model in a transport code. Extension of the model to include the entire duration of the discharge, including the initiation phase, will be carried out by developing reduced models, particularly for transport and stability, which can run more quickly than the full models. Extension to the entire region of the tokamak, including the edge and SOL, will be started. As the model develops, it will be incorporated into the realtime control system for optimizing plasma performance and avoiding instabilities. A very approximate schedule is shown in Fig. 6-1.

In this chapter the status and plan for development of state-of-the-art computational models for the individual components of an integrated model and their integration are described in Section 6.1. The plan for validation of those models is described in Section 6.2. Section 6.3 describes the plan for validation of the integrated model of discharge behavior, and Section 6.4 provides a view of how the models will be implemented and made available to staff.

Table 6-1
Development and Validation of Component Models

Model	Plans/Needs
• Equilibrium	
— Axisymmetric (EFIT)	— Modernize — Develop weekly nonaxisymmetric
• Sources/actuators	
— NBCD (NUBEAM) and bootstrap	— Test with co-, counter-, on-, off-axis NBI — Effect of AE modes
— ECCD (TORAY-GA)	— Test far off-axis
— FWCD (GENRAY, CQL3D, AORSA, ORBIT-RF)	— Fokker-Planck multiple species
— Particles (DEGAS, GTNEUT)	— Apply GTNEUT and compare to observations
• Transport	
— Fluctuation levels and profiles (GYRO, TGLF)	— Reconcile measured and calculated diffusivities
• MHD stability and control	
— MARS-F, . . .	— Plasma rotation and error field effects
— DCON	— Real-time stability calculations
• Pedestal	
— Peeling/ballooning stability (ELITE)	— Predict pedestal width
— Stochastic edge (M3D, NIMROD)	— ELM control
• Scrapeoff layer	
— UEDGE, SOLPS, . . .	— Flows — Plasma/wall interaction — Detached/radiative divertor

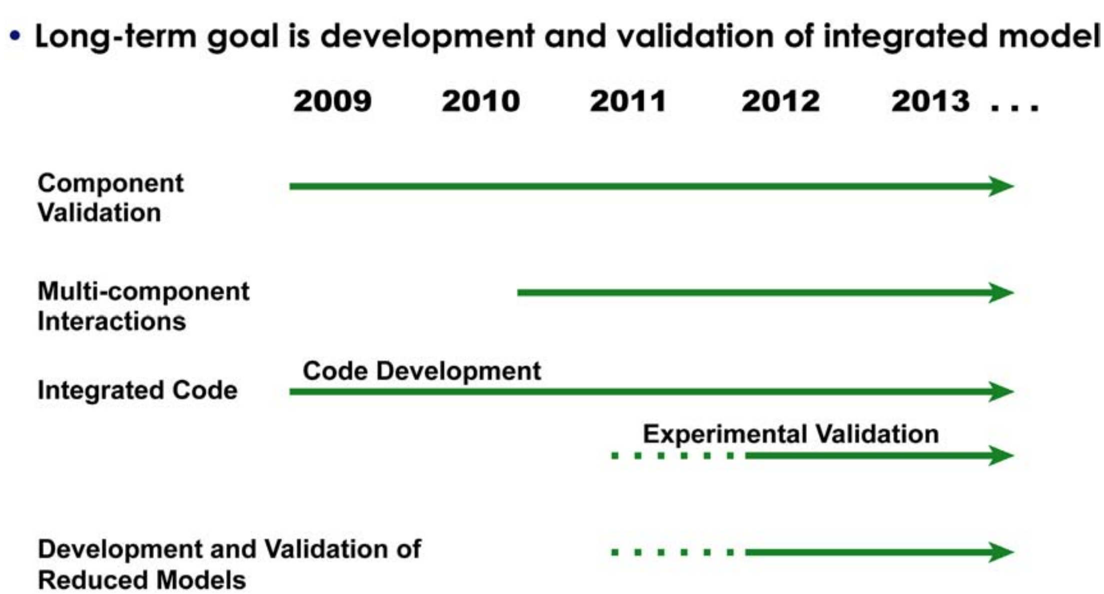


Fig. 6-1. Near term plan focuses on component validation and code development.

6.1. DEVELOPMENT OF STATE-OF-THE-ART COMPUTATIONAL MODELS

Integrated modeling is an important element of tokamak fusion research essential to the interpretation and planning of experiments, validation of theory against experimental results, and the design and construction of next step devices such as ITER and the Fusion Demonstration Facility (FDF). Many important aspects of tokamak physics involve physical processes that interact strongly. These include edge localized modes (ELMs), resistive wall modes (RWMs), plasma disruptions and their mitigation, and sawtooth oscillations and Alfven eigenmodes (AEs) and their interactions with energetic particles. Additionally, an important component of tokamak research, particularly in the DIII-D program, is the simultaneous optimization of various physics elements toward advanced tokamak (AT) operation. Thus, modeling necessarily involves the self-consistent integration of various physics elements from different topical science areas, such as transport, macroscopic equilibrium and stability, and heating and current drive (CD). This section summarizes various improvements and developments of tool and analysis capability and their necessary integration that will be carried out to support various key elements of the DIII-D Five-Year Plan. These developments are also critical to the advancement of the scientific basis for a predictive understanding of physics issues crucial to ITER and FDF. In the following, these elements are addressed:

- Extension of transport modeling towards the edge with new more accurate models such as TGLF to explore pedestal transport physics issues; codes include ONETWO, XPTOR, PTRANSP, and GYRO.
- Integration of transport, edge MHD stability, and boundary and SOL modeling to address edge pedestal, ELM, and boundary physics issues; codes include ELITE, BOUT, UEDGE, NIMROD, M3D, TEMPEST, and XGC-2.
- Development of damping models which include rotation and integration of MHD stability with transport to explore RWM rotational threshold physics; codes include MARS-F, GATO, DCON, and VALEN.
- Integration of rf and NBI heating with transport and MHD stability to address sawtooth and AE mode fast-ion physics issues; codes include ORBIT-RF, CQL3D, TORIC, AORSA, GATO, and NOVA-K.
- Integration of nonlinear 3D MHD stability with impurity transport to address plasma disruptions and mitigation physics issues; codes include EFIT and NIMROD/KPRAD.
- Integration of equilibrium, MHD stability, and transport to address plasma response to nonaxisymmetric perturbation magnetic fields; codes include EFIT, NMA, GATO, and V3FIT.
- Modernization of EFIT computation architecture and extension of its equilibrium reconstruction capability to model 3D equilibrium states due to externally applied perturbation and intrinsic error magnetic fields.
- Development of accurate NBI, rf, and CD source modules to support integrated modeling.
- Development of a modern integrated modeling framework to facilitate coupling among various equilibrium, transport, and stability physics modules for the plasma core, pedestal, and boundary.

The relationship between these code elements and the integrated model is shown in Fig. 6-2.

6.1.1. Extending Core Transport Model Towards the Edge

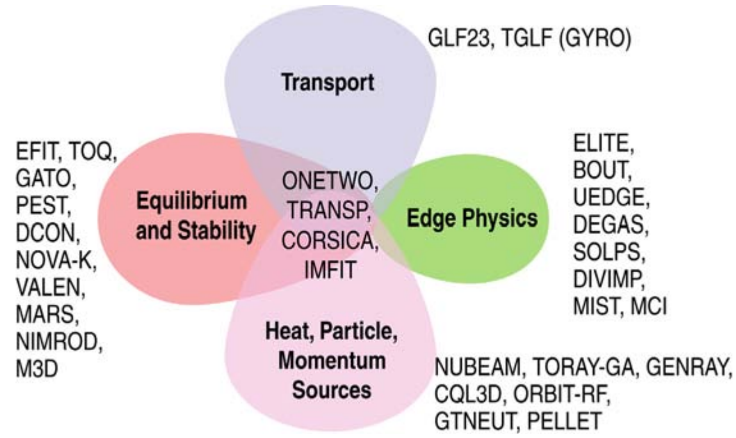


Fig. 6-2. Relationships of code models.

Predicting the H-mode power threshold and pedestal pressure are urgent needs for the next step burning plasma experiments. A transport model valid out to or approaching the separatrix boundary is needed. The drift-wave based GLF23 transport model can predict the core transport quite well in existing experiments. Its main limitation is a lack of real shaped geometry and inaccuracy at large Shafranov shift (α) or negative magnetic shear. The boundary condition for the transport model is usually taken well away from the edge pedestal region. The remedy is a more accurate, and therefore computationally more intensive, gyrofluid model TGLF with comprehensive physics that includes trapping and fully electromagnetic effects and general toroidal geometry [Staebler 2005]. TGLF is valid close to the magnetic separatrix, which is crucial for exploration of pedestal transport physics. Linear TGLF growth rates are still obtained hundreds of times faster than by the GKS code, to 11% accuracy. (The GKS - GYRO agreement is also 11%.) The TGLF fluctuation intensity saturation rule was fit to a database of 86 $S-\alpha$ GYRO flux tube simulations in the GYRO database. The result is a dramatic improvement over GLF23, as shown in Fig. 6-3 [Staebler 2007].

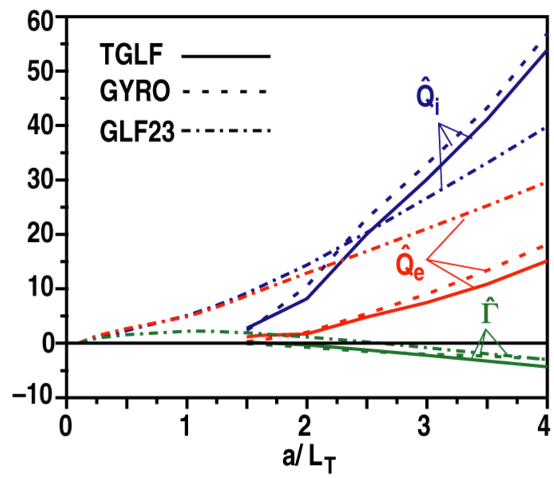


Fig. 6-3. Comparison of TGLF and GLF23 fluxes with GYRO for a temperature gradient scan about the GA-standard case.

To extend the transport modeling capabilities outward toward the plasma edge, the TGLF transport model will be implemented in the ONETWO transport code. The approximate Miller equilibrium model will be replaced by more accurate numerical MHD equilibria from EFIT. Similar to GLF23, the TGLF transport model will require stiff transport solvers like the parallel globally convergent Newton method (GCNMP) solver developed for use with GLF23 in ONETWO. Improvements to GCNMP will be made if necessary to meet the more demanding TGLF physics and computational requirements. Development of parallel algorithms will be essential. TGLF will also require more computational resources.

Benchmarking with the XPTOR code, which will contain the first implementation of TGLF, will be performed, followed by more extensive tests of the model using the GCNMP solver and MPI parallelization. Testing of the integrated modeling simulations of DIII-D discharges will then be performed, and the boundary conditions will be extended outward nearer the edge. After the ELITE edge peeling-ballooning stability code is coupled to ONETWO as described below, testing of the H-mode pedestal region through the ELM cycle will be carried out where the density, temperature and momentum transport will be simultaneously predicted. Simplified ELM crash models will be developed based on nonlinear simulation results from BOUT and NIMROD, and the radial structure of the unstable peeling-ballooning modes will be found from ELITE. Algorithms to relax the current, density, and temperature profiles will then be constructed and tested. For alternate regimes such as QH-mode, simple models of the MHD edge saturation and its resulting transport will be developed.

6.1.2. Edge Pedestal and ELM Modeling

To accurately model the pedestal region, in addition to a comprehensive transport model like TGLF as described above, it is necessary to couple to an edge peeling-ballooning stability code and develop a simplified ELM crash model to relax the current, density, and temperature profiles based on the results of the edge stability calculations. A number of MHD stability codes such as ELITE, GATO, and DCON are available for evaluation of edge peeling-ballooning stability. ELITE, in particular, is a highly efficient code designed specially for the study of the intermediate wavelength peeling-ballooning modes that are often the limiting pedestal instability [Snyder 2002]. Toroidal rotation has recently been implemented into ELITE. To allow a more accurate determination of the ELM onset condition, accurate and consistent models that include diamagnetic effects to replace the simplistic γ versus $\omega_*/2$ arguments will be developed.

Quantitative studies of ELM size, as well as the important issue of heat and particle pulse transport across the open field line region and into material surfaces, will require moving beyond linear studies to 3D nonlinear simulation and supporting theory. The results of 3D nonlinear simulations will also provide essential guidance for the development of a simplified ELM crash model for use with integrated modeling of the pedestal. Several 3D MHD codes are available. These include the 3D reduced-Braginskii BOUT [Xu 2002] code, the NIMROD [Sovinec 2003] code, and the M3D code. BOUT has been modified to include the kink term, allowing treatment of peeling-ballooning modes in the presence of diamagnetic and resistive effects. NIMROD has also included a model of the open field line region as a high resistivity plasma, allowing study of edge instabilities. Two-fluid effects have also recently been added to NIMROD to allow exploration of the importance of diamagnetic effects. The pedestal region plasma, particularly in devices such as ITER, will not generally be highly collisional, so it is important to explore kinetic approaches to edge simulation. A comprehensive Full-F gyrokinetic edge code TEMPEST is being developed under the Edge Simulation Laboratory (ESL) project for exploration of transport, and eventually MHD physics in the edge region. In addition, the Center for Plasma Edge Simulation (CPES) project is developing a gyrokinetic particle-in-cell code XGC-2 that will be employed for similar comparisons.

6.1.3. RWM Damping Models and Rotational Thresholds

Toroidal rotation and rotational shear have many beneficial effects on the stabilization of MHD instabilities and suppression of turbulence that are crucial for high-performance regimes. In particular, rotating the plasma relative to an external resistive wall can compensate for the inherent resistivity of the external wall and stabilize the resistive wall mode (RWM). The critical plasma rotation needed for the stabilization of the RWM depends not only on the rotation profile of the plasma but also on the angular momentum dissipation model. At present, two dissipation models, the ion sound wave damping model and the kinetic damping model have been utilized for predicting the critical rotation speed for rotational stabilization of RWMs using the MARS-F code. Broad qualitative features of the experiment were reproduced using either model [La Haye 2004]. With balanced beam injection, it was recently discovered that RWMs in DIII-D can remain stable at very low rotation [Garofalo 2006]. This was corroborated by experimental results from JT-60U. The MARS-F extended MHD code with kinetic damping has been employed to study the stability of these plasmas and found to generally agree with the experimental observations. The critical rotation appears not to have a strong dependence on β experimentally.

The kinetic damping model presently implemented in MARS-F is a large aspect ratio approximation to the usual analytic model. This model also neglects the diamagnetic and magnetic curvature drifts, and it is generally valid for plasma with large rotation but invalid for discharges with low rotation. To more accurately test the kinetic damping model, it is necessary to generalize the model in MARS-F to include both electron and ion diamagnetic drifts as well as the curvature drift. This will allow the code to be utilized over a much wider range of plasma rotation values, and permit more accurate experimental tests of comprehensive kinetic damping models. This is particularly important due to the recent DIII-D and JT-60U low critical rotation RWM results.

Magnetic field errors can interact with the plasma and slow down the rotation by breaking the toroidal symmetry. To study the development of the rotation profile and RWM dynamics, it is necessary to integrate MARS-F with the ONETWO transport code to self-consistently model the error field effects. The unstable eigenfunction from MARS-F will be utilized to compute the drag in angular momentum through applications of models such as neoclassical toroidal viscosity [Shaing 2003]. This can include both resonant and nonresonant responses. NSTX results suggest that the nonresonant response due to neoclassical toroidal viscosity can contribute significantly to the rotation dynamics [Zhu 2006].

6.1.4. Fast Ion Transport and Stability

Energetic particles can have significant effects on ideal MHD stability. This includes both sawtooth stabilization and destabilization of a range of Alfven Eigenmodes (AEs). The behavior of the energetic particles is one of the subjects targeted by the SciDAC initiatives. Modeling of fast ion effects involves two distinct issues: the effect of the fast ions on the stability and the effect of unstable modes on the fast ion population. State-of-the-art numerical tools exist for studying both facets. Given the fast ion distribution, the fast ion stability contribution will be evaluated using the NOVA-K code to compute the modification to δW. To facilitate analysis of actual DIII-D discharges, the kinetic package from NOVA-K will be coupled to the equilibrium and stability tools routinely available at DIII-D, as well as transport and rf tools for fast ion distribution. The ORBIT code [White 2006] provides a comprehensive tool to study the transport of energetic particles by MHD perturbations. This would have relevance not only on the relation of MHD perturbations on transport of energetic particles, but also on the possible effect on the current profile [Chu 2007].

Self-consistent simulations of the interaction of fast ions with MHD stability require iterations between these two calculations through a transport code such as ONETWO. This can be accomplished within the 1D framework by performing the iterations either within each time step or iterating alternately with the time step as needed. Additionally, the NIMROD MHD code has recent improvements that can treat the fast ion contribution to the nonlinear stability. We will also apply the GYRO code to obtain the stability with kinetic corrections nonperturbatively.

6.1.5. Plasma Disruption and Mitigation

An atomic physics package, based on the KPRAD code, has been developed for NIMROD for the purpose of simulating disruption mitigation experiments. The purpose of the simulations is to better understand impurity mixing by MHD and predict runaway currents, heat fluxes, and halo currents for a given mitigation strategy. A runaway electron diagnostic for the NIMROD simulations will be developed for experimental comparison and prediction. Computationally, the challenge will be to approach the experimental Lundquist numbers and, to the extent that they cannot be presently achieved, to understand the appropriate scaling in order to correctly predict disruption time scales for DIII-D and eventually for ITER.

The formulation of a good theoretical model for impurity injection/neutral penetration is required for any disruption mitigation simulation employing the combined NIMROD/KPRAD code. A predictive model for the penetration of neutral gas jet into the plasma will be developed based on solutions of the 1D conservation equations in the Lagrangian framework.

6.1.6. Plasma Response to Perturbation Magnetic Fields

The MHD response to error and nonaxisymmetric perturbations is increasingly understood to be important. The present understanding of the effects of these error and perturbation fields is largely based on vacuum analyses, but this has revealed several puzzles that indicate that the plasma response must be taken into account. The plasma can amplify, suppress, or modify these fields. In general, linear models essentially describe the response to an external perturbation as a sum of basis functions. The eigenmodes of the MHD operator L provide the most natural option but other basis sets can be used. This includes the approach of Nuhrenberg and Boozer (NB) [Nuhrenberg 2003] for the resonant response, the Normal Mode Approach (NMA) [Chu 2003] focused on the response of the RWM, as well as the recent IPEC code focused on perturbed equilibrium response [Park 2007]. We will modify the MHD codes GATO and NMA to obtain the eigenmodes of L. Extraction of specific features (resonant and nonresonant) of the linear response requires development of numerical tools to relate them to features or components of the external boundary perturbation.

The above approach will only yield the linear plasma response. To study the nonlinear response, we plan to pursue both the neighboring equilibrium formulation and initial value simulations via NIMROD. Plasma rotation will be included in the NIMROD simulations since this is important in the nonlinear response. The neighboring equilibrium formulation bypasses details of the dynamics between the perturbed initial state and final states but requires that the correct constraints be imposed to ensure that only accessible final states are obtained. This is the essence of the almost ideal MHD (AIMHD) concept. In contrast, the initial value approach guarantees accessibility, but at the expense of requiring full details of the dynamical evolution that is numerically demanding.

6.1.7. Transport Codes and Source Modules

Accurate source modules are crucial to the development of integrated modeling. We plan to improve the neutral beam injection (NBI), the rf heating and current drive, and the neutral particle source modules.

6.1.7.1. NBI Source Module. One of the biggest uncertainties in our transport modeling at the present time is related to the fast ion distribution function and its various moments that come into play in a transport code. Thus, for example, we find that often the plasma stored energy and total neutron emission rates do not match experiment with sufficient accuracy without making ad hoc assumptions about the spatial distribution or diffusion of the fast ions. Typically for DIII-D, the beam distribution has to be spread out in space in addition to any prompt or delayed orbit effects that are normally included in our modeling. MHD instabilities may play a key role here. The fast ion distribution problem will be theoretically investigated to determine the range of applicability using the NUBEAM neutral beam module. The challenge will be to implement physically plausible computational models from consideration of fast-ion driven MHD instabilities that ameliorate this fast ion uncertainty issue.

6.1.7.2. RF Heating and Current Drive. Linear calculations of neutral beam and rf current drive are performed using the NUBEAM, TORAY, CURRAY, and GENRAY modules. Present experiments involve substantial nonlinear current drive effects. Injected fast wave (FW) power may interact with the neutral beam injected fast ions at the cyclotron harmonics. The high power electron cyclotron (EC) system both causes quasi-linear distortion of the electrons and modifies the toroidal DC electric field conductivity as a result of the nonthermal electrons, in low density discharges. Some of these effects have been successfully modeled, particularly in steady state, with the stand-alone multi-species CQL3D Fokker-Planck (FP) code. However, coupling of the FP results back into the ONETWO transport code is presently carried out manually and is incomplete. Besides providing advanced modeling for the FW and EC systems, coupling of CQL3D into ONETWO will provide new models for diagnostic signals. The coupling will be accomplished in ONETWO by spawning Fokker-Planck steps at appropriate times in the transport calculation and passing data between the two codes in temporary files.

The DIII-D fast wave current drive (FWCD) experiments on neutral beam heated plasmas have demonstrated that the presence of beam ion cyclotron harmonic resonances may lead to strong damping of the FW on beam ions, resulting in a significant reduction of FWCD efficiency. Since FWCD provides a good means of control of the q profile and an improved off-axis electron cyclotron current drive (ECCD) efficiency by increasing β_e locally, conditions for strong damping of FW on energetic particles should be well resolved before high power FWCD is planned on DIII-D. For this, we propose to further investigate the validity of the quasi-linear diffusion model to provide a more accurate theoretical model of the resonant interactions between ICRF waves and energetic particles. This work will provide a more accurate model of the resonant interactions between ICRF waves and energetic particles, and may resolve the overestimated power absorption from ORBIT-RF/TORIC for C-Mod and DIII-D tokamaks. These investigations of harmonic damping of fast waves will continue to be integrated with the valuable SciDAC work on that topic, including extensive calculations with the full wave code AORSA.

6.1.7.3. Neutral Source Module. The modeling of neutrals plays an important role, since it influences both local and global particle source results. Our current neutral source module is based on a semi-analytic approach, which is not capable of modeling the 2D nature of the MHD equilibrium correctly. We plan to replace our existing module with the NTCC-derived module of GTNEUT, which is computationally efficient and contains more accurate neutral physics, including recycling at the boundary. An interface to link GTNEUT with ONETWO will be developed.

6.1.8. Modernize and Enhance EFIT Equilibrium Reconstruction Capability

Reconstruction of experimental equilibria is an important part of analysis and modeling and has been crucial to several of the past discoveries of new physics. The EFIT equilibrium reconstruction code [Lao 2005] was first developed in 1984 and has since evolved into a major computational tool that is used worldwide. To meet the new DIII-D Five Year Plan challenges, we plan to enhance and modernize the computation architecture and numerical algorithms of EFIT, so that it can more efficiently and robustly reconstruct experimental equilibria over the wide range of configuration space envisioned. We also plan to extend the EFIT reconstruction capability to 3D to model the important toroidally asymmetric effects due to the error and externally applied perturbation magnetic fields.

A new computational structure based on Fortran 90/95 with a unified interface that can conveniently accommodate different tokamak devices and grid sizes, as well as a Python-based GUI will be developed. We will also develop new computational links to allow easy integration with transport and stability physics modules to facilitate kinetic reconstruction and stability analysis as described below.

Development of a 3D reconstruction capability will require a unique formulation and numerical implementation. We plan to extend the EFIT equilibrium reconstruction code to account for the 3D effects using two approaches. In the first approach, the 3D effects will be included using a perturbation method based on the expansion of the MHD equations. The nonaxisymmetric correction to the poloidal flux function describing the perturbation magnetic field is approximately computed by expanding the MHD equations to obtain corrections to the toroidal current density. The method has been successfully applied to treat toroidal magnetic field ripple in the TORE-SUPRA tokamak [Zwingmann 2006].

In the second approach, the toroidally asymmetric error and externally applied perturbation magnetic fields will be included as perturbative corrections to the 2D equilibria provided by EFIT. Asymmetric effects from the external coils will be modeled using a Green’s function formulation. The local plasma response will be obtained by coupling the 2D EFIT equilibrium to 3D stability codes such as NMA or GATO. The 3D effects are included in a quasi-linear sense as local modifications to the 2D equilibria around the rational surfaces. This should be sufficiently accurate if the 3D perturbations are small. This perturbation approach to compute 3D equilibrium based on stability codes is similar to that used in the IPEC code [Park 2007].

EFIT is based on the cylindrical (R,ϕ,Z) coordinate system and can conveniently include the important effects of the magnetic separatrix, islands, and stochasticity. This EFIT 3D development is complementary to the on-going stellarator equilibrium reconstruction project V3FIT [Hanson 2006], that is based on the EFIT response function formulism and the 3D VMEC equilibrium code. Similar to VMEC, V3FIT assumes the existence of nested magnetic surfaces. The stellarator equilibrium reconstruction project will provide the necessary response function for 3D EFIT reconstruction as well as useful 3D reference equilibria to be used as starting points for investigation of the development of the stochastic magnetic regions.

6.1.9. Develop Integrated Modeling Capability and Framework

As described above, analysis and modeling necessarily involve self-consistent integration of various physics elements from different topical science areas such as turbulence and transport, macroscopic equilibrium and stability, and heating and current drive. Two examples are generation of the edge pedestal stability space diagram as shown in Fig. 2-7 and extension of transport modeling towards the edge as described in Section 6.1.1 for investigation of ELM physics. These tasks necessarily involve iterations among EFIT kinetic equilibrium reconstructions, ELITE edge peeling-ballooning stability analyses, and ONETWO transport simulations.

To facilitate the coupling and to allow the interaction to more efficiently take place, we will develop a modern integrated modeling and fitting (IMFIT) framework based on FORTRAN F90/95 and Python using Common Component Architecture (CCA). The goal is to develop an integrated modeling tool that can be efficiently applied to support key elements and operation of the new DIII-D five-year experimental program. The use of CCA will allow new physics modules to be integrated into the framework with ease as they become available. An important feature of this new modeling tool is the emphasis on experimental data analysis and self-consistent integration of equilibrium reconstruction, transport, and stability analyses to support experimental operations. It will be designed with a flexible data interface to allow easy communication with existing database clients such as MDSplus and physics codes. To optimize the available resources, we plan to make full use of available software and physics modules presently in use to support DIII-D and from other projects such as NTCC, and develop new ones if necessary to satisfy specific DIII-D modeling requirements. Close coupling with theory and direct numerical simulations of fundamental fusion plasma processes such as those carried out under SciDAC (see http://www.science.doe.gov/scidac/) as well as other integrated modeling projects are also clearly required.

This integrated modeling development is part of the collaboration with the EAST tokamak program, as described in the International Collaboration section. Our first joint goal is integration of the two key DIII-D data analysis tools: the EFIT equilibrium reconstruction code and the ONETWO transport code. This will then be followed by integration of stability codes such as ELITE, PEST3, DCON, and GATO. Additionally, to impact experiment and analysis, the speed of modeling codes and their ease of use are also key factors. Specifically, we propose to carry out the following tasks:

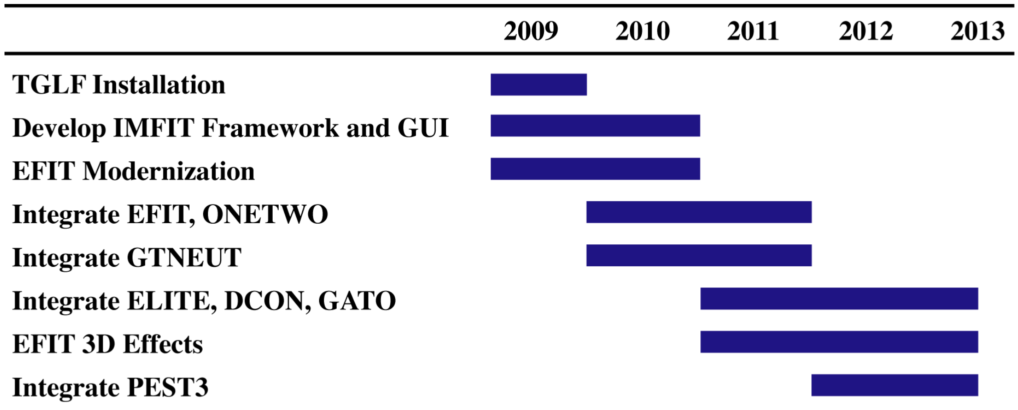
- Install the new TGLF transport module into ONETWO.
- Modernize EFIT and extend it to include weakly nonaxisymmetric effects.
- Develop the IMFIT integrated modeling and fitting framework using Python and Fortran 90/95.
- Integrate equilibrium and transport components EFIT and ONETWO into IMFIT.
- Integrate ideal stability modules ELITE, GATO, and DCON into IMFIT.
- Integrate resistive stability module PEST3 into IMFIT.
- Integrate neutral source module GTNEUT into IMFIT.

A very approximate schedule is shown in Table 6-2.

6.2. VALIDATION OF INDIVIDUAL MODELS

Individual models make up the core of the modeling process. In order to have a predictive integrated model, the individual component models must be known to correctly represent the relevant physics theory and to provide an accurate description of actual behavior. This process is called Verification and Validation.

Table 6-2
Approximate IMFIT Development Schedule



Computational models are being developed to emulate the physics that takes place in a tokamak discharge. These models include the heat, particle, current, and momentum sources and the transport of these quantities in a discharge. The sources available to DIII-D, namely neutral beam heating and current drive, electron cyclotron heating and current drive, fast wave heating and current drive, bootstrap current, and neutral ionization, are treated in Section 6.2.1. Core transport, stability, and energetic particle modes are addressed in Section 6.2.2. The models of the pedestal are discussed in Section 6.2.3, and the boundary is addressed in Section 6.2.4.

6.2.1. Sources Of Energy, Current, Particles, and Momentum

The applied sources of energy, current, and momentum are relatively well understood and modeled in present-day codes for simple quiescent plasmas. The codes for these sources are based on first-principles physics and have been shown to be reasonably close to experiment, as long as other phenomena like MHD activity or non-Maxwellian distribution functions are not present. Modeling of sources of particles is not as well advanced, particularly since 2D and possibly 3D effects are important, while diagnostic measurements with this spatial complexity are not available. The particle source is theoretically expected to have a large effect on the pedestal and remains an important object of research.

6.2.1.1. Neutral Beam Heating and Current Drive. The DIII-D neutral beam injection (NBI) system has been the primary system for plasma heating since the inception of the DIII-D program. Its seven beam sources are used to support a wide range of experiments, providing toroidal current drive, angular momentum input and control, and diagnostic capability as well as heating power. Rotation of one beam line (two sources) to the counter direction in 2005–2006 has enabled control of the angular momentum input to the plasma along with control of the power. Use of the counter beams also allows some measure of control of the NBCD, which provides greater flexibility in making noninductive advanced tokamak plasmas. The NBI system is frequently controlled by the Plasma Control System so that the beams can be modulated in a way that maintains simultaneously the plasma pressure and rotation at preset waveforms.

The physics of heating and current drive by fast ions is believed to be well understood, as long as the target plasma is quiet and stable. The most accurate calculations are done with the NUBEAM Monte-Carlo code package, which has been incorporated into the ONETWO and TRANSP transport codes. This code follows individual ions in a specified plasma as they slow down, scatter, and transfer energy and momentum to the background plasma. The geometries of the plasma and of the neutral beams are accurately modeled. The results are in excellent agreement with experimental observations, including the rate of neutron generation by fusion reactions. Because these calculations can be very time consuming, simplifying assumptions are sometimes used. These include using analytic expressions for the fast ion distribution function and considering only the first complete poloidal orbit, as in the NFREYA package.

In plasmas with strong fluctuations, experimental results are not consistent with the predictions of the existing codes. In the presence of electromagnetic modes such as Alfven eigenmodes (AEs) or neoclassical tearing modes (NTMs), the heating profiles inferred from experimental measurements are significantly broader than calculated.

Key objectives of the next five years include:

- Fully test the existing models for neutral beam heating, current drive, and torque making maximal use of the existing capabilities: co- and counter-NBI, an excellent motional Stark effect diagnostic (MSE) on both co- and counter-beams, and the Fast Ion D-Alpha (FIDA) diagnostic that measures the radial profile of characteristics of the beam ion distribution function. Off-axis beams will also provide a great opportunity for validation of the models. Validation of source models is done in quiescent plasmas where the transport is either understood or not much changed by the application of the heat source. In particular, MHD and TAE modes will be avoided due to their effects on the heating and current drive profiles.
- Develop and validate an accurate and robust predictive model for NB energy, momentum, and current source profiles in the presence of MHD activity. Validation of the model will include detailed comparisons to data from MSE, FIDA and loss measurements at the boundary. The models so developed will be of important applicability to the transport of alpha particles in a burning plasma. Again, current drive from the off-axis beams will provide an additional constraint which will be used to test the model. Accurate evaluation of the effects of MHD modes on the heating and current drive profiles requires confidence in the quiescent source characterization of the previous step.

6.2.1.2. Electron Cyclotron Heating and Current Drive. The physics behind electron cyclotron heating (ECH) and electron cyclotron current drive (ECCD) has been well encapsulated in computer codes (TORAY-GA, GENRAY, and CQL3D) based on first principles, as described in Section 5.5.1. These codes describe the trajectories of the waves in the plasma, where the waves are absorbed, and the resultant heating and current drive. These models have been extensively validated against experiment over a wide range of conditions (but not yet all conditions of interest).

Extension of the basic model for heating and current drive will be extended to highly relevant conditions not yet fully explored:

- **Evaluate far off-axis ECCD**, at normalized minor radius greater than 0.5. This regime is important for use of ECCD for control of some MHD instabilities, but the model has not yet been tested under this condition. MSE data from both the co- and counter-beams will permit the effects of changes in the radial electric field to be distinguished from changes in the driven current profile.

- **Measure ECCD in the presence of MHD activity.** The effects of MHD activity on the magnitude and profile of ECCD is required to predict the needs for MHD control in burning plasma experiments. This effect will be tested by applying modulated ECCD in the presence of static magnetic islands and inferring the response from the MSE signals. This same procedure will be used to measure the ECCD profile at a rational surface without an island, for comparison.
- **Integrate the ECH/ECCD models with transport models like TGLF.** While the power density and current drive from ECCD are accurately calculated by the present codes, the plasma response is less well understood. For example, the electron density when ECH is applied is frequently—but not always—observed to decrease in a process colloquially termed "density pumpout." This effect on the density profile does not appear to stem from the physics of ECH, but rather it appears to be a response of the plasma transport processes to the local heating. Similarly, when the ion temperature is much higher than the electron temperature in beam-heated discharges, the ion heat and momentum transport are usually seen to increase significantly when ECH is applied. Similar observations may be made about ECCD. When ECCD is applied, the global transient and steady state response of the current density profile to the applied ECCD cannot be determined from the ECCD models; rather, the Faraday and Ampère equations must be solved self-consistently with the electrical conductivity, which may be affected by the EC interaction if non-Maxwellian distributions are generated. Also radial transport of the current-carrying electrons can broaden the ECCD profile, which may be important for some applications such as NTM suppression. Experimental data will be compared with simulations to test the validity of the integrated model in predicting this behavior.

These important processes will be studied through development and application of integrated models. The integrated model will include the EC interactions, as predicted by codes like the linear ray-tracing code TORAY-GA or the Fokker-Planck code CQL3D, but also other effects. An integrated modeling code which self-consistently calculates the changes in the current profile due to the driven current and to the changes in the conductivity due to the localized heating and possibly distribution function distortions, the changes in the pressure profile due to the ECH power and to any modification of the transport coefficients, changes in the heating profile from other sources like neutral injection, and changes in the equilibrium due to the changes in the pressure and current profiles will be developed and validated. The effects of changes in transport due to changes in the local gradients may be addressed by applying GYRO or TGLF. TGLF is more suitable for an integrated modeling code since it is much less computation-intensive than GYRO. In all cases the MSE diagnostic is critical to a full evaluation of the driven current.

The ECH and ECCD systems on DIII-D are uniquely suitable for studies of validation of the integrated modeling code because they can be strongly localized and controlled. This makes ECH a highly suitable tool for perturbative transport studies, where the ECH power is modulated and the amplitude and phase responses of the plasma profiles are recorded. Extensive work on DIII-D and other devices has shown the value of this approach. Modeling of the plasma response, not only the electron temperature but the ion temperature and plasma density as well, will provide a sensitive test of an integrated model. The localized interaction will also be used to provide steady-state heating at a particular minor radius, which is often useful for changing gradients for testing modeling codes. The 110 GHz ECH system on DIII-D will be upgraded from 6 MW gyrotrons to 12 MW, thereby increasing the strength of the perturbations that will be possible. DIII-D's unique and extensive set of diagnostics for measurement of profiles and turbulence are key to performing effective code validation.

6.2.1.3. FW Electron Heating and Current Drive. DIII-D work on Fast Wave Current Drive (FWCD) since 1991 has shown that under conditions with reasonably high central electron damping and without significant competing damping mechanisms, the driven currents are in good agreement with ray tracing models. Such models are rather similar to the computer codes used to successfully predict ECCD and Lower Hybrid Current Drive (on other tokamaks). In fact, the GENRAY package can be used to model wave propagation in realistic tokamak geometry for all three frequency regimes.

However, in many cases of practical interest, damping mechanisms other than direct electron absorption are important. These other losses reduce the fraction of power that goes to the desired electron heating and current drive in the core. Damping at high ion cyclotron harmonics on energetic ion populations, such as are created by neutral beam injection or by the alpha population in a burning DT plasma, compete with direct electron absorption. Various edge loss mechanisms such as power loss in rectified rf sheaths on surfaces, parametric decay, and surface wave generation are potentially important in cases with less than complete single-pass absorption in the core. The frontier in the modeling efforts for FWCD is in these cases with multiple absorption mechanisms. The plasma response to most of the core damping mechanisms is intrinsically nonlinear, which makes the time response of the plasma to a change in the FW power complicated and introduces sensitivity to initial conditions. The RF SciDAC community is working on modeling such cases, with the goal of understanding DIII-D results from experiments on high harmonic damping on injected neutral beams. Full wave packages such as AORSA and TORIC are used in this modeling along with Fokker-Planck solvers like CQL3D and ORBIT-RF. A topic of current work is the modeling of edge losses as a modification of the boundary condition in these codes; the DIII-D results clearly point to the importance of edge losses in cases with weak core absorption. Continuation of this state-of-the-art modeling in conjunction with the SciDAC will be a key focus of ongoing research for DIII-D.

Time-dependent modeling is also important for developing AT scenarios in which FWCD is used to control the evolution of the central current. Such efforts will continue using the codes TSC, TRANSP, CRONOS, and time-dependent ONETWO. More generally, the development of fully integrated models is important for understanding the applications of FWCD for the same reasons described in the previous section on ECCD modeling.

One area that will be modeled in greater detail is the physics of FW coupling structures. Although work on DIII-D and on many other tokamaks has shown a general quantitative understanding of the resistive antenna loading, very little is understood about the nature of the breakdown mechanisms that limit the power that can be coupled to tokamak plasmas with loop antennas. Both experimental work on tokamaks and in laboratory experiments and modeling will be carried out to advance this area beyond the almost entirely empirical state-of-the-art. Since the requirements of ITER for reliable FW power coupling are considerably more stringent than any present-day experiment due to the much larger plasma-to-wall distances that are needed (on the order of 20 cm, to be compared with present high-power FW antenna operation at a typical distance of 4 cm), this topic is in urgent need of substantial development. Modeling will be carried out in conjunction with the RF group at ORNL and with the SciDAC.

Key objectives over the next five years include:

- Evaluate the effects competing with electrons for absorption of wave power and compare with theory for model validation.
- Determine the effects of a minority of hydrogen ions on ion absorption, and compare with models that include nonlinear effects.
- Provide data on breakdown that supports the design of a new low voltage antenna.

6.2.1.4. Bootstrap Current. The bootstrap current is a key feature of AT plasmas because it is needed for steady-state operation. Steady-state, fully noninductive operation of a reactor tokamak necessarily entails a high bootstrap fraction because of the high power cost of external noninductive current drive. The bootstrap current is also a factor in the physics of other important tokamak plasma phenomena. For example it is a key parameter in determining the total current density in the pedestal region, and therefore in setting the characteristics of ELMs. Also it is the feature that destabilizes neoclassical tearing modes. Understanding and verifying the physics of the bootstrap current thus is a significant part of the overall effort to produce advanced tokamak plasma scenarios.

The basic theory of the bootstrap current is an integral part of neoclassical transport theory. This current is, in essence, a diamagnetic current associated with the toroidal projection of the finite width of the banana orbits of trapped particles. These transfer momentum to passing particles, amplifying the pressure gradient driven toroidal current. Past experiments have shown agreement between the calculated and measured total bootstrap currents in tokamak plasmas. However, there is no local measurement of the bootstrap current density with sufficient precision to test neoclassical predictions. Working in plasmas with high overall bootstrap fraction, and using the ability provided by the DIII-D ECH system to locally modify the electron pressure gradient, we will measure the local bootstrap current with sufficient precision to test the theory. Single-parameter scans will be carried out to isolate and measure the roles of the contributing terms in the theory.

Further research will be carried out on theory and experiment in those regions of the plasma where the standard neoclassical theory breaks down. Near the magnetic axis, the radial extent of the orbits of trapped particles is no longer small compared to the minor radius, and in fact there are the so-called potato orbits, which are trapped particle orbits which encircle the axis. The presence of potato orbits modifies the neoclassical theory and allows a finite bootstrap current at the magnetic axis. Verification of this effect will enhance our ability to project tokamak physics to high bootstrap fraction reactor scenarios.

The other crucially important region of the tokamak plasma where the bootstrap current is modified is in the steep gradient region of the H-mode pedestal. Here the assumption that the drift orbit width is small compared to the pressure gradient scale length breaks down, at least for ions. Thus the electron terms in the bootstrap calculation should still be valid, but the ion contributions need to be modified. The calculation of currents in this region is further complicated by the presence of a loss region in the velocity space of the ions – certain orbits are lost to the wall – leading to a distortion of the ion velocity distribution. In this region and in the SOL just outside the separatrix, the calculation of the bootstrap current is thus strongly coupled to the calculation of plasma flows, and to the viscous transfer of momentum across flux surfaces. Both theory and creative experiments will be carried out to elucidate these processes.

Key objectives for research into bootstrap current effects includes:

- Measure and compare the bootstrap current profile in the core with standard theories for a wide variety of plasma conditions, specifically the different gradients that drive the core bootstrap current.
- Develop a model for pedestal bootstrap current and test its dependencies against experiment.

6.2.1.5. Neutral Ionization and Particle Sources. Neutrals play a role in the particle transport, energy transport and momentum transport equations in the region just inside the last closed flux surface. Neutral ionization in this region is the dominant particle source for most L-mode and H-mode discharges in DIII-D. Impact ionization of neutrals and charge exchange of confined ions with neutrals are two sinks for thermal energy. In addition, the charge exchange process damps toroidal and poloidal momentum. With existing knowledge, there are significant uncertainties about the magnitude of these effects. However, these effects are not negligible and it is necessary that we adequately characterize these effects in order that we can produce a predictive model for edge temperature and density profiles.

The atomic physics required to model these effects is well known. However, our knowledge of the neutral density and its spatial distribution are poor because we do not have direct measurements of the neutral density and because the neutral density has a 3D spatial distribution. There are various levels of sophistication that are used to estimate the neutral density and its effects. For instance, the ONETWO transport code implements a kinetic neutrals model to account for neutral transport within the plasma. The implementation of this model assumes that the neutral density is constant on a flux surface. The neutral flux arriving at the edge of the plasma is obtained in an ad hoc fashion. Typically, the impinging flux is adjusted so that the total computed particle confinement time is comparable to the energy confinement time. This approach is satisfactory for core transport calculations but leads to large uncertainties in transport quantities near the plasma edge. More sophisticated approaches (UEDGE, DEGAS, SOLPS, etc.) use as much experimental data as possible to compute profiles of temperature and density in the scrape-off layer (SOL), the source of neutrals at the wall and divertor plates, the transport of neutrals through this plasma to the plasma edge and the transport of neutrals in the confined plasma. Despite the sophistication of such calculations, the accuracy of the answer is not known because the models have many adjustable parameters and the SOL plasma is not sufficiently well known.

We propose to improve our knowledge of edge neutral effects in three ways, which act simultaneously as experiment interpretation and model validation:

- First, we will develop a capability for experimentalists to rapidly perform 2D calculations (assuming toroidal symmetry) of edge neutral density and ionization rate with a simple model for the SOL plasma. This capability could be a stand-alone code, such as the interpretive version of UEDGE, or a module in ONETWO, such as a coupled version of GTNEUT. This capability will allow experimentalists to compute edge neutral physics with considerably more realism than is now possible.
- Second, we will perform sensitivity analysis with the sophisticated models, including UEDGE, GTNEUT and SOLPS, to determine how uncertain our knowledge of the edge neutral density is and to determine what measurements are needed to further constrain these models.

- With guidance from these two steps, we anticipate that we will implement additional measurements to further constrain the models. For instance, additional pressure gauges and wall Langmuir probes in a few strategic locations would certainly be very helpful. Measurements of temperature and density near the inner strike point would be very valuable. Of course, direct measurements of neutrals are very desirable and could be potentially implemented at one or two strategic locations. Two possible techniques to do this are laser-induced fluorescence and heterodyne interferometry.

As a result of the work described here, we would achieve much improved accuracy and confidence in our modeling to compute neutral physics effects in DIII-D, which is a necessary ingredient in developing a predictive model for the pedestal (Section 6.2.3.1).

6.2.2. Core Transport and Stability

Understanding core transport and stability are critical to plasma performance and are therefore large and essential components of the DIII-D research plan for 2009–2013. Regarding transport, the Transport Sciences white paper (http://psfcwww2.psfc.mit.edu/ttf/transp_init_wht_paper_2003.pdf) of the Transport Task Force calls for a focus on electron thermal transport and the physics of the H-mode and pedestal as key research needs for the fusion program. In accordance, these topics form a core part of transport research in the DIII-D plan. Validation of computational models of electron and ion transport in the core and in the pedestal against measurements of diffusivity and turbulence levels is a key objective for the next five years. For stability research, the focus is shifting from ideal MHD, which is now well developed as a predictor of stability limits, to nonlinear, nonideal, and nonaxisymmetric effects. With its wide array of turbulence and magnetic diagnostics, its flexible heating tools, and its symmetric and nonaxisymmetric magnetic coil systems, DIII-D is in an excellent position to advance these studies.

6.2.2.1 Core Transport and Turbulence. The major emphasis in the next five-year period will be on theory-experiment comparisons to validate the simulation codes, like GYRO, and the transport models like GLF23 and TGLF. With our turbulence diagnostics and control tools, we can directly test the detailed turbulence predictions made by GYRO for particle, heat, and momentum diffusivities. (Off-axis and co- versus counter- neutral beams will be particularly useful for validating the GYRO model of momentum transport.) Using steady-state and modulated heat, momentum, or particle sources, we will perform detailed tests of the transport models for energy, momentum, and particles. For example, it is now possible to perform experiments where one nondimensional plasma parameter is varied while keeping the others fixed; this allows a direct test of the dependence of the simulation and transport models on that parameter. We also have the capability to design experiments to isolate the effect of one type of turbulence (e.g., trapped electron mode) in order to test the physics of that particular mode while suppressing the effects of the other modes (ion temperature gradient and electron temperature gradient modes). These scans of single nondimensional parameters are the best way to validate the computational models, particularly with the excellent set of turbulence diagnostics available on DIII-D.

A key component of validating codes for use in integrated and predictive modeling is the development of detailed synthetic diagnostics to facilitate direct, quantitative, "apples-to-apples" comparisons between simulation and experiment. Synthetic diagnostics are models which attempt to accurately generate an experimentally measured quantity from the fields calculated in a simulation. As such, they may for example include the degree of spatial localization (or nonlocality) of a specific measurement, as well as its frequency and wavenumber sensitivity, in addition to the relation between the measured quantity (e.g. intensity of radiation at a given frequency) and the physical fields of interest (such as density or temperature). This issue has been identified as a key priority by the USBPO, and the extensive diagnostic suite available on the DIII-D machine, along with the tight couplings to the GA theory group, means that it is well positioned to be a world leader in this area. Discussions of virtual diagnostics have most recently focused upon the area of microturbulence and anomalous transport, and efforts are currently underway to develop tools for post processing data from codes such as GYRO and TGLF which will generate synthetic BES, PCI, FIR and high-*k* backscattering, correlation ECE, etc. Many of these tools can also readily be combined with MHD codes such as NIMROD, and so can be used to help validate those codes as well. These new tools will complement existing synthetic diagnostics for linear MHD eigenmode codes, which have been successfully used to identify global Alfvenic eigenmodes in DIII-D.

In the coming five-year period, a high priority of the DIII-D program in this area will be the continued support and development of these synthetic diagnostics to represent new diagnostics as they are developed. The development of synthetic edge diagnostics is an area that will in particular receive significantly more attention, given the importance of this region for the success of ITER and DIII-D's extensive diagnostic set for the plasma edge. A second challenge will be the issue of data management. In general, experimental diagnostics measure a signal for hundreds of milliseconds or more with limited spatial resolution, whereas one has the opposite balance for gyrokinetic or MHD codes. Direct code-experiment comparisons will therefore require significantly longer simulation time runs than are commonly performed to date to generate synthetic signals long enough to ensure good statistical convergence in time/frequency. These long-time simulations will likely generate hundreds of gigabits or more of spatiotemporal data, which will require new approaches for efficient exploration and analysis of the results. Finally, as these synthetic diagnostics develop and mature, they should be used to develop specific, focused validation experiments on DIII-D, based on close collaboration between the relevant experimentalists and theorists. In ranking these proposals, the DIII-D group should recognize that although the optimal validation physics environment may not be a particularly "challenging" or directly ITER-relevant one, such tests are an essential part of validation process which is also key for the success of ITER, and one which DIII-D is uniquely poised to make contributions.

6.2.2.2. MHD Stability and Control. Over the past two decades, ideal MHD has become well established as a reliable predictive tool for tokamak stability limits. Numerous examples demonstrate the accuracy of ideal MHD stability limits, including beta limits caused by global kink modes, internal kink modes related to the strong gradients of internal transport barriers, and the peeling-ballooning mode model that explains edge-localized modes (ELMs). This success has resulted in part from the development of diagnostic instruments to provide the detailed, accurate measurements of the internal profiles of the plasma that are crucial for accurate stability calculations. More recently, attention has started to turn to nonideal, nonlinear, and nonaxisymmetric effects on MHD stability, and validation of these more realistic models will be a major emphasis of the next five years.

Real-Time Stability Calculations. The next frontier for ideal MHD stability models is to develop and validate real-time stability calculations for use in plasma control, in much the same way that real-time

MHD equilibrium calculations have become routine. An ideal MHD stability code such as DCON will be adapted for real-time execution in the DIII-D plasma control system (PCS). This is a challenging undertaking that will require the code to be streamlined for rapid execution and to be robust against imperfections in the input data. The results will be validated against standard stability calculations and experimentally observed stability limits. Ultimately, such a real-time stability calculation will be used to detect approaching stability limits as part of a disruption avoidance system.

Plasma Rotation. Validation of the interaction of plasma rotation with MHD stability will continue during the next five years. Here several kinetic models exist for dissipation that leads to the torque exchanged between the rotating plasma and a nonaxisymmetric field. Such torques and their effect on the plasma rotation are key elements of the physics of stabilization of resistive wall modes by plasma rotation, braking of plasma rotation by magnetic field errors, and the onset of error field-driven tearing modes. The MARS-F code will continue to be the primary vehicle for validating these models. The model predictions will be compared to experiments that take advantage of DIII-D’s tools for varying plasma rotation, including variable co and counter neutral beam injection and the wide range of magnetic perturbation spectra provided by the C-coils and I-coils.

Three-Dimensional Equilibrium. Closely related to the topic of plasma rotation is the three-dimensional response of the plasma equilibrium to externally imposed nonaxisymmetric fields, as modeled in codes such as MARS, VALEN, and NMA. Recent results with the IPEC code developed at Princeton predict that the spatial structure of the plasma response may differ significantly from vacuum field calculations, and thus nonaxisymmetric fields must be taken into account in understanding the effects of error fields and in designing the optimal correction of the error fields. These predictions will be validated, using the codes mentioned as well as a three-dimensional version of EFIT.

Extended MHD Modeling. In general, a major effort in the next five years will be the validation of nonlinear, three-dimensional, resistive stability models. These are embodied in codes such as NIMROD and M3D, which are the basis for a SciDAC activity on Extended MHD Modeling. Nonideal effects such as resistive and kinetic effects are also included in linear stability codes such as MARS, NOVA-K, and PEST3, while the BOUT code provides nonlinear stability calculations for higher-n modes in the plasma edge. The models will be compared to each other and to experimental measurements of sawteeth, tearing modes (including multi-mode coupling), ELMs, and disruptions. The nonlinear, three-dimensional codes potentially provide the most complete and realistic calculations of plasma stability, but are also computationally intensive and time-consuming to run and difficult to interpret quantitatively. Synthetic diagnostics will be developed in these codes as a key element in making detailed comparisons to experimental data.

Diagnostics. New diagnostics proposed for the next five years will be crucial to providing detailed experimental data for validation of stability models. These include upgrades of the soft x-ray arrays for toroidal resolution of rotating MHD modes and the static plasma response to nonaxisymmetric fields, additional toroidally distributed magnetic diagnostics for three-dimensional equilibrium reconstruction and the plasma response to error fields, magnetic diagnostics with higher spatial resolution for measurement of ELMs and other modes shorter wavelengths, and two-dimensional imaging of electron cyclotron emission to obtain detailed measurements of islands and other MHD mode structures. By constraining the shape of the internal flux surfaces, a two-dimensional ECE imaging system will also provide data on the equilibrium current density profile, thus supplementing the motional Stark effect diagnostic.

6.2.2.3. Energetic Particle Modes: Instability and Effects. Energetic particle modes are of crucial importance in tokamak discharges, particularly in burning plasmas that will have a significant component of energetic alpha particles. These modes are known to transport energetic particles, which alters the heating profile and bootstrap current profile, and in the case of neutral injection also the current drive profile. Modeling of present-day discharges has shown the importance of treating the instabilities and resulting transport correctly.

There are two distinct categories of modeling of instability and effects on the plasma of energetic particle modes. The first category is approximate analysis with modest computational requirements that can be used for control-room guidance and for data surveys. The second category is relatively complete models that can only be applied to a small number of cases and may be computationally intensive.

The between-shot models that will be developed and automated include the following:

- Gap structure using the CONT code, yielding envelopes, rotation, TAE mode frequencies, and so on.
- Classical fast ion profiles for comparison with measurements. The fast ion pressure profiles will be generated from diagnostic data and the ONETWO or TRANSP transport codes using kinetic EFITs and FIDA (Fast Ion D-Alpha) profiles. The classical neutron emission rate will be compared to measurements.
- Systematic comparisons of instability measurements and theoretical growth rates over a large range of experimental parameters to validate the predictive ability of the codes, particularly GYRO and TGLF.

Lengthier, more accurate, analysis includes validation and use of more elaborate models:

- Comparison of measured sawteeth with Porcelli model.
- Application of the NOVA, GATO, and NOVA-K codes. The NOVA-K code is a kinetic-MHD stability code for tokamaks with energetic particles. It uses the plasma pressure and safety factor and equilibrium quantities to produce a file that can be read by a test particle orbit code to study plasma transport. It also produces plasma density perturbations for comparison with experimental measurements. This code has been used routinely for the study of TAE and/or RSAE modes. It will be extended so that it can be utilized to study very low frequency unstable modes, such as the sawtooth and the fishbone. The NOVA-K code will be extended to include two-fluid effects, which is important for capturing the physics associated with the coupling of shear Alfven waves to the sound wave. Recent results on DIII-D indicate that the acoustic coupling of shear waves is essential in understanding many observations.
- Application of new codes from the SciDAC Energetic Particle program.
- Application of the ORBIT and ORBIT-RF codes for understanding effects related to the interaction of particles with magnetic fields. The intent of changes to ORBIT is to achieve high throughput of fast ion orbits and their interactions with Alfven eigenmodes observed in DIII-D and computed by NOVA. The integration of NOVA output into ORBIT and the easy initialization

of ORBIT runs based on DIII-D profile and equilibrium data is a high priority for the five year plan. The ORBIT-RF code, developed at GA, is based on the ORBIT code and solves the time dependent Hamiltonian guiding center drift equations. It extends the ORBIT code to include a quasi-linear high harmonic radio frequency diffusion operator with the wave fields produced from two-dimensional ion cyclotron resonance frequency full wave code. This code has recently been used to compute the energetic particle distribution function in a DIII-D discharge during the rf stabilized giant sawtooth in an effort to validate the combined sawtooth model and the calculated ion distribution function.

In the absence of a complete physics-based model for the fast ions, two approaches to fast ion profiles will be used for integrated modeling. One is the current approach: ad hoc fast-ion diffusion is added to TRANSP or ONETWO in order to match the neutron rate and EFIT and FIDA profiles. The second ad hoc model is a virtually instantaneous (compared to slowing-down times of the fast ions) modification of the deposition profile to match the experimental measurements. These two approaches effectively bracket realistic treatments of the true redistribution and will be tested for optimum accuracy.

6.2.3. Pedestal

The temperature and density of the H-mode pedestal are boundary conditions required for predictions of plasma performance with core transport models. For present day machines, these boundary conditions are obtained from experiment. However, for predictions of global performance in future machines, such as ITER or DEMO, it is necessary that we develop and validate models for the pedestal height.

The pedestal height is controlled by a combination of heat and particle sources, fine scale transport due to turbulence and collisions, and larger scale transport due to MHD events. Significant theoretical and experimental advances are required to understand how these elements interact to determine the pedestal height. At this time, key issues for pedestal height are the following:

- Determine how the pedestal width is set. For ELMing H-mode discharges, it has been shown that the peeling-ballooning theory for edge stability can be used to predict the pedestal pressure height if the width of the pedestal is known. Thus, it is necessary to understand how the pedestal width is set.
- Develop an understanding of the nonlinear evolution of an ELM. For ELMing H-mode discharges, it is necessary to be able to predict the particle and energy losses from an ELM, so that we know both the impact on plasma-facing components and the starting conditions for pedestal evolution after an ELM crash.
- Develop an understanding of the physics of the enhanced particle transport. For ELM-suppressed discharges (QH-mode or via application of RMP fields), the particle transport is increased, which allows the pedestal to remain in an MHD-stable state. It is necessary to understand these regimes.

There are major efforts within the U.S. and within the international community to develop the theoretical elements of a comprehensive model for the pedestal that can resolve these issues. These efforts are crucial for understanding the pedestal and we believe that they will enable the worldwide community to obtain a good understanding of pedestal physics within the next decade. Benchmarking of these models against experimental data will also be crucial in this effort. DIII-D will perform experiments to validate the individual elements of a pedestal model and ultimately to validate a complete predictive model of the pedestal.

6.2.3.1. Transport and Turbulence. A key issue for predicting the pedestal height is obtaining an understanding of the physics that sets the pedestal width, the region of steep edge gradients. One possibility for this physics is that the pedestal exists where ExB shear is sufficiently strong to reduce or stabilize turbulence. Within this framework, the pedestal width is set by the physics that causes shear in the edge E_r and by the transport at the top of the pedestal, which halts the inward growth of the pedestal. Another theoretical idea is that the width of the pedestal is controlled by ion orbits and losses.

There is intense work within the U.S. theoretical community to develop "first-principles" theory-based predictive models for turbulent and neoclassical transport within the pedestal. These models will provide much better capability to study the physics of the pedestal width, whether it is controlled by one of the mechanisms noted above or by other physics. These models are extensions of the gyrokinetic equations to conditions valid in the pedestal and model the standard gyrokinetic instabilities (ITG, ETG, TEM and KBM modes) as well as ion neoclassical transport. These models include TGLF (Trapped Gyro-Landau Fluid), developed by the GA theory group, XGC-2, developed under a SCIDAC contract to the CPES (Center for Plasma Edge Simulation) group, and TEMPEST, which is an OFES/OASCR activity and is under development by the ESL (Edge Simulation Laboratory) group. These models will be available within the next five years and will form the basis for a concerted effort to unravel the physics of the pedestal.

DIII-D is in a very good position to test such models due to its edge diagnostic set and to the flexibility of the machine, particularly for plasma shape. The validation process will include the following elements:

- Perform careful scans to determine the response of the pedestal, particularly the width, to variations of single physics parameters. A successful model must be able to correctly predict pedestal trends in such variations. Candidates for these physics parameters include the ion toroidal or poloidal gyroradius, the pedestal beta, the edge magnetic shear and the edge particle source.
- Characterize the effect on the pedestal width of loss channels for energy and particles in the pedestal. Perform sensitivity studies in the analysis so that we can determine the real level of our uncertainty with respect to edge transport analysis and to highlight areas where more work is needed to improve measurements or analysis capability.
- Test whether the models predict the observed build-up of the pedestal temperature and density profiles during the ELM-free phase after an L-H transition, when predictive models are available in transport codes.
- If models successfully pass these tests, test them at a more microscopic level. In particular, we will obtain measurements of fluctuation intensities, wave and frequency spectra and of turbulent fluxes. The results will be compared to the predictions of the models.

At this time, there is very little empirical or theoretical understanding of the enhanced particle transport observed in ELM suppressed discharges, obtained via QH-mode or application of RMP fields.

This enhanced transport may be related to effects of three-dimensional magnetic fields. Key issues that will be addressed on DIII-D to develop this understanding include:

- Characterize edge particle transport in QH-mode and RMP discharges and determine its dependence on control parameters.
- Determine the 3D magnetic structure in RMP discharges. In particular, determine the degree to which the plasma shield out magnetic perturbations.
- Interact with the theoretical community to develop physics concepts for enhanced particle transport and develop experimental techniques to test these concepts.

Some upgrades or additions to the DIII-D diagnostic set are very desirable and may be needed to perform adequate tests of models. In particular, measurements of the particle source and of the magnetic shear (or edge current density) are very important. Direct measurements of main ion parameters, particularly of rotation, are very desirable and may be necessary to test neoclassical aspects of the models. A possibility for this measurement would be CER spectroscopy of a lithium beam on the plasma deuterons.

6.2.3.2. ELM Stability. Edge MHD stability in the pedestal must be understood in addition to the transport in order to model and predict the pedestal height in both ELMing H-mode discharges and in ELM-suppressed discharges. In the former, the ELM instability causes repetitive loss of particles and heat from the plasma edge and places stringent limits on the evolution of the pedestal height. During the last few years, the linear peeling-ballooning model of the ELM onset has been developed and measurements in DIII-D and elsewhere have provided very strong evidence that this is a valid model to predict the ELM onset. In addition, this theory has provided a way to understand regimes with no ELMs, produced with stochastic edge magnetic fields, by showing that the edge pressure is below the levels that trigger an ELM. An extended version of this model, including rotation, has also provided a promising explanation for the ELM-free quiescent H-mode as being a regime in which access to the low-n branch of the peeling-ballooning mode coupled with strong rotational shear in the edge leads to saturation of the instability.

Continuing advancement towards a fully predictive model of the pedestal requires that we understand nonlinear MHD in the edge. For instance, the saturation of the ELM instability and the losses of heat and particles are due to nonlinear effects. In addition, additional physics, such as rotation, appear to be important for both linear and nonlinear MHD and these effects must be understood. Several theoretical models, including NIMROD, M3D and BOUT, are available and are being further developed to study the nonlinear effects of edge MHD. DIII-D will obtain data to test these models:

- Losses of particles, electron heat and ion heat caused by an ELM will be measured over a range of conditions. This is best done in careful variations of single physics parameters, as discussed in the previous session. These losses will provide basic tests of theoretical models for ELM losses.
- We will continue to measure the evolution of edge plasma quantities during the ELM cycle. Fast 2D and possibly 3D measurements will be essential for some of this work. Data from such studies will provide more stringent tests of nonlinear models of the ELM cycle.
- We will make measurements of important parameters in extended MHD theories. In particular, we will study the role of rotation in forming the QH-mode and will test extended nonlinear theories of the peeling-ballooning theory to see if they can explain the QH mode.
- Determine the effect of nonaxisymmetric magnetic fields on the peeling-ballooning MHD limit.

Improvements in diagnostics are needed to make progress in validating models of the pedestal. The radial resolution of the Thomson scattering system will be improved since it is currently marginal for standard cases and inadequate in cases with highly localized edge particle sources. Reduction in the signal-to-noise ratio, increase in number of digitizer samples, and improved mode structure analysis software will be made for magnetics and soft x-ray diagnostics. Improvements in the edge current density diagnostics, either multi-view MSE or improved Lithium beam Zeeman effect measurements, are needed since the peeling-ballooning mode is highly sensitive to current density. A gas puff imaging system could also improve understanding of ELM dynamics in DIII-D as it has on other tokamaks. The UCSD fast camera will also be more fully exploited for understanding ELM dynamics.

6.2.4. Boundary

Two of the most pressing issues in future, burning-plasma fusion devices, such as ITER or DEMO, are the preservation of the integrity of the plasma-facing components and the minimization of the retention of tritium in the vacuum vessel. Many of the plasma parameters in the boundary predicted for ITER, e.g., the collisionless plasma conditions in the main scrape-off layer and the highly collisional divertor conditions, cannot be achieved simultaneously in current fusion devices. Developing theory and validating numerical models that simulate the physics processes in the boundary plasma against measurements from existing fusion experiments is therefore mandatory for successfully operating next-step fusion devices. In this context, the boundary plasma comprises the pedestal region just inside the magnetic separatrix and the scrape-off layer (SOL) region in the divertor and main chamber. As measurements of complex plasma phenomena in the boundary have significantly improved over recent years, several issues with boundary plasma modeling became apparent.

Within the next five years of the DIII-D experimental program we propose to address the following questions through improvements of the physics models implemented in our existing codes and their validation:

- Do the numerical tools adequately predict the measured locations of significant plasma-wall interaction and impurity production?
- Do the models adequately predict the measured particle and heat fluxes to plasma-facing components? Do the model allow robust scaling to larger fusion devices (JET, ITER)?
- Do micro-turbulence and classical drifts explain the measured SOL flows?
- How significant is chemical erosion in detached divertor conditions?
- How does carbon migrate from the source to its final deposition, and how does carbon migration affect the location of tritium co-deposition?
- How do the models explain momentum and heat removal in standard and radiative divertor operations?
- What is the effect of partial detachment of the divertor on core plasma fueling, pedestal formation and core plasma purity?

To answer these questions we will apply theory and make use of numerical tools developed within the DIII-D program as well as by our domestic and international partners. This proposal is well aligned with the tasks set by the U.S. Edge Coordination Committee and the International Tokamak Physics Agreement (ITPA) Pedestal and Divertor/SOL working groups.

Within the next five years several, state-of-the-art numerical codes will be tested against experimental data produced by an improved DIII-D diagnostic system. The suite of numerical tools includes, in order of sophistication:

- Interpretative 2D Onion-skin method (OSM) and neutral Monte-Carlo code EIRENE to empirically reconstruct the SOL plasma and to characterize the plasma-wall interaction
- Interpretative 3D code LIM to simulate carbon erosion and transport at the material surface
- Predictive 2D fluid code UEDGE to calculate particle, momentum, and heat transport in the pedestal and SOL regions and to simulate the plasma-wall interaction
- Predictive two-fluid Bragiinski code BOUT to simulate turbulent fluxes in the boundary in realistic x-point geometry
- Predictive 4D (2 in space, 2 in velocity) kinetic code TEMPEST and particle-in-cells code CPES to describe boundary phenomena in the limit of low-collisionality as predicted for next-step fusion devices

Interpretative codes have been successfully used to explore new physics and identify the controlling processes. The OEDGE code, which combines the OSM and EIRENE, is used to simulate carbon transport, deposition, and migration, most notably demonstrated for the DIII-D ^{13}C deposition studies in 2003 and 2005. The impurity Monte-Carlo code LIM was recently upgraded to include transport of hydrocarbon and carbon in 3 spatial directions. Both codes will be used to model methane injection into well-characterized divertor plasmas to advance our understanding of carbon chemical erosion and migration in fusion devices.

Steady-state heat and particle fluxes calculated with the UEDGE code were shown to be consistent with a large set of experimental data obtained in a variety of plasma conditions and magnetic configurations [Porter 2000, Groth 2005, Groth 2007]. Existing and new UEDGE simulations will be used to clarify the disparity in the heat flux scaling database recently published in the ITER physics basis report [Loarte 2007]. Together with the turbulence code BOUT, our investigations will also concentrate on better understanding of SOL flows, thereby making use of detailed measurements with reciprocating Langmuir probes and spectroscopic and imaging techniques. Strong SOL flows of the order Mach ~ 0.5 have been measured in the main chamber of several tokamaks. However, theoretical understanding of the driving terms is still at an immature stage, and even our most sophisticated models cannot robustly reproduce the direction and magnitudes of these flows (Fig. 6-4). It is expected that flows in the main chamber and divertor SOL affect the degree of divertor detachment, and hence the poloidal fueling profile of the core plasma. BOUT will also be used to simulate other complex phenomena, such as edge localized modes (ELMs), the physics of the L-H mode transition, and H-mode turbulence.

Volumetric heat flux dispersal can be aided by injection of impurity gases that radiate at conditions characteristic of the SOL plasma. This radiative divertor approach is critical for protecting the wall in future burning plasma devices, but it requires an integrated scenario combining a high-performing core plasma with a strongly radiating edge or divertor plasma. Within the DIII-D Advanced Tokamak program, high-density radiating operational scenarios have been developed using the injection of argon and impurities, has remained challenging, but preliminary results underline the importance of classical

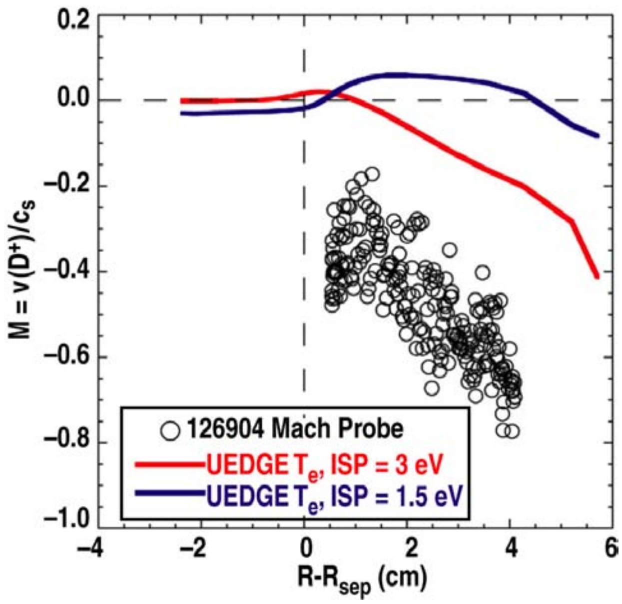


Fig. 6-4. The deuterium ion flow calculated by UEDGE in the main SOL at the top of the plasma (red line, attached inner divertor, $T_{e,ISP}$ = 3 eV) overestimates the magnitude of the measured flow from a reciprocating Langmuir probe (open black circles). The Mach speed is the ratio of deuterium ion velocity measured in the direction parallel to the magnetic field **B** to the plasma sound speed. Negative Mach speed values denote flow toward the inner divertor. Here, the Mach speed is plotted as a function of the distance from the separatrix at the outer midplane. For a detached inner divertor plasma, $T_{e,ISP}$ = 1.5 eV, more closely resembling the experimentally observations in these plasma, the deuterium ion flow is also in the opposite direction as measured (blue line) [Boedo 2006, Groth 2006].

cross-field drifts in understanding divertor detachment physics and impurity leakage out of the divertor chamber. Simultaneous inclusion of moderate impurity species (e.g., argon or neon) with carbon and hydrogen is accomplished within UEDGE, although numerical performance degrades for such large systems. Algorithm improvements and/or utilizing reduce-ion impurity models are anticipated to improve simulation throughput for these plasmas.

As the fluid description of the boundary plasma breaks down in the limit of low plasma collisionality – a plasma state predicted for the ITER pedestal region – two national efforts are presently developing kinetic codes for the boundary: the Edge Simulation Laboratory (ESL) is a partnership involving GA and LLNL, among others, in developing a continuum kinetic code, and the Center of Plasma Edge Simulation (CPES) is a SciDAC funded project developing a PIC-based code. For ESL, the TEMPEST code is beginning to perform simulations in DIII-D divertor geometry, and CPES has a similar capability. Upon successful verification of these kinetic codes we will validate them against the same sets of experimental data used for benchmarking the fluid codes.

Validation of the available numerical tools requires extensive experimental characterization of the spatial and temporal distributions of particle and heat sources and sinks, their fluxes in the pedestal and SOL regions, and recycling and concomitant impurity generation at the surrounding walls. Detailed measurements will be carried out within the next five years, utilizing DIII-D’s unique operational flexibility to control the plasma shape and conditions, including toroidally symmetric injection of recycling and nonrecycling gases for flow measurements and radiative divertor studies. Furthermore, code validation will also benefit from several new measurements, including reciprocating Langmuir probes in the outer and top main SOL and in the private flux region, fixed Langmuir probes in the main chamber at the inner wall and limiters, pressure gauges in the inner divertor, and quartz microbalance deposition monitors in the lower divertor.

6.3. VALIDATION OF INTEGRATED MODELS

Verification and validation (V&V) of integrated models is a complicated and very technical process. To quote the Fusion Simulation Project report [FSP 2007] on this topic,

"The computational fluid dynamics community has probably addressed the V&V issues over a longer period of time and with a greater degree of seriousness than any other community addressing problems comparable to those faced in plasma physics. Formal definitions for V&V concepts have been adopted by professional societies such as the American Institute of Aeronautics and Astronautics. Model validation is defined as "substantiation that a computerized model, within its domain of applicability, possesses a satisfactory range of accuracy consistent with the intended application of the model." It is important to note the highly conditional nature of the adopted definition. Codes are validated in the context of a particular problem or set of nearby problems, for a particular set of parameters, in a particular range and to a particular level of accuracy. Formally a code is not validated, but rather a particular calculation is validated. There is no unambiguous way to define 'nearby', since transitions or boundaries between regimes may be crucial and confounding. The emphasis on accuracy implies quantitative measures and attention to errors and uncertainties. At the same time, it must be understood that experimental measurements are almost always incomplete and subject, themselves, to significant uncertainties and errors. For optimum progress, simulations and experiments must be seen as complementary not competitive approaches."

Validation of a complex integrated model clearly starts with validation of the component models of which it is formed. Validation of the integrated model is made much more difficult by the large number of potential interactions between the component models. In the DIII-D program validation of the integrated model will consist primarily in evaluating the ability of the code to reproduce or predict experimental measurements. In many cases it will be necessary to modify input quantities within their experimental uncertainties to test the sensitivity of the calculations to the input conditions.

6.3.1. Comparison to Experiment Discharge Evolution

Application of integrated modeling to the evolution of experimental discharges is an important way of interpreting experiment results and planning new experiments. Success provides confidence in projecting performance of future devices like ITER. Significant progress has been made in developing and applying the integrated modeling codes. The successful demonstration of steady state AT discharges in DIII-D was obtained through strong coupling between integrated modeling and experiment. The theory-based, gyro-Landau fluid (GLF23) model with self-consistent sources and sinks within the ONETWO transport code is routinely used both for designing experiments and for interpreting their results. Repeated cycles of modeling and comparison to experimental results help develop modeling tools that are validated against experiments. The same model may then be applied to ITER simulations. The benefits of the coupling with experiment are not just validation of the model but also development of optimized approaches to the ITER steady state scenario.

Transport codes like ONETWO or TRANSP are used to help plan and interpret experiments focused primarily on the plasma core, while codes like SOL or UEDGE are used to address the boundary. The ONETWO code solves the flux surface averaged transport equations for energy, particle density, toroidal rotation, current density (or more precisely poloidal magnetic field) and equilibrium evolution using the GLF23 transport model with self-consistent source and sink calculations. The source models used for these simulations with ONETWO are the ray tracing codes, TORAY-GA for electron cyclotron heating (ECH) and CURRAY for the fast wave (FW), and the Monte-Carlo code NUBEAM for neutral beam heating and current drive. The ONETWO/ GLF23 model is validated against DIII-D experiments that aim at fully noninductive operation at high beta, as illustrated in the experimental profiles in Fig. 6-5. The predicted T_e and T_i profiles reproduce reasonably well the experimental profiles, while the toroidal rotation profile (W_{rot}) somewhat overestimates the measured rotation near the axis despite an additional *ad hoc* factor (twice the ion thermal neoclassical value) in the momentum diffusivity. The modeling has been also validated against parametric dependences in a series of steady state AT experiment on DIII-D. The ONETWO/GLF23 model also reproduces reasonably well the trend of the noninductive current components in DIII-D density scans.

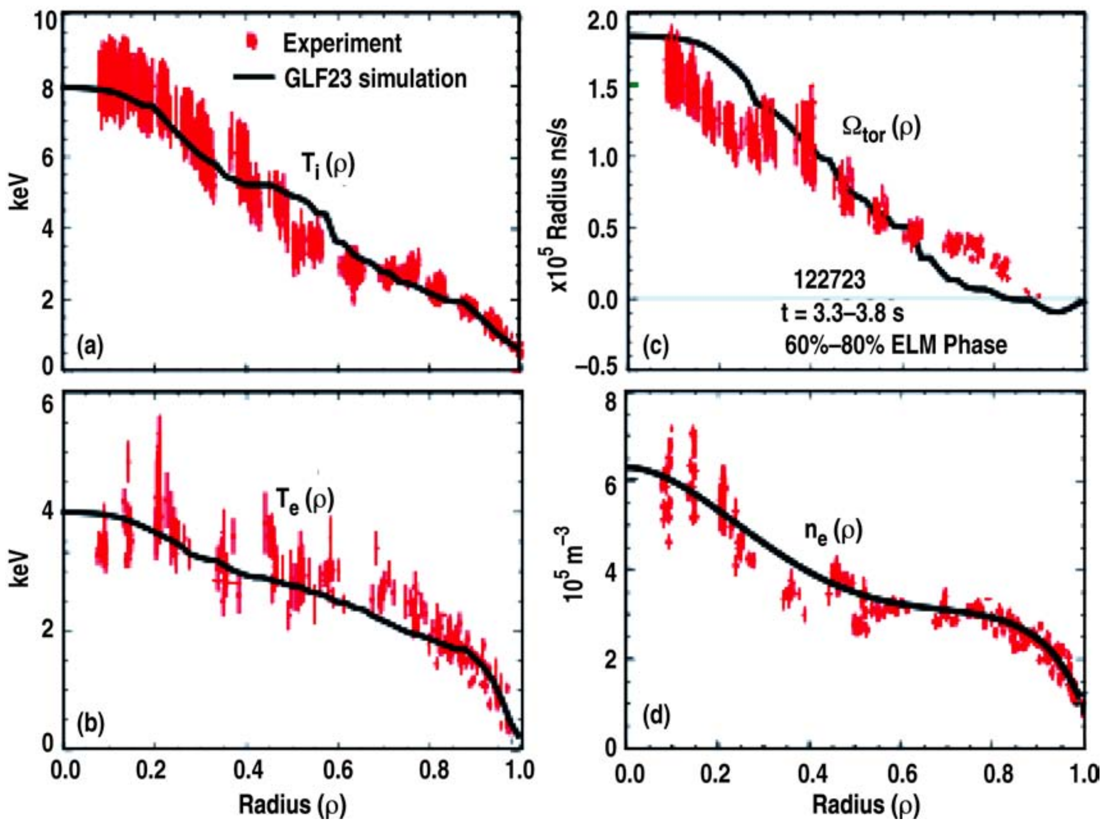


Fig. 6-5. Theory-based (GLF23) model predictions agree with experimental measurements of ELM-averaged profiles of (a) ion temperature, (b) electron temperature, and (c) toroidal angular rotation velocity. The transport simulation uses the prescribed density profile (d) and boundary conditions based on experimental pedestal values at normalized radius of 0.9.

Over the period of the proposal, the modeling capability will be extended in four key ways:

1. **Extension to the edge pedestal region.** In present modeling the density and temperature profiles in the edge pedestal region are given by experiment rather than from the model, and this has limited our overall predictive capabilities significantly. This is partly because the GLF23 model is

not applicable in the edge region. The new TGLF model will allow extending the theory-based transport modeling to the edge pedestal region. The TGLF model is also expected to provide better prediction for electron transport as well as momentum and particle transport.

2. **Addition of MHD effects.** Effects of MHD instabilities will be added. Models like GATO and DCON will be incorporated into the transport codes to address stability limits. Resistive MHD effects will be addressed through models like NIMROD and M3D to evaluate the effects of nonideal modes. For example, the equilibrium amplitude and effects on confinement of a time-dependent neoclassical tearing mode will be addressed in this manner.

3. **Extension to the entire region of tokamak through coupling of the core and the edge.** A number of theoretical models and numerical codes have been developed separately in the core region or in the edge/SOL region of the tokamak to determine the time evolution of plasma profiles on the diffusion time scale. In the core plasma region, where magnetic field lines form nested flux surfaces, the plasma temperature and density does not vary significantly on a flux surface because of strong classical parallel transport along the field lines. Thus, the plasma transport in the core region is usually described within the framework of one-dimensional transport code such as ONETWO. On the other hand, in the SOL region where the field lines are diverted into a target plate away form the core plasma, the plasma profiles are determined by the competition between the parallel and perpendicular transport, so that the plasma transport must be treated as two-dimensional. However, separate treatment of each region is not realistic because of strong coupling between the core and edge/SOL plasmas. For 2D edge/SOL transport, the core plasma conditions should be provided as boundary conditions, while the edge plasma conditions are required for the 1D core codes.

4. **Extension to the entire duration of discharge.** Integrated scenario modeling has focused on the stationary phase of discharge evolution. Predicting that a scenario can exist in the current flattop is necessary, but not sufficient, to show that it can be achieved in an experiment; an additional requirement is to determine a viable trajectory to get to that condition. AT scenarios require control from very early in the discharge to achieve the desired current profile. Therefore, it is necessary for the theory-based scenario modeling to extend simulations to the entire duration of discharge. For realistic comparison to the longer time scale of discharge evolution, it is desirable to increase the speed of the integrated modeling code significantly. Robustness and efficiency will be addressed by developing new numerical schemes and by parallelizing each module. New models such as TGLF may need more computing time. Moreover, some of the individual models may not be coupled directly to the discharge evolution code due to their enormous demand on computational facilities. Therefore it is necessary to develop reduced models based on the full-physics simulations.

The 2D plasma transport code "C2 (Coupled two-dimensional)" will be introduced to DIII-D to calculate the time evolution of the plasma density, temperature, and parallel flow, along with self-consistent diamagnetic and ExB drifts in the tokamak core, edge pedestal and SOL. The unified 2D numerical descriptions across the entire cross-section of the tokamak enables investigations of the strong interactions between regions of the plasma in a self-consistent way. Furthermore, a unique aspect of the

C2 code is that it is able to address the 2D characteristics of the core plasma that can be exploited to study the appropriateness of the flux-surface averaging, e.g.,

- ExB shear (Hahm-Burrell versus Waltz model)
- In-out density asymmetry
- Stiffness in transport (role of parallel transport)
- 2D neutral penetration
- Ballooning nature of turbulent transport.

6.3.2. MHD Stability and Transport

Many important aspects of tokamak physics involve core and edge processes in which MHD instabilities interact with plasma transport to strongly impact the plasma performance. These include interactions between ELMs and transport that play a key role in setting the pedestal height and width, interactions among RWMs, error fields, and plasma rotation, and sawtooth oscillations and AEs and their interactions with energetic particles. To develop a predictive understanding of these interacting processes, in addition to validation of individual models as described in the previous section, it is important to also validate the integrated physics models and simulation codes.

As described in Section 6.1, an integrated computational model to address edge pedestal physics issues will be developed by combining the state-of-the-art turbulent transport model TGLF with models of the ELM trigger based on calculations using the ELITE peeling-ballooning stability code within the integrated modeling framework provides by the EFIT and ONETWO equilibrium and transport codes. First, predictions from this integrated modeling package will be compared against discharges from DIII-D ELM experiments. Then, new experiments will be designed to test specific features of the predictions. Improvements to the models will then be made if necessary based on the results of the comparisons and tests. For discharges in the QH-mode regime, simple models of the MHD edge mode saturation and its resulting transport will be developed that can be directly validated against DIII-D experimental observations.

Similarly, in the RWM and energetic particle areas, predictions from the MARS-F/ONETWO and NOVA-K/ORBIT/ONETWO integrated modeling packages will be validated against discharges from DIII-D RWM and energetic-particle experiments.

6.4. IMPLEMENTATION OF MODELS

Computational models of plasma behavior are of greater value when they are accessible to and usable by all interested staff. In the past, it has been observed that codes that are targeted to be operated primarily by the code developer frequently fail to obtain broader usage due to a steep learning curve. In order to have our models reach a broader audience and therefore reward the investment in time and resources in developing codes, efforts will continue to be made to simplify the use and interpretation of the models. Therefore, tools will be developed to allow efficient set up of code runs and to visualize results in ways that promote physical understanding.

6.4.1. Access to Models

It is important to provide the general staff with the ability to run the key codes in an efficient manner; that is, without the very lengthy learning curve that is usually required to run codes correctly. Most codes have a very lengthy list of control parameters that are nontransparent and confusing. In many cases it is possible for an expert to determine sets of these parameters which will be satisfactory for most users, and then the user need only choose his or her general objectives in order to obtain a satisfactory result. In the present five-year contract, this process has been applied to the ONETWO transport code, which is now run routinely between shots for the two purposes of guiding the experiment (that is, what to change for the next discharge) and for providing a database of information for later analysis and use. A graphical user interface was also developed for post-shot analysis using ONETWO using different inputs (different ways of determining the kinetic profiles, for example), and both the between-shot and the post-shot ONETWO results are stored in MDSplus for later use. In this way, users can access transport code runs without having to know details about the running of the code. Similar interface techniques were applied to operating TRANSP on FusionGrid as part of the National Fusion Collaboratory Project with a resultant increase in user-base.

In the near term, the TGLF (Trapped Gyro-Landau Fluid) drift wave model will be made available in a similar way. This model approximates GYRO calculations but with much smaller investment in computations. It is anticipated that the usage of codes to address stability and energetic particle modes can be simplified with similar techniques and work will start in these areas after the TGLF effort is completed.

A method to increase the reach of computer models beyond the DIII-D program is to offer them as network accessible computational services on FusionGrid. This eliminates the need to have new users install models locally or to run models from DIII-D computers requiring running the X display protocol over ESnet that is rather slow. Both ONETWO and GATO have been prototyped as FusionGrid computational resources and this work will be completed. Additional codes, like ELITE and DCON for example, will be addressed as required.

6.4.2. Data Visualization and Interpretation

The physics models are aimed at advancing fusion science through development of testable predictive computer codes. Critical is the ability to assess the reliability or accuracy of these models. Assessment of predictive models is divided into two distinct activities: verification, which assesses the degree to which a code correctly implements the chosen physical model; and validation, which assess the degree to which a code describes the real world. Since confidence in the ability to predict is ultimately based on code performance against experimental data, a vigorous and ongoing validation activity is a critical component of this effort. To facilitate this work by the staff, a software infrastructure must be available to allow this to take place in a seamless and efficient manner together with visualization capability and access to experimental data as appropriate.

Analyzed experimental data on DIII-D is stored in a distributed MDSplus data acquisition and management system. Examples of analyzed data include distillation of raw diagnostics measurements into quantities of physical interest (e.g. density and temperature from Thomson Scattering) and the synthesis of numerous measurements into a coherent picture (e.g. power balance analysis from ONETWO between pulses). For the models developed in this effort, a substantial fraction of their output will be available in MDSplus. Although all data can be stored in MDSplus, certain file-based quantities that may not be of critical interest to a broad audience and thus their storage in MDSplus is not required. Having the same MDSplus API (application programming interface) to retrieve data for both experimental and model data will greatly enhance the ability to perform model validation. Additionally, synthetic diagnostics will be developed for the stored model data that allows for calculation of the results that the various plasma diagnostics actually measure (Fig. 6-6).

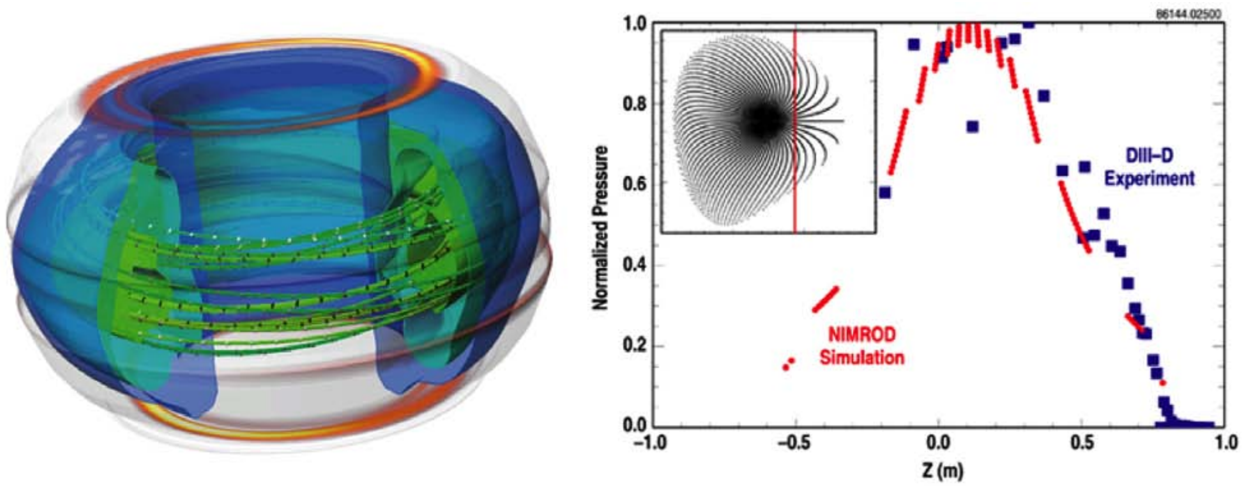


Fig. 6-6. Synthetic diagnostics can facilitate model validation. In this figure, a full 3D MHD simulation of DIII-D by the NIMROD code is diagnosed to compare to the measured pressure profile from the Thomsen Scattering diagnostic.

Visualization of model results and their comparison to experimental data is another critical component of this activity. Building on the successful IDL-based tools like EFITViewer, ReviewPlus, and ProfileViewer, new model specific visualization tools will be rapidly developed as required by utilizing the GAPlotObj graphics library. Requirements for more advanced 3D visualization will be satisfied by deploying OpenGL graphics tools as has been done to display the real-time EFIT on the control room display wall. Utilizing OpenGL allows for more rapid visualization of complex imagery through the utilization of hardware-accelerated graphics.

6.4.3. Interface with Plasma Control System

The DIII-D plasma control program during the 2009–2013 period will focus heavily on model-based control design. A substantial number of the necessary physics-based models required for this effort are expected to be derived from integrated modeling (IM) efforts. In general, these models will be “control level” models, which are typically simple, often linearized, allow rapid optimization or iteration, and can run in a complex integrated simulation with minimal incremental demand on cpu cycles. For example, low-order representations of profile evolution dynamics will be essential for profile control design. Simplified forms of transport equations making use of flux averages and/or various simplifying assumptions, as well as numerically reduced systems generated by 1.5D transport codes, could be used to generate such models.

In addition, key enabling actuator models will be developed under the IM effort. Control level models of ECH/ECCD derived from TORAY-GA or other codes will enable real time determination of current deposition locations and will be usable in control simulations. Reduced versions of neutral beam codes will produce similarly useful modules for control simulation.

A systematic approach to generation, validation, and maintenance of such models as a key product of the larger IM mission at DIII-D will provide a model for next-generation devices such as ITER that must rely heavily on this approach to control design.

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7. INTEGRATED PLASMA CONTROL

As the world fusion community approaches the goal of achieving reactor grade self-heated plasmas in ITER, high performance plasma control plays an increasingly essential role. Virtually all new advances in plasma physics must be accompanied by new and enabling control techniques. Advanced tokamak regimes, characterized by operation beyond various open loop stability limits, are particularly demanding of control advancements, and continue to motivate DIII-D to maintain its leadership role in plasma control [Ferron 1992, Walker 2003, Humphreys 2003, Humphreys 2007].

The DIII-D integrated plasma control program will continue to maintain DIII-D at the forefront of control development among tokamaks worldwide, and will support the rapidly expanding requirements of the DIII-D experimental program. The 2009–2013 experimental period will feature a greatly increased focus on model-based control design, state-of-the-art control research, and the development of transferable solutions for ITER and other next-generation devices. This aggressive program of focused plasma control research and experimental application of novel control solutions will ensure that algorithms developed at DIII-D will continue to support and energize tokamak control around the world, and illuminate the path to a robust and reliable advanced tokamak fusion reactor (Table 7-1).

Table 7-1
Integrated Plasma Control Program Elements and Initiatives

Program Elements and Initiatives	2009–2013 Plans
TokSys Control Modeling/Design Environment	• Routine use of TokSys for model-based controller development • Integrated boundary/profile control • Divertor performance control
Enhanced Physics Operations	• Use of TokSys design/simulation tools to reduce need for experimental time to develop and tune control gains • Support of physics operations in next-generation devices by DIII-D personnel
Advanced MHD Instability Control	• NTM control advances (steerable launchers, modulation phase/frequency control) • Model-based RWM control • Active ELM regulation • Realtime use of 3D magnetics, MHD spectroscopy
Integrated Off-Normal Event/Disruption Detection and Response	• Disruption avoidance/response • Demonstrate complete physics-based, layered-response ITER solution for general fault handling • Demonstration on EAST, KSTAR
Collaborations	• Joint development of ITER control solutions with other U.S., international tokamaks • Expanded collaborations with superconducting tokamaks including KSTAR, EAST • Expertise in academic/professional control community leveraged for fusion solutions
Facility Modifications and Exploitation for Control Research	• Modification of PF circuit to enable individual control of each coil in DIII-D • Use of DIII-D as emulator/testbed for superconducting tokamak control solutions • Training of Physics Operators using DIII-D PCS at many tokamaks

Integrated plasma control initiatives supporting this vision include further development and dissemination of the TokSys model-based control design environment, design and implementation of advanced controllers on DIII-D and other devices using the DIII-D Plasma Control System, and expanded collaborative control efforts with other tokamak teams and experts in the academic control community.

7.1. PROGRAM OVERVIEW

7.1.1. Mission

Integrated plasma control (IPC) efforts at DIII-D have two key goals: (1) to enable and support DIII-D physics goals through continual improvements in control; and (2) to advance the state of the art in plasma control to prepare for and support ITER and other next-generation devices. The first goal includes isolation and elucidation of physics processes, achievement of well-controlled advanced operating regimes, exploration of new physics, and improvement of DIII-D performance, operational flexibility, and scientific productivity. The second goal includes supporting and extending the use of the DIII-D PCS and control design tools at tokamaks worldwide, as well as development and demonstration of control solutions essential to ITER in a manner that allows inter-machine transfer.

7.1.2. Program Elements

The IPC effort at DIII-D includes a variety of programmatic and technical elements that presently allow DIII-D to lead the world in many areas of plasma control. The DIII-D PCS provides a flexible and highly scalable platform for implementing control algorithms of arbitrary complexity, and is presently in operation on six tokamaks worldwide. Highly skilled physics operators (responsible for programming discharge scenarios in the PCS) constitute another essential element of the program. A team of specially trained physicists rotate between their usual physics responsibilities and time spent supporting experimental control needs. A set of computational tools for modeling, algorithm design, and simulation, collectively known as TokSys [Humphreys 2007], is another key element of the IPC program. Use of physics-based models allows the same tools to be applied to many devices, and facilitates cross-machine validation. Another essential element of the DIII-D IPC program is the extensive set of active collaborations between DIII-D and other experimental programs, as well as with other laboratories with specialization in various areas of plasma control. An important and growing portion of these collaborations is the work with academic control mathematics specialists at universities both in the U.S. and internationally.

7.1.3. Summary of Initiatives for 2009–2013

The 2009–2013 period will focus in particular on a set of key plasma control initiatives which will continue to support evolving DIII-D control needs, bring the use of model-based control to full operational maturity in many areas, support and extend the use of the DIII-D PCS at tokamaks worldwide, and provide validation on many devices to enable eventual transfer of control solutions to ITER with high confidence in predicted performance. Development of the TokSys environment will be a key unifying effort, providing the mechanism for developing and validating control models, implementing model-based design, and enabling routine verification of controller performance before experimental use. As a common environment for models of every device sharing the DIII-D PCS, as well as ITER and many other tokamaks, TokSys also provides a mechanism for control model and control algorithm transfer among all of these devices. Physics operations at DIII-D will be enhanced by expanded training of operators in the use of simulations for verification of controller performance before experimental use. The increased expertise in and reliance on such simulations will provide a demonstration of the operations approach taken by the present ITER CODAC design. Specific advancements in control will include modifications to the DIII-D PF coil circuit to improve flexibility, as well as a wide array of new control algorithms supporting advanced MHD stabilization, profile control, fault detection and response, and new scenario control requirements.

7.2. PRESENT STATUS

7.2.1. DIII-D PCS

The DIII-D digital plasma control system (PCS) continues to be a key tool for tokamak physics research [Penaflor 2004, Leuer 2005]. The PCS is responsible for acquiring appropriate realtime sensor data and controlling and coordinating all of the actuators used to produce the plasma, including coil power supplies, heating and current drive sources and gas valves. Improvements to the PCS during the present five-year period have enabled a wide variety of new experiments. In addition, the PCS software infrastructure and hardware design concepts support versions of the PCS at other devices in the U.S. and around the world.

Upgrades that have been made to the hardware and real-time data acquisition capability of the DIII-D PCS include:

- Acquisition of electron density and temperature profiles from Thomson scattering.
- Acquisition of ion and temperature and toroidal rotation profiles from charge exchange recombination spectroscopy.
- Acquisition of a 32 channel electron temperature profile from ECE.
- Total number of data signals acquired in real time increased to 500.
- Increase of the resistive wall mode feedback control command update frequency to 90 kHz enabling stabilization close to the ideal wall limit.
- The real-time CPU power was more than doubled to greater than 20 CPUs.
- Implementation of a real time display of the discharge important waveforms and the plasma boundary.

7.2.2. Physics Operations

Today DIII-D has a dozen trained Physics Operators who are drawn from all areas of the Experimental Sciences program and include several operators who are on long term assignment from collaborating institutions in addition to GA staff. The principal duties of Physics Operators involve programming the PCS to accomplish the physics goals of an experiment, but extend to monitoring, communication, and control of other subsystems such as neutral beams. Closely coupling Physics Operators with experimental teams early in the experimental planning stages has led to consistent excellence in planning and executing cutting edge tokamak experiments. Each experiment has a team of two Physics Operators, enabling higher productivity as problems and situations are more rapidly dealt with, errors are reduced, and the daily shot count is increased. DIII-D Physics Operations has begun to train staff members from other institutions using the DIII-D PCS in understanding diverted plasma control with this common control platform. Increased use of TokSys design tools and simulations by Physics Operators will reduce the need for experimental time to empirically develop control gains during operations.

7.2.3. Experimental Control Achievements

By taking advantage of an expanding set of control actuators and a powerful PCS, during the 2004–2008 period DIII-D has continued to lead the world in the breadth of application of plasma control to experiments. Key achievements in experimental plasma control during this period include [Ferron 2006, La Haye 2005]:

- Addition of Motional Stark Effect (MSE) data in real time equilibrium reconstruction for real time safety factor (q) profile determination
- Real time control of the central or minimum safety factors using ECH, ECCD, and neutral beams
- Resistive wall mode (RWM) suppression using the internal coil (I-coil) set and audio amplifiers
- Neoclassical tearing mode (NTM) suppression using electron cyclotron current drive (ECCD)
- Simultaneous regulation of beta and rotation using new counter-injection beam supplied during the Long Torus Opening Activity of 2005–2006, real time CER measurements of rotation profile, and calculation of plasma beta from real time equilibrium reconstruction
- Studies of RWM stabilization in low rotation regime
- Control of Te at two locations simultaneously using ECH
- Demonstration of advanced model-based multivariable shape controllers
- Control of the internal inductance using the plasma current ramp rate
- Real time spectrum analysis of magnetic fluctuation measurements
- Coupled nonlinear and complex logic structures in fault detection algorithms
- "Dud" detector system allowing identification of a variety of conditions in real time which render a discharge uninteresting or useless from that point on in the pulse
- General fault detection system allowing flexible switching to different control programs for appropriate fault response
- Device-specific versions of the DIII-D PCS used in NSTX, Pegasus, and EAST operations

Figure 7-1 illustrates an experiment in which ECCD was aligned with a saturated NTM island via real time control, suppressing the mode. Combined use of real time q-profile reconstruction and alignment control enabled the suppression to be sustained for as long as the gyrotron power was on.

7.2.4. TokSys Development

A collection of Matlab functions known as TokSys (for Tokamak System Toolbox) has been developed at GA over several years of work with DIII-D and many existing or proposed fusion devices. This toolbox automates most of the process of generating models and simulations for tokamak and plasma poloidal field systems. It is presently in constant use for development and testing of plasma control algorithms in DIII-D, EAST, and KSTAR, and its use is maturing on NSTX and Pegasus. The TokSys toolbox provides a common framework, toolset, and set of data structures among the multiple devices, allowing control designs developed for one machine to be easily transferred to any other and allowing improvements in physics models discovered while working with one machine to be instantly available to all others. TokSys models and design tools for ITER allow validated models and controller design approaches for these operational devices to be applied to ITER with high confidence. The models developed using this toolbox are also tightly coupled to the variants of the DIII-D PCS used on several tokamaks via the simserver (simulation server), which provides a method for running the actual PCS software in closed-loop simulation with a simulation of the device to be controlled. The simserver provides a method for validating both the effectiveness of the control algorithm and the correctness of its implementation in the real-time control code prior to use in experiment.

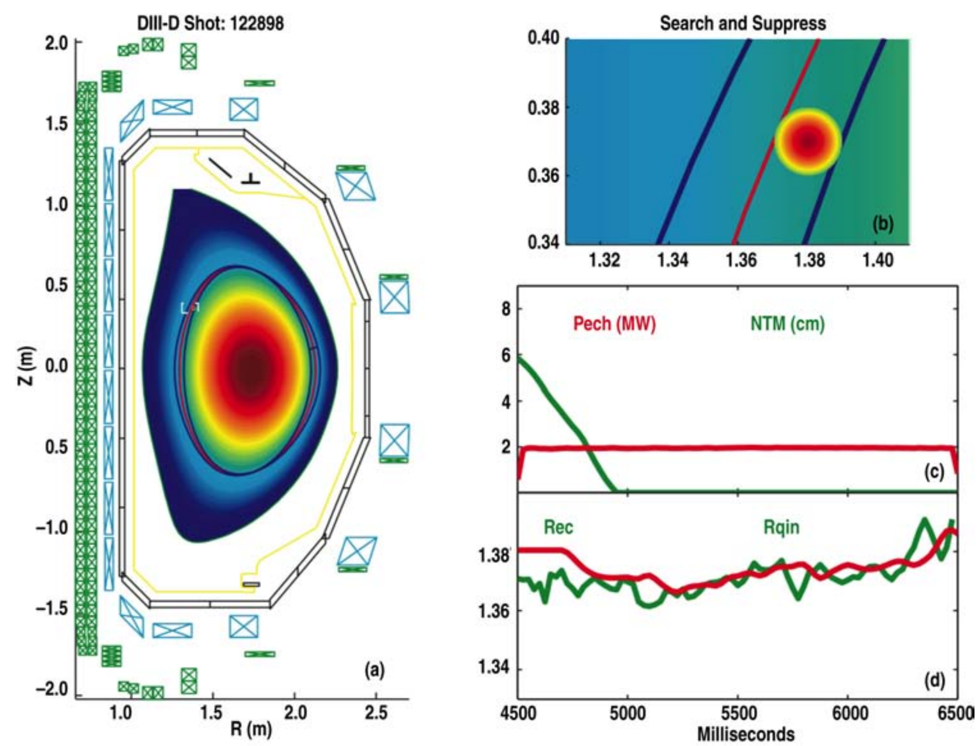

Fig. 7-1. Neoclassical tearing mode control in DIII-D aligns the electron cyclotron (EC) current drive spot (top right, and Rec in bottom right) with the magnetic island (top right and Rqin in bottom right) by varying the toroidal magnetic field, and thus the EC resonance location. After the island width is reduced to zero (NTM, bottom right), Rec continues to be dynamically varied to preserve the alignment and keep the island suppressed [Humphreys 2006].

7.2.5. National and International Collaborations

Several new versions of the DIII-D PCS were developed and implemented at the NSTX and Pegasus spherical tori in the U.S., as well as at the MAST spherical torus and the EAST and KSTAR tokamaks internationally. Development of the NSTX PCS resulted in routine experimental use of realtime equilibrium reconstruction (RTEFIT) with boundary control. The EAST PCS enabled the successful 2006 first plasma campaign and a subsequent elongated/diverted plasma campaign in the EAST superconducting tokamak. The KSTAR PCS is in the final phases of development prior to first plasma startup, expected in mid-2008. Each of these collaborations has benefited the DIII-D program in different ways. NSTX collaboration has given rise to greatly extended TokSys PF circuit modeling tools, as well as invaluable validation of plasma response codes. Success of model-based designs of breakdown and startup scenarios for EAST has served to validate many design tools in the TokSys suite. A general fault detection architecture and specific algorithms implemented in both the EAST and KSTAR PCS codes extends the capabilities of the basic approach originally taken in the DIII-D PCS and provides valuable experience specifically applicable to ITER.

Collaborations between DIII-D and other experimental programs have also contributed to control efforts at other devices, and led to fruitful enhancements of DIII-D control. A long-term collaboration with JET has placed DIII-D profile control experts on teams performing internal transport barrier and advanced scenario experiments on JET, and brought EU experts to participate in DIII-D experiments. Early approaches at JET on model-based design for neutral beam regulation informed the design of integrated plasma rotation and beta control at DIII-D.

Significant progress has been made in expanding collaborations within the mathematical and engineering control communities in the U.S. and abroad. Specific collaborative efforts in control design developed with Chalmers University, Lehigh University, and UCSD have produced valuable solutions in areas from resistive wall mode controllers to profile control algorithms. A workshop on tokamak plasma control held at GA brought together control community experts with fusion control specialists. A satellite meeting held at GA on the occasion of the IEEE Conference on Decision and Control enabled plasma control specialists attending the conference to interact at a more focused technical level than afforded by the conference. Two special issues of the IEEE Control Systems magazine were co-edited by DIII-D personnel to describe the range of topics in tokamak fusion of potential interest to the control specialist community.

7.2.6. ITER Control

Much of the focus of experimental control efforts on DIII-D over the last several years has been on ITER-relevant solutions, including NTM, RWM, axisymmetric control, and disruption mitigation [Humphreys 2007]. The DIII-D program has also continued to contribute to ITER control design and analysis in various ways. Wherever possible, a philosophy of physics-based model development and assurance of transferability to ITER has been followed in implementation of DIII-D control solutions. In addition, solutions developed for DIII-D control have been chosen to specifically align with ITER CODAC requirements. For example, the TokSys environment represents an explicit candidate solution for ITER model-based design. A facility much like the DIII-D “Simserver” simulation, which can connect to versions of the DIII-D PCS in order to confirm control function and performance, is specifically required in the ITER CODAC design as well. DIII-D control experts have consistently participated in CODAC design efforts, and have contributed to various ITER control assessment tasks in the present period. The DIII-D control program continues to play a leadership role in the U.S. portfolio of control resources, including in recent years participation in the leadership of the Burning Plasma Organization plasma control group.

The DIII-D program has also benefited from ITER-driven collaborations. Collaborations with international teams engaged in ITER control tasks have enhanced DIII-D control tools and contributed to new PCS algorithms for DIII-D. Close engagement with the goals and requirements of ITER control has served to focus design choices made for DIII-D on many occasions.

7.3. INTEGRATED PLASMA CONTROL INITIATIVES 2009–2013

7.3.1. Overview of Vision

The 2009–2013 period will be characterized by an increased focus on model-based control design in DIII-D, preparing for the essential role this approach must play for ITER and other next-generation devices. DIII-D will increasingly demonstrate the routine application of such controllers with minimal need for empirical tuning, which has traditionally consumed substantial experimental time. Wherever possible, new control designs prepared in support of DIII-D and next-generation device needs will make use of physics-based models, and the supporting tools developed will continue to be made generic so as to maximize applicability to all machines being analyzed. We use the term “integrated plasma control” to refer to a particular systematic approach to plasma control design that uses models implicitly taking into account the relevant coupled interactions in the system. The approach is characterized by using validated physics-based models to design controllers, and verifying controller performance specifically by operating actual realtime implementations against sufficiently detailed simulations. This systematic process, illustrated in Fig. 7-2, has been under development for several years at DIII-D. IPC will provide the framework for model-based design at DIII-D and is expected to be the approach followed for all ITER control. Following this framework in the 2009–13 period will entail a significant reliance on development and validation of control-level models generated within DIII-D Experimental Science and with collaborators.

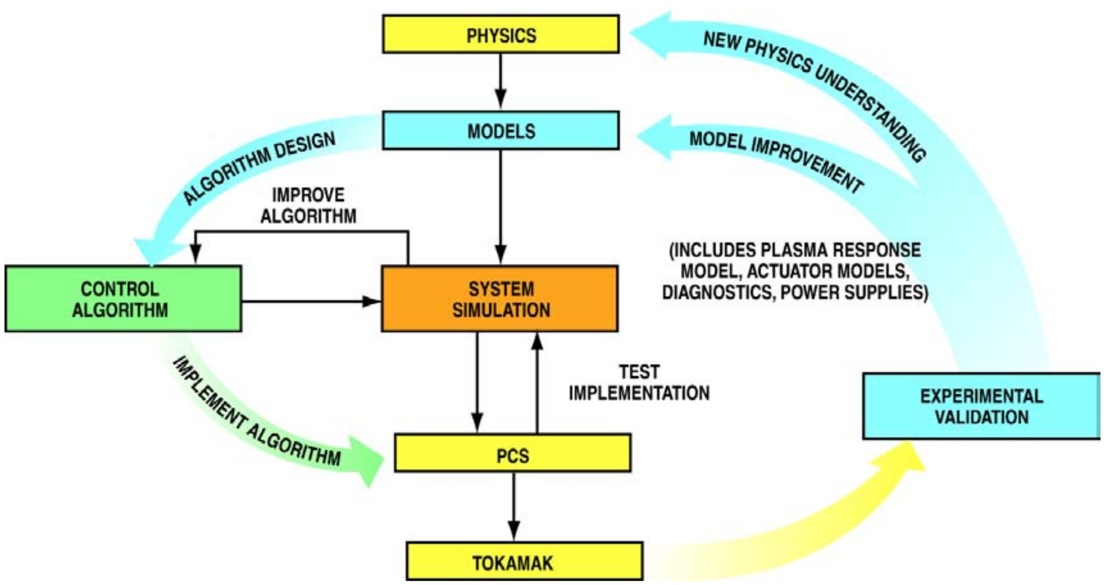

Fig. 7-2. The integrated plasma control design process uses validated physics-based models to construct control algorithms, and verifies actual control system performance against detailed simulations prior to operational use.

Application of systematic design will fuel an aggressive program of control development in support of the DIII-D experimental program, with a key mission of demonstrating integrated control of advanced tokamak (AT) scenarios. Because the advanced tokamak regime is characterized by operation near or beyond challenging stability limits, there must be a heavy reliance on robust, high performance active control. The highly coupled complex responses of AT plasmas require high performance integrated controller designs based on accurate models. Advancements are particularly essential in current and pressure profile regulation, and control of a variety of key instabilities (e.g., RWM, NTM, and ELMs). An AT operating in a burning plasma regime will critically require a high reliability fault response system, incorporating effective fault proximity detection, intelligent corrective action, soft landing responses, and damage mitigation. Prediction, detection, corrective response, and mitigation of disruptions will be a major control research focus of this planning period, coupled with the disruption physics effort.

7.3.2. TokSys: Integrated Tools for Control Design

The objective for TokSys development in the next five-year period is to establish routine general use of systematic, model-based design for development of controllers on DIII-D and other devices that use the DIII-D PCS. Tools for modeling and control of nonaxisymmetric instabilities, for example, RWM, NTM, and disruptions, will continue to be developed. In addition, models and control design tools for previously separate control problems will be integrated to enable development of controllers that can coordinate use of various types of actuators to simultaneously control multiple plasma characteristics.

Universal attention to validation is an essential requirement of model-based controls, since a model must reproduce the plasma response for a model-based controller to be effective. Existing approaches to experimental model validation will be extended, systematized, and applied to data from multiple devices.

Use of simserver simulations will be expanded. They will be used to routinely support experimental design as well as testing of implemented controllers. The simulations themselves will also be expanded to allow simultaneous simulation of multiple subsystems and actuators, enabling the testing of algorithms that perform coordinated control of multiple plasma characteristics.

7.3.3. Enhanced Physics Operations

DIII-D physics productivity will be significantly enhanced through the use of TokSys and Matlab/Simulink tools to develop and test controllers prior to operational application. These tools will allow the Physics Operations staff to thoroughly test and verify new control schemes before commissioning them into service. These tools will also enable the training of physics operators, both experienced and new, in how to use new control algorithms effectively rather than having to gain this knowledge solely in DIII-D operations. Preparing for new experiments, especially new shapes, can be done with the simulator tool set, reducing the necessary development time on the tokamak. Gain tuning, boundary control locations and limit tradeoffs can be optimized off-line rather than on the tokamak. This will raise the productivity of the device by shunting much of the physics operator’s learning curve onto the simulator tool set and minimizing the use of experimental time for control optimization.

7.3.4. Advanced Axisymmetric Control

A primary objective during the next five-year program is to demonstrate control solutions on DIII-D that enable regulation of AT plasmas in steady-state operation. DIII-D can explore and demonstrate many of the solutions required for AT operation because of its substantial diagnostic and control actuator capabilities and sophisticated digital plasma control system (PCS). In addition, many of the operational issues that limit shot length can be addressed, including evolution of plasma profiles, coil current proximity to current limits necessary to optimize plasma shape, nonlinear responses due to nonlinear or limiting actuators or to profile evolution, various types of system faults, and other aspects of the control that currently require between-shot intervention by skilled operators.

In order to meet increasing demands for control performance and speed of algorithm production, a series of developments of model-based controllers and plasma parameter estimators have been undertaken over the past few years [Walker 2005]. These methods have combined a wealth of existing control technologies with increasingly mature physics understanding of how plasma equilibria evolve and respond to external actuation. Model-based controllers are expected to be in routine use for DIII-D equilibrium control by the beginning of the 2009–2013 period. The challenges remaining will be to simultaneously make them sufficiently robust so as to accommodate evolutions in plasma profiles, prevent current limiting, and handle appropriately and in real-time the various types of nonlinearities and system faults, all of which are necessary to preclude operator intervention and thereby demonstrate their applicability to steady-state control. Removal of a presently existing bus constraint on the PF-coil circuit connections will accelerate this effort significantly (Section 7.4.2). This constraint forces the sum of currents in the set of inboard shaping coils to be zero, effectively decoupling those coils from the ohmic coil. While this circuit minimizes the need for power supplies on shaping coils, it also complicates the use of high order multivariable controllers, and makes the DIII-D PF system very different from all of the long-pulse superconducting tokamaks coming online in the near future. By combining the proposed power supply and inboard-coil bus constraint upgrades with its advanced model-based control design capability, DIII-D will be uniquely positioned to develop and demonstrate controllers for these devices and fully exploit ongoing and active collaborations in plasma control expected to grow in the 2009–2013 period.

7.3.5. Integrated Operating Point Control

Optimization of performance in long pulse, steady-state plasmas in a specified advanced operating regime will require further advances in control of the basic discharge parameters. Presently, parameters such as the discharge shape, plasma rotation, beta, plasma current and electron density are regulated with independent feedback controllers. Improvement of control will require consideration of the way these various parameters interact with each other; this is a key aspect of integrated operating point control.

Integration of operating point control will involve model-based controller design leading to multiple input, multiple output control algorithms. Techniques for implementation of this type of controller will be investigated. In addition, the influence of global parameters such as the discharge boundary shape on local parameters such as the H-mode pedestal height and the resulting ELM characteristics will be considered in controller design. Opportunities for control of this type of local parameter will be explored.

Other physics studies planned for DIII-D may lead to techniques that need to be taken into account for feedback control of the operating point. An example is the application of nonaxisymmetric magnetic fields for the control of ELMs. These external fields have been shown to affect, for example, the plasma density and toroidal rotation.

7.3.6. Profile Control

A principal focus of the DIII-D physics program is the study and demonstration of the AT regime and its application to steady-state high beta operation leading to an economically attractive power reactor. A key component of the AT regime is safety factor and pressure profiles that are consistent with 100% noninductively driven current at high beta. Active control of these profiles is thought to be necessary in order to optimize AT performance. Thus, a key area of research will be the use of the available actuators at DIII-D (ECCD, ECH, FWCD and neutral being heating) for profile control. Physics models will be developed to enable design of high order integrated controllers to regulate these actuators. Collaboration of experts in the area of controller design will be encouraged in order to facilitate this work.

A more extensive description of profile control physics in AT operating regimes in DIII-D is provided in Section 3.4.

7.3.7. Advanced MHD Stability Control

The DIII-D experimental program will continue its focus on AT operation, with its attendant reliance on high performance MHD stability control. Suppression of NTMs will be enhanced by the addition of steerable launchers along with corresponding control algorithm improvements. The modulated ECCD capability under development now and the addition of realtime steerable launchers will make DIII-D a model for the ITER configuration. Along with the increased emphasis on controller design using physics-based models, the ITER-like configuration will allow solutions to be directly transferred to ITER. This direct transferability will also characterize improved RWM control solutions, increasingly based on validated models which can be applied to ITER, and allow calculation of controllers using the same validated control-level codes. ELM control solutions will be integrated with overall operating point control and will be similarly developed for direct transfer to ITER.

An enhanced diagnostic set providing high resolution 3D magnetic measurements will enable 3D equilibrium calculation and an increased use of nonaxisymmetric fields in control solutions. MHD spectroscopy will play an increased role in active stability determination for use in control and fault response algorithms.

7.3.8. Integrated Off-Normal Event/Disruption Detection and Response

An explicit initiative of the DIII-D experimental program will be the demonstration of solutions required by a reactor for disruption-free operation (Section 3.5.2). This goal will entail research in realtime disruption proximity determination, intelligent corrective response, and robust disruption avoidance including soft-landing approaches, rapid shutdown, and disruption effects-mitigation. Sophisticated computational approaches will be required to achieve reliable realtime disruption proximity prediction, including realtime stability calculation and MHD spectroscopy approaches. Next-generation approaches to disruption mitigation may include study of rupture discs, multiple sources and new forms of massive impurity injection, mixed impurity species, or combined use of nonaxisymmetric magnetic fields and impurity injection.

To maximize the effectiveness and simultaneously guarantee the safety of a fusion device, an integrated fault response system which includes disruption avoidance and response is required. Such a system must provide a layered response to handling off-normal events; to try to continue the shot, if it makes sense to do so; to perform a controlled shutdown, e.g. ramp the PF and plasma currents to zero, if it is safe to take the time to do this; to perform a more rapid termination if danger to the device or personnel does not allow for a fully controlled shutdown; or to take action to mitigate potentially damaging effects of an already-developing disruptive plasma termination. Development of this fault fault/disruption response system will be a key initiative of the Integrated Plasma Control Program.

The DIII-D PCS incorporates a highly flexible architecture for fault detection and response, which finds a variety of experimental uses at DIII-D and other devices using the PCS. However, most currently operational methods for off-normal event detection and response, at DIII-D and elsewhere, are rather simple. Existing systems are also far from comprehensive and therefore a solution for the critical ITER need has never been demonstrated. We plan to implement a larger set of fault detection and response algorithms in the DIII-D PCS, including more sophisticated methods such as an algorithm for disruption detection and response. This work is intended to provide the operational experience and further development of the software architecture needed to provide a complete solution for ITER. Devices such as DIII-D with high robustness to disruption effects can serve to validate the methodology without risk to the device. Long pulse devices like EAST and KSTAR can similarly serve as test beds, because they derive immediate benefit from such solutions.

7.3.9. Leveraging Resources of Worldwide Fusion Control Community

The size of the worldwide fusion control community has been steadily growing over the last 15 years, fueled over the last decade by the imminent reality of ITER. The DIII-D program is closely involved with this community and expects to increasingly leverage active collaborations with other fusion institutions with strong control activity, as well as with academic control specialist institutions. The 2009–2013 planning period represents a critical time for this initiative, as the world fusion program begins a gradual evolution toward a mission-oriented, engineering-dominant, commercially viable alternative to present forms of energy production. The demands of the ITER design itself have already begun to illustrate the need for advanced control solutions, requiring the best techniques and finest minds from this powerful and mature field. The exciting growth in the worldwide portfolio of next-generation, long pulse steady state tokamaks also motivates the increased involvement of control specialists, as these devices pose many of the difficult control problems soon to be encountered by ITER. A focused effort to draw in the collaborative involvement of nonphysics institutions at the forefront of the control field will be needed in order to realize this goal.

In addition to the fundamental benefits of exploiting superior expertise to reduce the cost of development of control algorithms, a greater coupling between DIII-D and other control-intensive institutions will benefit the program in many other ways. Establishment of common fusion control standards can improve the sharing of models, codes, and control tools. Attention to interoperability and maintenance will improve the ability to share actual algorithms from machine to machine. Common access to grid-based computational resources will improve the ability to validate models on many devices and increase confidence in applicability in control regimes beyond those attainable in a given machine.

Remote operation of next-generation devices from DIII-D will serve to tightly couple the operations and control expertise and resources of DIII-D with those devices, and transfer knowledge directly back to the DIII-D program. Direct experience with superconducting long pulse operations will serve to focus the DIII-D program to provide better, specifically usable solutions to ITER and DEMO. A robust collaboration among these machines involving common training in the DIII-D PCS (shared by all of these devices) will enable a greater degree of exchange of physics operators for optimal sharing of personnel, as well as cost-effective leveraging of international currency and labor differences.

7.3.10. DIII-D As Emulator, Testbed, and Training Facility for ITER and Other Next-Generation Devices

With DIII-D's flexible PCS architecture, world class actuators (PF, ECH, NB, ICRF, ICC), and extensive set of diagnostics, we propose to "map" the DIII-D system to dynamically emulate properties expected in other tokamaks such as EAST, KSTAR, ITER, and FDF. The large number of PF coils and arbitrary configurability of the PCS in particular make DIII-D unique in its ability to emulate each of these devices in many aspects of actual operations. These characteristics have already enabled DIII-D to be used in physics similarity tests with every other major tokamak in the world. We will extend these studies into machine operations to explore many of the important control issues expected in next generation devices, including shaping limits due to positional and dynamic superconducting constraints, as well as plasma reconstruction constraints imposed by diagnostic limitations and real plasma noise as it is related to plasma position control. This approach will also support the development of a fully validated simulation environment which can be extended to project performance and limits of future tokamak power plant facilities.

A flexible tokamak emulation facility such as this will also make DIII-D the ideal facility for training operators of next-generation devices. ITER operators in particular will require a high-fidelity, high performance ITER emulator to train on which will offer the widest possible range of operating scenarios attainable in present tokamaks. Both the simulation facilities being pioneered in the DIII-D control program and the DIII-D emulator experimental operation capability will fill this need in the critical period just before ITER begins operation.

7.4. PROPOSED UPGRADES

7.4.1. DIII-D PCS Realtime Hardware and Operations Infrastructure

The increased need for control to support the physics mission of DIII-D will require expanded control hardware and software. The existing PCS architecture is flexible, allowing for expansion of data acquisition, actuator command output and real time computing power. Therefore, major changes to the PCS architecture are not planned. Instead, it is expected that the PCS will be steadily expanded to meet the needs of the experimental program by adding additional data acquisition hardware as real time analysis of new diagnostic data becomes necessary, adding computing power as more sophisticated real time analysis and controller techniques are implemented, and adding additional analog and digital output command ports as new actuators become available. Additions to real time computing power are expected to be straightforward as manufacturers increase the power of commonly available, economical computer systems.

The following are some examples of additions to the PCS that are anticipated.

- Identification of an NTM island location from Te fluctuation profiles measured with ECE. This would require acquisition of ECE data at a higher sampling rate than is presently used and computing power for real time analysis of the fluctuation profiles.
- Algorithms to predict and avoid an approaching disruption. In previous work, a beta limit predictor was implemented using a neural network. This would be expanded to include real time

stability analysis. A full bolometer profile and associated analysis would be used to avoid radiation limit disruptions. Other possible approaches will also be investigated. Disruption prediction and avoidance would require additional computing power and data acquisition.

- Real time reconstruction of the plasma pressure profile using measured temperature and density profile data. This would be an input to real time stability analysis. This would require additional real time computing power.
- Acquisition of data from additional measurement locations of the CER diagnostic. This will require data acquisition hardware that is compatible with the CER hardware.
- Expansion of the RWM identification and stabilization capability to toroidal mode numbers greater than 1. This requires additional data acquisition and computing power.
- Expansion of the real time data display capability in order to provide immediate information on discharge performance to operators. This would require new analysis, display, communication and computing hardware and software.
- A dedicated test system, similar to the existing PCS, will be constructed in order to allow for off-line development and testing of new PCS algorithms.

7.4.2. Improved Actuators and Sensors

The DIII-D set of PF coil current actuators comprises a group of five 600 VDC power supplies of various I^2t capabilities and an associated set of 20 ‘choppers’ that act as modulator-regulators between the DC supplies and the PF coils. Each chopper can run in only the positive current – positive voltage quadrant, can handle up to 3 kA and can be run in parallel with other choppers connected to the same DC supply and PF coil. There is another set of two 1200 VDC power supplies and 16 associated choppers. The higher voltage provides faster PF current rise times and more total PF current capability in the larger outer coils. The present set of 600 VDC supplies/choppers is a minimal set that limits the range of shapes and plasma currents that can be achieved in DIII-D. It also prevents committing to a more linear controlled-voltage bus constraint (Section 7.3.4) control system or an even more flexible circuit that eliminates this constraint and allows a much less complicated advanced controller. Figure 7-3 illustrates the stages proposed for gradually evolving away from use of the inboard-coil bus circuit in the 2009–2013 period. Following the final upgrades in 2011–2012, DIII-D will have enough capability to operate in a VFI-less configuration that can access a wider range of operating space than is now available. DIII-D will also have enough reversible current capability to use the DIII-D PF coil set for ‘wind tunnel’ emulation of startup, current rampup and flattop scenarios for ITER and other superconducting (SC) tokamaks.

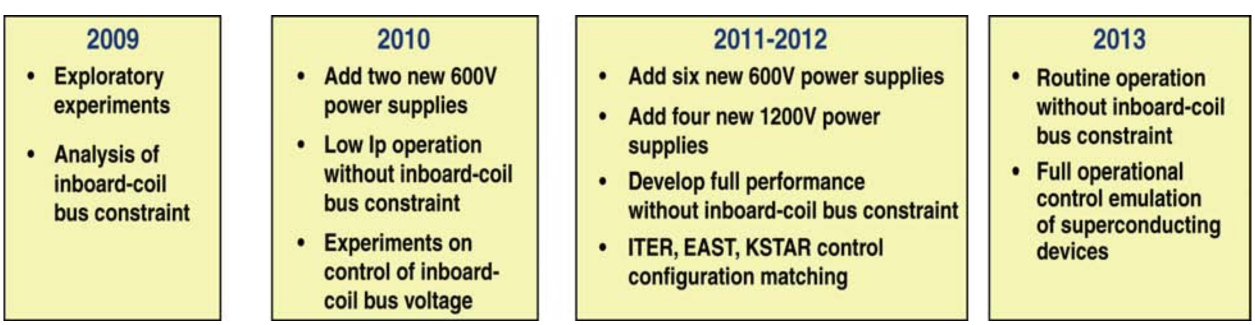

Fig. 7-3. Timeline and stages of development of operation without inboard-coil bus constraint in DIII-D, 2009–2013.

7.4.3. Analysis Software and Computational Upgrades for Analysis

Accordingly, extensive upgrades of software and computational systems will be required to support the ambitious goals of the DIII-D integrated plasma control program. These include:

- Additional Matlab toolboxes and extensions of the Simulink simulation package
- Additional software licenses for toolboxes already installed to support growth in the user base for TokSys in DIII-D operations and collaborations
- Commercial products that facilitate parallel computation for control simulation
- Commercial software for design and documentation of integrated node/network performance (e.g., Telelogic Rhapsody; the Rhapsody/Simulink combination is a proposed standard for ITER CODAC)
- Expanded and higher-performance computational hardware, including Linux workstations

References for Section 7

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8. DIAGNOSTICS — PLASMA MEASUREMENTS

The ability to accurately measure the relevant parameters in fusion plasmas is the key enabler of making progress in predictive understanding and theory/model validation. To adequately test theories, a comprehensive set of diagnostics is required which not only measures all relevant equilibrium parameters [$T_i(\rho)$, $T_e(\rho)$, $n_e(\rho)$, $V(\rho)$, $J(\rho)$, …] with appropriate spatial and temporal resolution, but also measures the turbulence fields. Measurements are needed in the plasma core, the scrape-off layer, the divertor region, and on the first wall material interface. Comprehensive measurements are also required for control of the plasma shape, equilibrium profiles, and MHD stability, and such control enables the optimization of the tokamak concept.

The need for diagnostic measurements and new diagnostic techniques for advances in scientific understanding and plasma control are well recognized, as indicated in the Plasma 2010 report for the NRC: ***"The required progress in […] key areas will not be possible without significant expansion of our plasma diagnostic capabilities. Quite simply, we cannot understand what we cannot measure."*** The Plasma 2010 panel recommend in their report that a new initiative in diagnostic development be formulated at the DOE-OFES level. Although DIII-D presently has the most comprehensive diagnostic set of any magnetic fusion facility, diagnostic improvements planned over the next five years will greatly enhance our understanding of and our capability to control fusion plasmas.

The diagnostic set assembled on DIII-D is the result of many fruitful collaborations with national and international partners. Developing and fielding a diagnostic on DIII-D remains a key involvement for many groups, especially from universities, and offers the capability to participate directly in experiments and scientific discoveries. This large involvement and integration are particularly visible in Table 8-1, the summary of the diagnostics presently found on DIII-D. In addition, the operation, development and maintenance of these diagnostics largely extends across institutional boundaries, through integrated teams.

Table 8-1 also shows (in bold, green letters) the systems which were added or significantly upgraded in the last five years. Separately, shown in Fig. 8-1 is a view of the interior of the tokamak showing some of the diagnostics and the very good access (ports) available. Presently, more than 180 access ports are available, with the majority assigned to diagnostic use.

The previous success encountered in fusion research at DIII-D required the pursuit of three important aspects found especially in diagnostics.

These are:

- QUALITY measurements, accurate, precisely calibrated
- RELIABILITY of the measurement to support experiments in both fusion science and plasma control
- COMPLETENESS of the set (in both coverage, resolution, and parameters)

The scientific goals are also coupled with a renewed challenge in the required coverage (spatial, temporal and spectral). New and upgraded diagnostics are thus required to address the new challenges. When it comes to reliability, there is an even bigger question than the simple availability for physics analysis. We are moving towards to a fully controlled system, where sensors (i.e., diagnostic) are called

Table 8-1
Summary of DIII-D Diagnostics

		Lead Institution
Electron Temperature and Density		
Multipulse Thomson scattering	8 lasers, 44 points	GA
ECE Michelson interferometer	Horizontal midplane profiles	UTexas
ECE radiometer	Horizontal midplane, 40 channels	UTexas
Multichannel vibration compensated (infrared) interferometer	3 vertical chords, 1 radial chord	GA
Microwave reflectometer	Midplane edge profiles	UCLA
Ion Temperature and Velocity		
Charge exchange recombination spect.	24 vertical, 31 tangential, 1 radial chords	GA
Fast ion density profile	8 radial chords	UCI
Core Impurity Concentration		
VUV survey spectrometer (SPRED)	Radial midplane view	GA
Visible Bremsstrahlung array	Radial profile at midplane, 16 channels	GA
Radiated Power		
Bolometer arrays	2 poloidal arrays, 48 channels each	GA
Fast bolometers	3 poloidal arrays, 90 channels	UCSD
Divertor Diagnostics		
Visible spectrometer	12 channels, upper and lower divertor	ORNL
Tangential TV (visible)	2-D image of lower divertor	LLNL
Tangential TV (VUV)	2-D image of lower divertor	LLNL
Infrared cameras	4 camera views	LLNL
Fast neutral pressure gauges	6 locations, 5 in divertors, 1 main chamber	ORNL
Penning gauges	Under divertor baffle (upper and lower)	GA
Baratron gauge	Under divertor baffle	GA
Langmuir probes	26 in lower divertor, 28 in upper divertor and centerpost	SNL
Moveable scanning probe	Scannable through lower divertor outer leg (X-point)	UCSD
Tile current monitors	34 lower divertor, 6 upper	PPPL
Magnetic Properties		
Rogowski loops	3 toroidal locations	GA
Flux/Voltage loops	44 poloidal locations	GA
B_θ probes	88 poloidal locations	GA
Diamagnetic loops	8 toroidal locations	GA
External B_r loops	4 arrays, 32 loops	GA
Internal B_r loops	3 arrays, 18 loops	GA
Internal B_T loops	4 toroidal locations	GA
Plasma Edge/Wall		
Plasma TV	4 cameras, radial view, rf antennae, main chamber	GA, LLNL
Fast framing camera	Tangential view	UCSD
IR camera	Inner wall and ceiling views, floor	LLNL
Visible filterscopes	24 locations	ORNL
Moveable scanning probe	Scannable across outer midplane	UCSD
Fluctuations/Wave Activities		
Beam emisson spectroscopy	2-D, 48 channels (64 in ‘09)	UWisc
Microwave reflectometers	2 radial systems	UCLA
Far infrared scattering	Radial view	UCLA
Phase contrast imaging	Vertical view, 16 channels	MIT
Mirnov coils	Toroidal, poloidal, and radial arrays, 60 coils	GA
Li beam injector	Radial beam with 32 vertical viewing channels	GA
X-ray imaging system	100 channels, 5 arrays	UCSD
RF probes	5 plasma-facing antennae, 6 recessed loops	GA
Scanning probe (midplane)	Temperature, potential	UCSD
Particle Diagnostics		
Fast neutron scintillation counters	2 radial channels	UCI
Lost fast ion detector	2 toroidal locations	UCI
Neutron detectors	4 toroidal locations	UCI
Plasma Current Profiles		
Motional Stark polarimeter	68 channels	LLNL
Li beam injector (edge current profile)	Radial beam with 32 vertical viewing channels	GA
Mod |B|	8 chords	UCSD
Miscellaneous		
DiMES	Lower divertor	GA
Hard x-ray monitors	4 toroidal locations	GA
Torus pressure gauges		GA
Residual gas analyzer		GA

Fig. 8-1. View of the tokamak interior showing some of the internal diagnostics and large port access.

upon to control actuators (power, fields, fueling, etc.) in an increasingly complex way in order to control and sustain performance. Reliability will become a major issue in a successful reactor, where systems have to be available and reliable for thousands of seconds. Finally, the need for completeness is crucial when one contemplates the interconnections between particles and fields, in a multi-dimensional system (space, velocity, and time).

Significant progress in our scientific understanding of fusion plasmas will require the development of new diagnostic techniques. For the DIII-D program, this includes:

- A concerted and collaborative effort between facilities, at the national and international levels
  - Small lab development and testing (e.g., universities, small business)
  - Sharing of experience, engineering design capability and proof of validity
  - Testing and exploitation on larger device such as DIII-D
- A continuous thrust into the introduction of new technologies
  - Small scale, increased sensitivity and ultra-fast detectors
  - Upgrading data acquisition systems for speed and reliability

Our plan in new, improved measurement capabilities is derived directly from the mission and objectives of the experiment. These objectives lead to a set of measurement requirements, which are then turned into techniques (i.e., diagnostics) that can be fielded. We identified a series of objectives in the various topical areas as described in previous sections. These scientific objectives will require a certain number of tools, many being new or upgraded diagnostics. For each objective, physics measurements have been identified and a series of proposed diagnostic techniques elaborated. In some cases well-known techniques can be applied. In others, the development of a new technique will be required. The overall plans for the implementation of these new or upgraded measurements are shown in Fig. 8-2. In each subsection, the items are arranged by priority (top: high priority). Detailed design and available resources will affect the details of the timeline. Triangles indicate known techniques whereas circles indicate that the technique has not yet been selected or determined. These needs are further detailed in the next sections arranged by topical areas.

Fig. 8-2 Planned timeline for the implementation of new or upgraded measurements. Within each subsection, the priority runs from high (top) to lower (bottom). Triangles indicate known or identified techniques, circles indicate presently unknown or unidentified techniques.

8.1. INTEGRATED STEADY-STATE OPERATION RESEARCH

The long term goals for the development of steady-state operation include a variety of sensors/ diagnostics, and their full integration into the control system. While the standard feedback diagnostics, magnetics, density and others have been routine for quite some time, others are being integrated for the optimization of the tokamak approach. These include measurements of the electron temperature (Thomson, ECE), ion temperature and rotation (CER) and current profile (MSE and lithium beam). A clear push for reliable and dedicated measurement will be necessary. Others critical areas include the measurement and control of radiating impurities in the divertors together with a real-time measurement of heat flux. The proposed additional and/or upgraded capability is described in Table 8-2.

Table 8-2
Integrated Steady State Operation Measurement Needs

Scientific Objective	Physics Measurement	Possible Diagnostic Technique
Control current profile evolution	Current profile (core, edge)	Fast MSE, upgraded lithium beam, mod B (full coverage)
Control heat flux to divertor plates	Heat flux	Fast IR cameras, fast thermocouples, calorimetry and improved spatial coverage
	Radiated power	Improved bolometric coverage in divertor

8.2. UNDERSTAND CORE TRANSPORT PHYSICS

In the last few years significant progress has been obtained in the study of the different roles of the turbulent mechanisms (ITG, TEM, ETG, etc.) in heat transport. This progress is also helping to guide the selection of the next generation of key measurements. These key elements are the basis for scientific understanding and model validation which will be required for an optimized utilization and operation of devices such as ITER. In particular, a renewed focus is being developed for the study of particle transport, a natural complement to heat and momentum transport studies already well underway. The proposed additional and/or upgraded capability is described in Table 8-3.

Table 8-3
Core Transport Measurement Needs

Scientific Objective	Physics Measurement	Possible Diagnostic Technique
Understand role of turbulence	High k density turbulence	Upgraded Scattering, Upgraded PCI
	Turbulent flux	Expanded BES, MIR, ECE Imaging
	Magnetic fluctuations	Polarimeter
	Ion temperature and velocity fluctuations	HF-CHERS
	Electron temperature fluctuations	Upgraded CECE
Understand evolution and role of rotation	Main ion temperature and velocity	Main Ion CER
Understand role of fueling	2D neutral population	LIF, laser desorption, imaging

8.3 UNDERSTAND PEDESTAL PHYSICS

One of the largest leverage in expected performance in burning plasma experiments such as ITER is based on the size (height and width) of the edge pedestal. This region of the plasma represents a significant challenge for modeling, theory and measurements, because it is relatively small in scale (approximately centimeters), exhibits large gradients and very fast events (e.g., ELMs), and is not necessarily poloidally and/or toroidally symmetric. It is clear that core confinement of H-mode discharges is strongly influenced by the boundary conditions set by the pedestal values of pressure or temperature; it is necessary to develop a scientific understanding of how fluxes of heat, particles and momentum control the pedestal structure. Three main themes will require new measurements; the study of fueling (neutrals) and the dynamics and evolution of ELMs, including the dynamics associated with their mitigation through resonant magnetic perturbation. The proposed additional and/or upgraded capability is described in Table 8-4.

Table 8-4
Pedestal Measurement Needs

Scientific Objective	Physics Measurement	Possible Diagnostic Technique
Understand role of fueling	2D neutral population	LIF, laser desorption, imaging
	High resolution n_e, T_e	Expanded edge Thomson scattering, fast reflectometer
	Ion dynamics (T_i)	Main Ion CER
Understand role of rotation	Ion dynamics (V_i)	Main Ion CER
Characterize ELM structure	Mode structure	Magnetics, imaging, internal diamagnetic loop
	ELMs dynamics	Fast reflectometer
Characterize ELM mitigation	Heat flux to divertor plate	Fast IR camera, fast thermocouple arrays
Characterize edge stability	High resolution edge current	Upgraded Lithium beam, Fizeau
Characterize edge turbulence	Measure turbulent flux	Expanded BES, additional probe coverage, GPI, fast reflectometer

8.4. UNDERSTAND BOUNDARY PHYSICS

Arguably, the next frontier in magnetically confined plasmas resides in the development of a scientific and technological solution to the challenges encountered in the boundary of a magnetically confined plasma, and especially in the case of burning plasma devices. The boundary layer of a magnetically confined plasma encompasses vastly different conditions over a small physical scale. Temperatures of the order of kilo-electron-volts and high densities are found very near inside the last closed flux surface (e.g., pedestal), whereas much lower temperatures are encountered at the plasma-wall interface. This wide contrast depicts the challenge encountered in diagnosing this region of the plasma. Several underlying physical issues complicate our attempt in understanding boundary physics. Transport, MHD stability and atomic physics all play a role in controlling the conditions encountered in that region. The boundary physics has a strong two-dimensional character due to various poloidal asymmetries and the presence of an X-point, and strong variations are encountered in the radial direction from the inner edge of the pedestal out to the limiting surfaces. In addition, stocastic/ergodic edges produced by nonaxisymmetric fields introduce explicit 3D geometry and added complexity. The interaction of the hot plasma with the first wall material and the impact of any eroded material on the plasma core are important and relevant issues for understanding boundary physics. The proposed additional and/or upgraded capability is described in Table 8-5.

Table 8-5
Boundary Physics Measurement Needs

Scientific Objective	Physics Measurement	Possible Diagnostic Technique
Understand particle (fuel and impurity) transport and generation in SOL	Flow velocities	Spectroscopy, probes (fixed and inner-wall scanning)
	Particle deposition and composition	Quartz micro-balance, colorimetry, in-situ ellipsometry, laser induced breakdown spectroscopy, fast IR camera
	Hydrocarbon generation	Cavity ringdown spectroscopy
Understand and control heat flux to divertor plates	Heat flux	Fast thermocouples, IR camera, calorimetry
	Ion heat transport	Retarding field analyzer, lithium beam ion CER
Characterize dust dynamics	Dust velocity and density	Thomson scattering, fast imaging
Characterize edge turbulence	Density, electric field fluctuations	Gas puff imaging, 2D array of fixed probes

8.5. UNDERSTAND PLASMA STABILITY PHYSICS

The establishment of the scientific basis for understanding and predicting limits to macroscopic stability of magnetically confined plasmas has many control implications. While a large part of the research is aimed at investigating and validating basic MHD stability physics, making use of DIII–D's extensive set of diagnostics for precise, detailed measurements of the pressure and current density profiles, along with details of the internal structure of MHD modes can increase substantially the operating regimes of the tokamak approach. In fact, stability research includes critical issues for both conventional and Advanced Tokamak (AT) plasmas. With recent advances in scientific understanding and technical tools, we are beginning to study plasmas compatible with steady-state operation, and to develop active means of controlling stability, which will require appropriate sensors. In addition, the detection and mitigation of disruptions (Section 8.6) will require specific diagnostics, which includes the development of sensitive, dedicated and reliable sensor for the feedback control. The proposed additional and/or upgraded capability is described in Table 8-6.

Table 8-6
Plasma Stability Measurement Needs

Scientific Objective	Physics Measurement	Possible Diagnostic Technique
Characterize NTM, RWM and TAE radial mode structure	2D electron temperature	ECE imaging
Characterize error field, NTM and RWM poloidal and toroidal mode structure	Magnetic equilibria and perturbations (first wall)	Additional magnetics coverage (3D)

8.6. UNDERSTAND DISRUPTION PHYSICS

It is well understood that for high performance burning devices, such as ITER and eventually DEMO, disruptions need to be avoided, and in the last resort mitigated. Because of the unique characteristics of disruptions and the associated mitigating technique, their study requires dedicated diagnostics which encompasses very fast time scales, localized events and emission and difficult environmental conditions. A focus of the DIII-D research includes the understanding of the electron run-away generation and its mitigation through massive gas injection and pellet injection. The proposed additional and/or upgraded capability is described in Table 8-7.

Table 8-7
Disruption Measurement Needs

Scientific Objective	Physics Measurement	Possible Diagnostic Technique
Characterize runaway formation	Width current channel, energy of runaways	Gamma spectroscopy, IR tangential camera, EUV camera
Characterize magnetic structure	Halo currents	Tile current monitors, halo sensors
Understand mitigation relationship with plasma and first wall	Neutral and low state emission, first wall temperature	Fast imaging (visible and IR), spectroscopy, EUV camera

8.7. UNDERSTAND ENERGETIC PARTICLES PHYSICS

It has long been recognized that energetic particles bring new challenges (and opportunities) in reaching proper conditions for burning plasmas. The confinement of these particles is particularly important and their impact on plasma instabilities, such as Alfvén instabilities is very critical. At DIII-D, a renewed effort has been applied to the study of energetic particles in the last few years. That development has been possible with the capability enabled by new and upgraded diagnostics such as the fast interferometer, scattering, electron cyclotron emission (ECE), beam emission spectroscopy (BES) and fast-ion D alpha (FIDA). These new developments have focused on the internal structure of the modes. A major focus of the proposed plan is study the effects of these modes on the energetic particles themselves. The proposed additional and/or upgraded capability is described in Table 8-8.

Table 8-8
Energetic Particles Measurement Needs

Scientific Objective	Physics Measurement	Proposed Diagnostic Technique
Understand mode structure	Measure mode polarization	2-axis magnetic probes (wall)
	Measure toroidal mode number	Toroidally displaced interferometer chords, toroidally displaced ECE measurement
	Measure radial mode structure	Expanded BES array
Understand interaction of mode with fast ions	Measure confined fast ion population (profile)	Upgraded fast ion D alpha (FIDA) to 8 channels, selected neutral particle detectors (diamond and/or silicon)
	Measure loss of fast particles	Array of fast detectors on outer wall

8.8. DEVELOPMENT OF DIAGNOSTICS FOR BURNING PLASMA EXPERIMENTS

Diagnostic development for a Burning Plasma Experiment is also sorely needed. In a burning plasma experiment, the application of standard techniques and the fielding of specialized diagnostics are facing challenges including environmental issues (e.g., radiation), access, long pulse, etc. Additional requirements (e.g., alpha particles) are particularly difficult to meet. The development of these new or alternate techniques presently lack testing capability in an existing tokamak. In DIII-D we plan to address many of these issues, including coordinating activities with U.S.-procured diagnostic packages for ITER.

Specifically, these activities aim to address the following questions/issues:

- The development of a new technique where needed
- The test of a new technique in a large tokamak with relevant parameters
- Development of alternative technique for problematic measurements
- Complete test of control techniques (reliability, versatility)
- Continue study of erosion and deposition (for eventual input to diagnostic design)
- Continue evaluation of measurement requirements for a BPX, in regard to profile, divertor and/or control associated needs

The development of burning plasma diagnostics and related technology requires a coordinated effort with the U.S. Burning Plasma Organization (USBPO), U.S. ITER project Office (USIPO), the International Tokamak Physics Activities (ITPA) and ITER international organization.

We anticipate that in the time frame covered by this proposal, a series of prototypes will be fielded on DIII-D to test proposed designs for the U.S.-procured systems to ITER. That list includes both interferometers (tangential and divertor), ECE, MSE, viewing systems (IR and visible), reflectometer and possibly X-Ray crystal spectrometers.

In addition, we propose to develop alternate techniques that may be required for ITER and/or other burning plasma experiments, including but not limited to FDF and DEMO.

They include such techniques as:

- Demonstration of Fast-Alfvén reflectometry for isotope mix ratio measurement
- CER based measurement for q profile reconstruction
- CER based measurement of fast ion population
- New soft-x-ray concepts
- New concepts in polarimetry and interferometry
- Fizeau effect interferometer for electron velocity diagnostic

8.9. SPECIFIC DIAGNOSTIC REFURBISHMENTS

While periodic maintenance diagnostic systems ensures their reliability, it has become necessary to refurbish some key systems. In those cases, maintenance is prohibitive or impossible, due to the availability of parts (detectors, electronics, etc.). In the last five year period, the interferometer, magnetics integrators, and poloidal SXR have been replaced by a new design and components. The electronics and data acquisition for the Thomson scattering system are being redesigned and will be complete during the first part of the next five year period. The MDS spectrometer and filterscopes systems are also expected to be complete by the beginning of the next period. Other planned refurbishments include the toroidal SXR system (3 cameras), gradual replacement of the CER cameras (~10) and full replacement of the IR cameras. In each case mentioned, the refurbishment has been accompanied by significant upgrades of their capability, in large part due to the advancement in technology over the last 20 years. The refurbishment of the data acquisition system is also planned and details can be found in Section 10.12.

9. DATA ANALYSIS AND REMOTE PARTICIPATION

Efficient data analysis is fundamental to the successful accomplishment of the DIII-D science mission. Data analysis prior to tokamak operations is critical to developing an effective experimental plan, between-pulse analysis allows for effective decision making during an experiment, and post-experimental analysis is required for full understanding and subsequent publication of results. The geographically diverse nature of the DIII-D National Team necessitates software technology that offers the full suite of data analysis capabilities independent of a team member's location. Furthermore, since human interaction is often a critical component of the scientific method of discovery, the remotely distributed DIII-D scientific team requires communication tools that enriches rather than hinders these interactions.

Significant progress was made in the previous five years to support the needs of the DIII-D National team. Previously, MDSplus was deployed to unify the storage of all analyzed data and to provide an event/dispatching system for usage during tokamak operations. The successful deployment of MDSplus is being enhanced through the adoption of distributed MDSplus that yields a scalable software and hardware architecture to handle an arbitrarily large amount of data. This scalability is critical, as the amount of data being stored in MDSplus per shot has more than doubled in the previous five years. This increase is a combination of greater data analysis and several new diagnostics that use MDSplus for data acquisition. Significant advances were made in the area of between-shot data processing including increasing the available computational hardware as well adding new computational codes such as ONETWO that performs an automatic power balance analysis. New visualization tools were created and deployed. Work during the previous five years has been enriched by the close coupling with the SciDAC National Fusion Collaboratory Project [Schissel 2005] that created the software necessary for a secure distributed computing environment (FusionGrid) and novel communication and application sharing techniques that can be used in a distributed environment. The software partnerships with C-Mod/MIT and NSTX remain strong and will continue to enhance the DIII-D program.

This section outlines work in the areas of computer systems, programming, remote collaboration, and user support for data analysis that are required to effectively accomplish the mission of the experimental facility and the distributed researchers over the next five years. The specific goal is to allow rapid, secure, access to all analyzed data, data analysis codes, and visualization applications on a 24/7 basis, and the ability to effectively communicate with the team that is conducting tokamak operations. To lower the barriers for secure access to DIII-D resources, single sign-on certificate based authentication will be broadly deployed combined with unique authorization systems. Data storage technology using MDSplus will be expanded to allow faster data retrieval rates to a larger user base in a secured way. Enhancements to between shot processing will be deployed and this large computational resource will be made available to the general scientific staff during periods of no operation. Web clients will be more broadly used to extend the reach of our applications. Novel techniques will be created and deployed to allow the DIII-D Team to analyze data at a greater rate and to assimilate that knowledge between pulses to effect the creation of the next pulse. Advances in remote collaboration technology will be deployed in an effort to provision a working environment for off-site personnel engaged in experimental operations that is every bit as productive as what is on-site. Table 9-1 summarizes some of the recent progress in data analysis and remote participation and plans for future work in this area.

Table 9-1
Progress and Plans for Data Analysis

Recent Progress	Challenges/Plans
Analysis Software Improvements	
▶ New data analysis applications * SAV for signal analysis * ProfileView for profile comparison ▶ Session Leader startup transitioned to web client interface * Session Leader Checklist web site	▶ Expand reach of tools to DIII-D team * Transition to web-based clients for appropriate tools like Electronic Logbook and visualization tools ▶ Improve quality and quantity of analyzed data * More detailed overnight analysis with CERAUTO, Kinetic EFIT, ONETWO, and Stability analysis * Investigate new profile fitting algorithms ▶ Deploy scientists' specific codes to a broader user base * User interface and software engineering * Python toolkit as an example ▶ Increase usability of existing codes * EFIT, TRIP3D, TGLF, ELITE
Between Shot Data Analysis	
▶ 48-node STAR cluster operational * Equilibrium, Kinetic Profiles, time independent ONETWO analysis	▶ More analyzed data needed in control room * Advanced queuing system * Perform kinetic EFIT * Perform stability analysis ▶ More complicated analysis infrastructure * Advanced monitoring system * More efficient analysis to fit everything into the between-shot time window
Data Storage	
▶ MDSplus deployed on Intel-based Linux for raw and analyzed data * 2 TB of total storage	▶ Scalable storage solution at modest incremental cost with good performance * Deploy distributed MDSplus with new hardware * Incorporate camera data ▶ Plant data available for physics analysis * Deploy long-pulse MDSplus to store critical plant data on 24/7 basis ▶ Increasing data repository size increases complexity in search data * Increase meta data in relational database
Analysis Infrastructure Improvements	
▶ Expanded LSF cluster ▶ Deployed Wiki Web Site ▶ GoogleMini search ▶ Data Analysis Monitoring	▶ More efficient utilization of STAR cluster * Deploy new queuing and job management system * Enhance computational power as required ▶ Enhanced computational and data analysis monitoring ▶ Enhance Web security * Username/password and/or certificates
Control Room and Remote Participation	
▶ Display wall in control room * Realtime EFIT and customized data monitoring ▶ Control room audio/video functionality * Access Grid, VRVS, high-resolution web cam ▶ Enhanced remote capability * 8:05 and Friday science meeting * Specially equipped room	▶ Make more data available in control room * Integration of camera data and RT EFIT on display wall ▶ Increase control room communications robustness * Integrate H.323 and EVO into control room * Integrate communication (e.g., IM) into analysis tools ▶ Many distinct tools to accomplish an experiment * Unify tools where appropriate to create simpler environment ▶ Meeting attendance for scientists spread over many time zones * Record meetings and create simple playback

9.1. ANALYSIS SOFTWARE IMPROVEMENTS

Improvements to the DIII-D analysis infrastructure are aimed at making the DIII-D team more scientifically productive (Fig. 9-1). This will be accomplished by increasing the scientific value of each DIII-D shot and by the ability to analyze more data. Specifically, work in the next five years will focus on:

- Consolidating existing tools to streamline analysis
- Improving user interfaces to reduce complexity
- Creating and deploying new data analysis algorithms
- Enhancing or creating documentation
- Deploying scientists' specific codes that have a broad appeal for general usage
- Advance how CPU intensive codes are distributed for execution
- Easing communication hurdles for both local and remote scientists

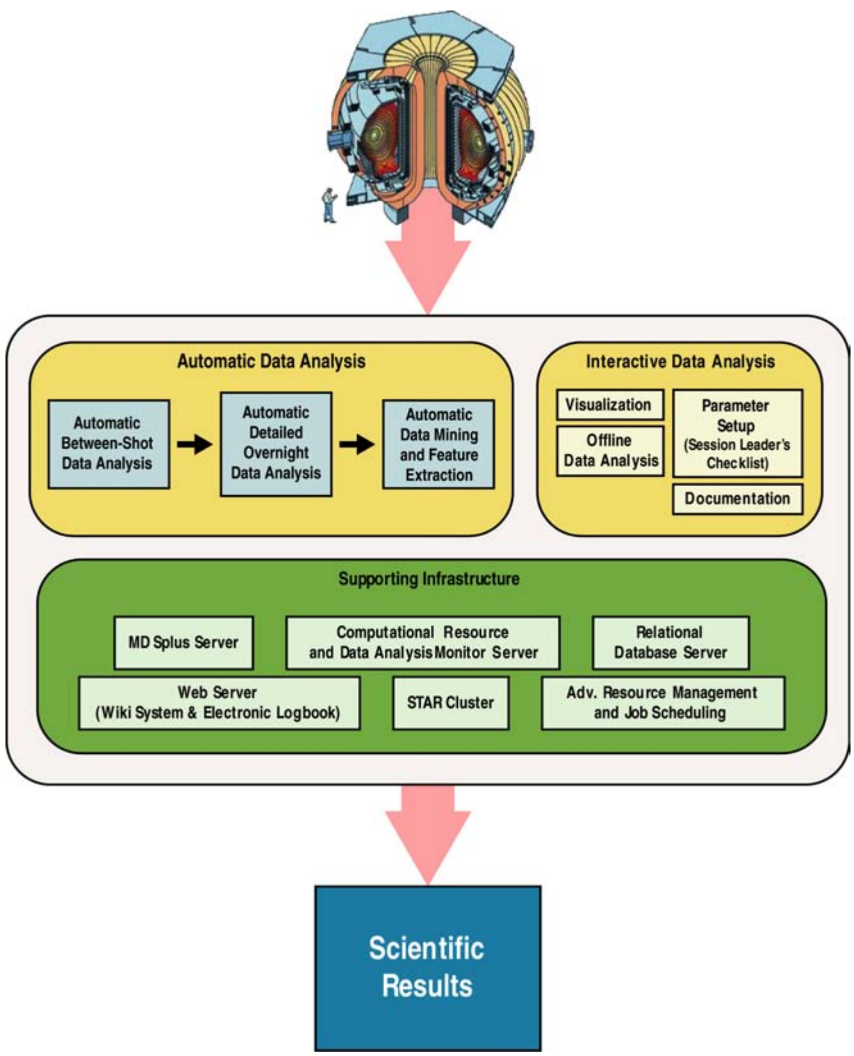

Fig. 9-1. DIII-D's scientific data analysis will be enhanced in the next five years through increasing the amount of automatic and interactive data analysis and by expanding the supporting infrastructure as required.

There already exists within DIII-D a large set of data analysis tools. Many of these tools are distributed across the team and not designed to smoothly interact with each other. This lack of interoperability raises the barrier of adoption and can increase the time for a new team member to become productive. A major thrust for the next five years will be the consolidation of existing tools into a general umbrella user interface to streamline analysis. The successful consolidation will require both new graphical user interfaces and inter-code communication. Details of this activity are discussed for integrated models in Section 6.4. Yet, the goal is designed to target both modeling and analysis tools. It is anticipated that tools associated with profile fitting, plasma equilibrium, stability, and transport will all be examined. Specific codes include GAProfiles, EFIT, ONETWO, TRIP3D, TGLF, GATO, and ELITE.

The increasing amount of both raw and analyzed data being generated by the DIII-D facility requires new and improved ways to efficiently mine these repositories to extract knowledge. Work at DIII-D and other tokamaks have illustrated the potential value of automated knowledge discovery software for producing models in a form that is common in tokamak plasma physics research. This new capability has the potential to broaden data retrieval beyond shot based to feature based extraction. Additionally, anomaly detection can be possible where a scientist looks for a feature that is unique or unexpected. New data mining capabilities will be created to formulate, analyze, and implement basic induction processes that facilitate the extraction of meaningful information and knowledge from unstructured data. These capabilities will allow the semi-automatic discovery of knowledge in the form of patterns, changes, associations, anomalies, rules, and statistically significant structures and events from the large DIII-D data repositories.

Improvements in diagnostics at DIII-D have resulted in complete time histories of kinetic profiles such as ne, Te, Ti, Vr, Prad, and Zeff. These profiles are fit automatically between-shots (ZIPFIT) and also in an interactive fashion for a subset of shots. The interactive fitting is done mathematically using spline techniques at each time point. Throughout a pulse profiles are measured at several hundred times and therefore it is tedious and time consuming to create a temporal evolution of kinetic profiles for input into analysis codes (e.g. power balance). New mathematically fitting capabilities will be investigated to allow the researcher to simultaneously fit profiles in time and space. This capability has the potential to dramatically increase the amount of profile data that can be analyzed and therefore increase the quantity of more refined analysis (e.g. power balance, simulations, stability) that can be generated.

The implementation of improved analysis of camera diagnostics yielding heat flux per pixel will be possible due to the usage of distributed MDSplus and the availability of the STAR cluster for overnight processing. Additionally, new analysis codes such as that being utilized on TEXTOR will be investigated for routine deployment at DIII-D. The desire is to have a broader range of analyzed camera data available to the entire DIII-D staff.

The Electronic Logbook [Fredian 2006] has proved to be a very valuable as a tool for tokamak operations. It allows real-time distribution of scientific comments, something that is important for a geographically distributed research team, as well as a rapidly searchable historical archive for documentation purposes. The IDL-based tool Entry_Display is used to enter and read comments; presently the web browser interface only allows comments to be read. In an effort to increase the reach of the log book the web interface will be modified to allow entering of comments. Additionally, a mechanism will be put into place that will allow rudimentary graphics to be pasted into a comment. Therefore a comment such as “unknown source of impurity at 1400 msec” can be accompanied by the graph that led to the comment. Adding the visual capability to the existing text functionality of the logbook should significantly increase is usefulness in transmitting and archiving scientific information.

The increase in complexity of data analysis requires a concurrent increase in both data and code documentation. The transition to a Wiki-based DIII-D web site has increased the amount of scientists who can keep the present web-based documentation current. However, a richer documentation set will be created concurrently with the increase in analysis tools and a streamlined analysis path already discussed. New documentation will include text as well as test cases and benchmarking. The goal of this effort is to reduce the time it takes a new DIII-D team member to become productive.

Given the ubiquity of web browser clients on all operating systems, more client software will be transitioned away from custom applications to web-based systems. An early example of this is the transition from Java-based LeadList program to a web interface. Client interfaces that control analysis programs that are traditionally written in IDL, will be transitioned to a web interface where appropriate. This work will be performed in concert with the task above to create a more streamlined analysis path. This transition will eliminate the need for a commercial IDL license that are sometimes not available to DIII-D’s off-site scientific staff thereby allowing more scientists to easily access DIII-D analysis codes.

During the previous five years, FusionGrid was deployed to give secure worldwide access to fusion energy sciences resources (data, codes, visualization tools) [Schissel 2004]. In an effort to expand the computational capability of the DIII-D organization, the ONETWO and GATO codes will be deployed with dedicated hardware as FusionGrid computational services. This task will utilize the queuing system being deployed on the STAR cluster to manage multiple job requests.

The significant enhancement to analyzed data needs to be accompanied by an increase in visualization capability. General 2D visualization is handled by the Review and ReviewPlus applications. However, the significant increase in calculated profiles really requires a dedicated code. To meet this need the existing Profile_Viewer application will be extended and a new version of PLOT12 that is used for power balance data visualization will be created.

9.2. BETWEEN SHOT ANALYSIS

Between-shot data processing enhancement is a critical element towards increasing the utilization efficiency of the DIII-D facility. Towards that goal, the STAR between-shot data analysis Linux cluster has grown from the original 12 computers to today’s equivalent of 48 computers representing a 20-fold increase in computational power. The original software queuing system has become inadequate to handle the increase in processor quantity and the total number of codes that can be completed. The open source CONDOR queuing system was investigated but has proved lacking in certain key areas. Thus, a major part of this work plan will be the evaluation and deployment of a new queuing system that will allow for substantial new growth in the STAR clusters computational capability. The goal is to deploy an architecture that allows *m* codes to be operated on *n* computer cores.

During the previous five years, a number of new computational codes have been added to the STAR cluster including profile analysis and the ONETWO transport code (Fig. 9-2). During the next five years, new codes will be added including kinetic EFITs and TGLF. The architecture discussed previously will allow these new codes to be deployed. Additionally, there are a number of critical between-shot computer codes that run on the general-purpose cluster of Linux machines (LSF cluster) that utilize Load Sharing

Facility from Platform Computing for load balancing of code runs. Traditionally, these codes were on the LSF cluster because the original STAR cluster did not have sufficient computational power. Today that is no longer the case. Operating these codes on various LSF cluster nodes introduces another point of failure for operations related computations with no measurable gain. Therefore, these codes will also be moved to the STAR cluster.

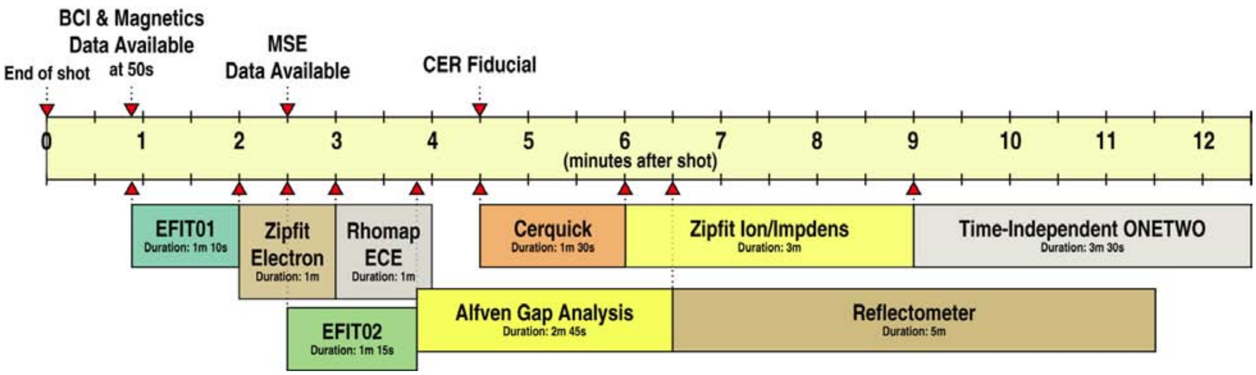

Fig. 9-2. The amount of data analyzed between shots has grown dramatically with the deployment of the 48-node STAR computational cluster. This cluster will be used for additional between-shot processing as well as more detailed overnight processing over the next five years.

With an increasing amount of computational data analysis occurring between shots, and with the projected growth of available data, more centralized monitoring capabilities are required to maintain a manageable workload by the physics and operation teams. This means that as new analysis codes are added to the between shot analysis software, and as the current codes are improved, it is important to concurrently develop a monitoring system to allow researchers and developers to quickly monitor the processes that are run between shots. The Data Analysis Monitoring (DAM) System [Flanagan 2004] is already being used to track the data analysis codes launched for between shot analyses at DIII-D. With development in web browser displays, real-time software, and expert systems, operators can quickly and intelligently, be notified to potential errors that may be remedied during the run. By adding these new capabilities to DAM, an advanced monitoring system will be developed for the U.S. fusion community and DIII-D specifically. This will allow physics teams to increase the amount of valid data produced from each run, as well as allowing them to manage the inevitable increase in analysis processes taking place between shots.

9.3. DATA STORAGE

The MDSplus data system was adopted by DIII-D in September 1997 to organize, under one common client/server interface, the storage of analyzed data. Since that adoption, the flexibility and versatility of MDSplus has resulted in a number of DIII-D diagnostics utilizing the system also for data acquisition. Complimenting MDSplus data storage is a relational database that stores highlights of this vast data repository. Users can rapidly search through the data highlights stored in the relational database and find the subset of pulses that have special interest. The secure and efficient worldwide distribution of analyzed data is critical to the success of the DIII-D mission and both of these data systems will continue to be the vehicle for distribution.

During the previous five years, the Atlas MDSplus server was transitioned from a single processor DEC workstation (Alpha chip running OSF) with 500 MB of storage to a dual processor Intel-based Linux system with 2 TB of storage. This enhancement was necessary for both processing speed during experimental operations as well as total storage capacity. To further enhance the MDSplus system deployed at DIII-D, the transition to a fully distributed MDSplus installation will be completed as part of this effort. One benefit of distributed MDSplus versus traditional MDSplus is that it allows faster data retrieval by a large number of scientific users (Fig. 9-3). Additionally, it allows an arbitrary number of distinct computer systems to appear to the scientific team as one unified MDSplus server. This architectural design allows an array of servers to be used for MDSplus rather than one-large system. The advantage from a computer infrastructure standpoint is that additions to the system are easy and rapid and these incremental improvements are low cost compared to replacing the entire system. This design should accommodate the data needs of DIII-D into the foreseeable future. Presently DIII-D adds to the MDSplus repository approximately 3.9 GB per run day. Assuming no increase in this number over five years yields a minimum estimate of an additional ~ 2 TB for 21 operation weeks per year. It is anticipated that the daily amount of data will increase (see the camera discussion below) but even if it were to double (it has increased 30% over the last three years), an additional 4 TB of disk space is easy to handle in today's world. Increases in capacity, both computer cycles and storage, will be deployed as required.

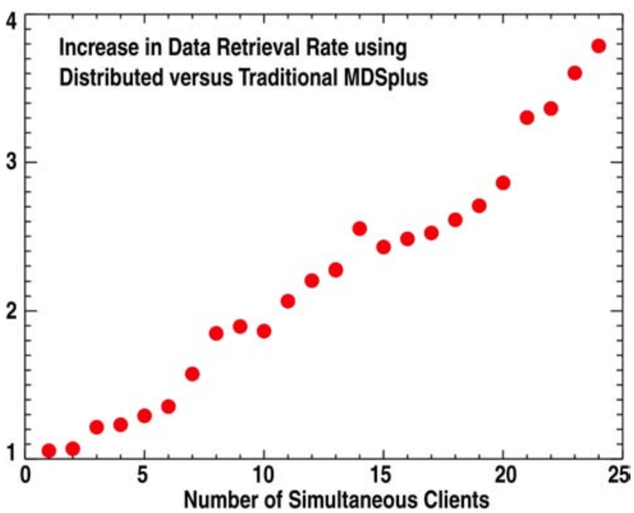

Fig. 9-3. Distributed MDSplus allows for faster data retrieval rates by a large number of scientific users.

New datasets are continually being added to the MDSplus data repository and the associated relational database. Such activity will of course continue and assistance will be provided as required to the scientific staff. An already identified area of work is the storage of the data from numerous camera diagnostics allowing easier distribution to the entire scientific staff including projection on the control room display wall. Part of this work will include the CHIRON MDSplus server for camera data becoming part of DIII-D's overall distributed MDSplus data repository. Additional discussion regarding support for camera diagnostics is given in a later subsection.

The recent long-pulse extension to the MDSplus server software allows writing and reading data in segments before the entire pulse is completed. Although DIII-D tokamak data does not require this feature, certain plant data such as vacuum pressure or wall temperature, that are acquired on a 24 hour cycle, can now be stored in MDSplus. The intention is to add to the MDSplus data repository those plant specific quantities that might have an impact on analysis of the plasma physics although any plant data can be added if its universal access is deemed beneficial.

9.4. ANALYSIS INFRASTRUCTURE

The Load Sharing Facility (LSF) computational cluster (interactive computation and visualization) was deployed originally to present a unified computational environment from a collection of individual HP and DEC workstations. Over time, the HP and DEC computers have been retired and replaced by several multi-core Linux workstations. However, the amount of available computational power in the LSF cluster has not grown to keep up with the interactive demands of the DIII-D National Fusion Facility. Therefore additional nodes will be deployed into the LSF cluster over the next five years. There is a potential increase licensing cost associated with the LSF software from Platform Computing on multi-core computers. This issue will be investigated and if necessary new load sharing software will be deployed.

Traditionally, even though the LSF cluster was deployed for interactive usage it has been utilized for some batch processing as well. At times, these batch jobs have overwhelmed the cluster. Examination of the between-shot STAR cluster utilization integrated over an entire year finds that its percent utilization is low since it receives heavy usage only during the daytime when the tokamak operates. To provide more computational power to the DIII-D scientific staff the between-shot STAR cluster will be configured to allow batch jobs during non-tokamak operations. This task will dovetail with the modernization of the queuing system on the STAR cluster. It is envisioned that a STAR cluster queuing system will allow scientists to submit batch jobs for processing and will automatically handle issues associated with DIII-D operations. Opening up the STAR cluster to the scientific staff will represent an approximate two orders of magnitude increase in available computational power.

Prior to the introduction of the STAR cluster there was a significant amount of shot-specific analysis that was performed overnight. As the STAR cluster was rolled into production, those overnights codes were run between-shots and with no new codes being run overnight. Recently there have been requests to refine some between-shot analysis and the STAR cluster will be utilized for this purpose as well. These refinements will be first applied to CER, ONETWO, and certain camera diagnostic analysis and will be done automatically.

The transition to utilizing more of the web for DIII-D services requires the need to be ever more vigilant on security. Traditionally, DIII-D’s web site has been restricted only through a check of the accessing domain. However, this methodology is no longer proving sufficient and a transition to a username and password system will be undertaken. It is envisioned that this transition will utilize the user X.509 certificates issued through FusionGrid [Burruss 2006]. By utilizing user certificates a one-time login system can be created for the entire DIII-D web site that is actually comprised of several physical machines. Additionally, the user’s certificate can be used to validate access to computational resources that are made available via a web interface. The usage of certificates on an individual basis also allows security based on authorization to be done at a much finer granularity. Certificates can be issued to any DIII-D Team member and are coordinated through DIII-D and ESnet.

9.5. CONTROL ROOM AND REMOTE PARTICIPATION

During the previous five years, a display wall was installed in the DIII-D control room along with a variety of new software (Fig. 9-4) [Abla 2007]. One of the software improvements allows for a real-time visualization of the real-time EFIT calculation. The improvements to analysis and display of camera diagnostic data will include the ability to visualize camera data on the display wall side-by-side to the

EFIT display. Additionally, a variety of visualization techniques will be investigated that allow for the fusing of EFIT and camera data to allow a richer understanding of the observed phenomena. This new visualization will be available both on the display wall and on individual displays.

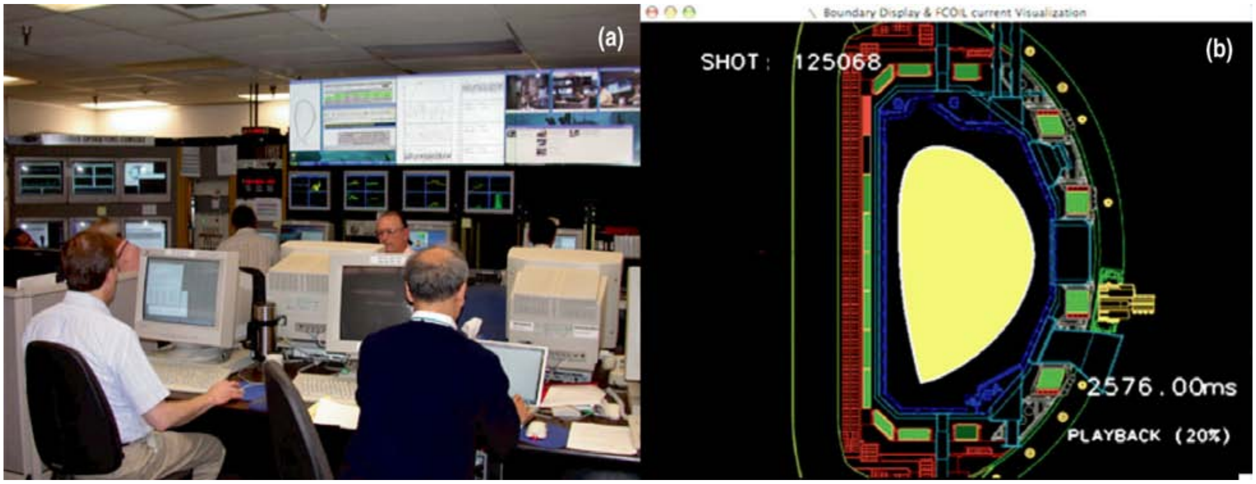

Fig. 9-4. (a) The DIII-D control room has a 3-tile display wall for enhanced communication within the DIII-D control room as well as communication from remote participants. (b) A recent addition to information on the display is the real-time EFIT calculation showing not only the plasma boundary but also coil current information.

One aspect of remote participation is to join, from off-site, seminars, meetings, and data analysis sessions that are held on-site and vice versa. Often, time zone issues or just schedules preclude the ability to participate remotely. Even on-site scientists may not be able to attend a meeting due a higher priority obligation. To help alleviate this difficulty, the DIII-D pre-operations meeting will be recorded and available via the DIII-D web site. These recordings will also be available via a podcast that allows the distribution of the meeting over the Internet using syndication feeds for playback on portable media players and personal computers. After the technology is hardened, recording will be transitioned to other seminars and meetings as required. It is recognized that some meetings are not suitable for recording and re-broadcast such as those where highly speculative results are being presented for the first time. Yet, there is clearly a class of meetings where recording and re-broadcasting is not only appropriate but beneficial. This capability will be developed in close collaboration with DIII-D management.

Remote access to the DIII-D control room has been provided through either an Access Grid or VRVS interface (Fig. 9-5). This capability has allowed both remote individuals to participate in the day's experiment as well as to act as the experimental session leader. Improvements to this communication capability will be investigated and deployed as required. The DIII-D pre-operations meeting has been changed to be broadcast with H.323 videoconferencing and this will be investigated for the control room as well.

Additionally, new capability that is created and deployed as part of the EAST collaboration, discussed elsewhere in this plan, will be transitioned into the DIII-D control room as appropriate. Of particular interest will be developments that unify the number of different systems. Presently, a DIII-D scientist is presented with a large array of possible tools. The desire is to have one unified toolkit that can be used for a multitude of situations yet presents one simplified interface. Modules that support ad hoc and structured interpersonal communications, persistent collaboration environments along with shared displays and applications have the potential to significantly impact the efficiency of remote scientific participation.

Fig. 9-5. Remote participation on DIII-D experiments utilizes a variety of technologies. In this figure VRVS is being used to bring remote JET scientists into the DIII-D control room.

9.6. USER EDUCATION AND TRAINING

The rapid rate of technological advance must be met equally with an education program that keeps the DIII-D National Team up to date with relevant hardware and software technology. The work described in the preceding sections is all aimed at creating a more productive data analysis environment. Yet, this work will require some team members to modify how they presently conduct their data analysis. Information dissemination at the Friday Science meeting will continue. Specific technology classes will be taught on as needed basis to keep the researchers informed of these new technologies. These will be similar to classes taught previously on IDL, object oriented programming, MDSplus, and SQL. New classes will be recorded and available for download via a web browser to provide the beginnings of a reference library.

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10. THE DIII-D NATIONAL FUSION FACILITY – OPERATION AND ENHANCEMENTS

The DIII-D National Fusion Facility is a world-class facility capable of carrying out a wide range of experiments to explore high performance tokamak discharges as well as fundamental fusion science. In this section, we describe improvements to the device hardware and infrastructure that will enable steady research advances while maintaining high system availability. Table 10-1 summarizes the proposed improvements and the research needs that are driving the changes. Major upgrades will include significant increases in the heating and current drive power and pulse length, new coil systems for ELM control and divertor heat flux expansion, tokamak pulse extension to 10 s, an upgraded vessel armor system compatible with high power and long pulse operation, and upgrades to the RWM stabilization system and disruption avoidance and mitigation systems. The proposed schedule permits significant operation each year while providing new capabilities to the research program throughout the five-year plan.

Table 10-1
Major Hardware Upgrades

New Capability	Hardware Upgrades	Research Elements	Section
Reduced error field	30 deg TF feed modification	Low rotation physics	10.3
EC: 12 MW, 10 s	8–1.5 MW gyrotrons; ECH PS#4; 2 transmission lines, electronics, launcher	J(ρ), NTM, $T_e \sim T_i$	10.4
NB: 10 MW, off-axis	Tilt 2 beamlines	J(ρ), energetic particles, Toroidal/Poloidal rotation;	10.5
20 MW, 10 s	Upgrade beamline internals	Long pulse AT	
FW: 6 MW, 10 s	Advanced high power antenna	J(ρ~0), Te~Ti, energetic particles	10.6
Flexible resonant magnetic perturbation	48-element inner wall coil	ELM control, heat and particle control	10.7
Divertor flux expansion	2-4 element axisymmetric coils in lower divertor	Heat and particle control	10.7
RWM amplifiers/network	24 additional amplifiers; cross-over network	Dynamic error control with n=1,2 RWM stability	10.7
Deep fuel/impurity penetration	Inverse jet; custom pellets; large pellets; liquid jet	Disruption mitigation	10.8
300 MJ heat removal (60 MJ present)	Divertor and vessel armor upgrade (CFC tiles; upgrade water cooling)	10 s high performance, physics of heat removal	10.9
Hot Wall Operation	Modified Air-water cooling system	Hydrogenic co-deposition and removal	10.9
Longer Pulse Operation (10 s)	Additional DC supply and current regulators; thermal interlock/monitor upgrades	10 s high performance	10.10
Prime power for long pulse	New 138 kV and 12.47 kV transformers	Enables long pulse AT	10.11

Section 10.1 summarizes the present system capabilities and Section 10.2 contains a short summary of each of the upgrades and the proposed schedule for research operations and system upgrades. More detailed descriptions of enhancements and refurbishments to existing systems and upgrades and new systems are presented for each subsystem in the following sections.

10.1. OVERVIEW OF CURRENT CAPABILITIES

At the heart of the facility is the DIII-D tokamak, which is capable of operating at plasma currents up to 2.5 MA with a magnetic field of 2.2 T. The DIII-D tokamak is renowned for its operational flexibility, which enables a wide range of research in highly shaped limiter and divertor plasma configurations. Substantial plasma heating and current drive capability is available from 17.5 MW of neutral beam heating, 5 MW (source) of Fast Wave heating and current drive, and 4 MW (source) of EC power. The DIII-D diagnostics set provides over 50 diagnostics systems capable of providing definitive measurements of plasma parameters in the core, edge, and boundary regions of the plasma. A summary of all major systems is shown in Table 10-2.

Table 10-2
Summary of All Major Systems

System	Description
Poloidal field	7.5 V-s OH transformer Eighteen independently controlled field shaping coils Fourteen phase controlled dc supplies 36 switching current regulators (2.5 kA)
Toroidal field	2.2 T on axis (1.695 m) for 5 s
Nonaxisymmetric field	
C-coil	Six external coils on midplane, $B_{2,1}$ ~ 5 G on q=2 surface Five phase controlled dc supplies (7 kA); Four switching current regulators (5 kA)
I-coil	Twelve internal coils above and below midplane, $B_{2,1}$ ~ 5 G on q=2 24 amplifiers (190 A, 0–20 kHz)
Vessel/first wall	Water-cooled inconel vessel, 90% graphite coverage
Vessel conditioning	350°C induction bake system for vessel walls Boronization, He glow cleaning
Fueling	Gas puffing — Eleven valves, 19 inlet locations, at 1–200 Torr-l/s ea. valve — Six fast valves for massive puff — combined 400,000 Torr-l/s for 5 ms — Pneumatic pellet injector 80 pellets at 10 Hz
Pumping	Three turbopumps at 5000 l/s each Three in-vessel cryopumps — one at 40,000 l/s, two at 20,000 ls
Prime power	Motor generators — 2.25 GJ @ 525 MVA 138 kV Xfmr — 20 MVA(CW), 110 MVA (10 s) (available late CY08)
Computers	Primarily Unix and Linux based, 4 GB raw data/shot, gigabit Ethernet
Cryogenics	150 l/h He liquifier, 11,000 gal LN_2 tank, 3000 gal LN_2 tank, 1000 gal He dewar

The DIII-D tokamak uses conventional water-cooled copper coils to provide the magnetic field configuration. The coil systems are designed to operate in a pulsed mode with the joule heat stored in the coil mass during the discharge and removed in the ten minute interval between discharges. DIII-D routinely operates at 2.2 T toroidal field and at 1.6 MA plasma current with a discharge flattop duration of 5 s (Fig. 10-1). Operation for longer duration at lower field and plasma current is also possible. Eighteen independently controlled poloidal field shaping coils provide a wide range of highly shaped, noncircular plasma cross sections. A set of six external picture frame coils mounted around the midplane (C-coils) corrects small magnetic imperfections arising from nonaxisymmetries in the coil systems and provides the capability to stabilize MHD instabilities. A set of 12 water-cooled internal picture frame coils (I-coils) mounted on the interior vessel surface (six above and six below the midplane) provides improved error field correction, and improved instability control, allowing control of the Resistive Wall Mode and Edge Localized Modes.

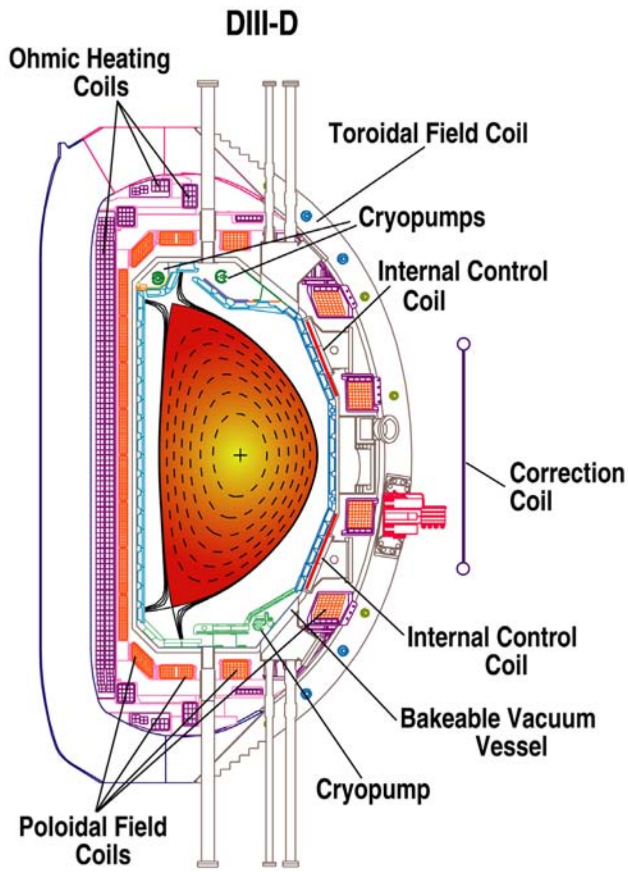

Fig. 10-1. DIII-D capabilities allow a wide range of research and technology issues to be addressed.

Graphite tiles cover more than 90% of the interior plasma-facing surface. The tiles absorb heat during the discharge and are cooled by water channels in the vessel wall in the period between discharges. In the high heat flux areas of the upper and lower divertor regions and centerpost, the edge-to-edge tile misalignment and tile gaps are less than 0.25 mm to reduce erosion and provide axisymmetry. Wall conditioning techniques include high temperature baking to 350°C, boronization (deposition of a thin boron layer during high temperature bake) prior to each operating period, and helium glow cleaning between discharges. These techniques enable rapid recovery of good plasma discharges following vents with personnel activity in the vessel and robust operation following plasma disruptions.

An extensive gas puff system and pellet injector provide the tools for plasma fueling. The gas puff system permits independent edge fueling with up to 5 different gases from more than 10 locations around the plasma including the inner wall and the upper and lower divertor regions. A 10 Hz pneumatic pellet injector provides fueling deeper into the plasma by injecting frozen fuel directly from the low field side or via curved tubes from the high field side. A pellet dropper to be installed at the end of the FY07 campaign will deposit pellets at the plasma edge to influence ELM behavior. Massive gas puffs used for disruption mitigation experiments are provided by a multi-valve injector in which each of six high speed valves are independently controllable and are designed to deliver fast rise time gas puffs.

Three in-vessel baffled, cryopumps provide pumping of neutral gas in both the upper and lower divertor regions. The two pumps in the upper divertor regions separately pump both the inner and outer strikepoints (S~20,000 l/s and 37,000 l/s respectively for D_2) of high triangularity upper single null or double null discharges. The geometry of the lower divertor was modified in 2005–2006 to pump the edge of high triangularity, single or double null divertor discharges (S~ 20,000 l/s for D_2), thus improving the density control in high triangularity Advanced Tokamak discharges. The new geometry consists of a water cooled shelf extending from the pump aperture to the outer baffle plate and permits the operation of lower triangularity divertor discharges at high power with the strikepoint(s) located on the top of the shelf. The pumps operate at liquid helium temperatures and actively pump both the D_2 fuel and all volatile impurities during the discharge. An argon frosting technique has been used to provide effective pumping of He.

The present and proposed capabilities of the three heating and current drive systems are summarized in Table 10-3. The seven neutral beams delivers 14 MW for 5 s or 17.5 MW for 3.8 s and are routinely used in most experiments for the primary source of heating and as a critical part of key diagnostic systems: charge exchange recombination for ion temperature, rotation speed, and impurity concentrations; beam emission spectroscopy for fluctuation measurements; and motional Stark effect for current profile and radial electric field measurements. Five of the sources are injected in the normal direction of the plasma current (“co-” sources) and two sources are injected in the counter direction. By pulse-modulating the sources and adjusting the mix between the co- and counter sources, the injected power and momentum can be continuously and independently controlled. This capability permits independent control of the plasma energy and toroidal rotation. By reversing the direction of the plasma current, experiments can be performed with a full range of co and counter injection. The electron cyclotron system presently consists of five long pulse (10 s) 1 MW class gyrotrons. The six-gyrotron system should be operational by early CY08. Approximately 70% of the power is delivered to the tokamak via low loss corrugated waveguide into six launchers all of which are independently steerable (poloidal and toroidal) between discharges. The 5 MW Fast Wave system consists of three transmitters; two (ABB1 and ABB2) are operated at 90 MHz and are capable of delivering 1.5 MW into a matched load, while the third system (FMIT) is capable of delivering 2 MW into a matched load. The ABB systems are connected to DIII-D via a coaxial transmission line into two 4 strap, water-cooled antennas with Faraday shields. The FMIT system is similarly connected to an inertially cooled, four-strap antenna whose energy handling is limited to 4 MJ or a 2 s pulse at full power.

Table 10-3
Auxiliary Heating System Power (Early CY08)

System	Present Power (MW)	Pulse (s)	Proposed Power	Pulse (s)
Neutral Beam	17.5	3.8 s	20 MW	10
Fast Wave (Source)				
— ABB (90 MHz)	3.0	10 s	4.0 MW	10
— FMIT (60 MHz)	2.0	2 s[a]	2 MW	10
Electron Cyclotron (Source)	5.0	5 s	12	10

[a]Limited to total 4 MJ by present antenna.

The plasma control system provides state-of-the-art high speed digital control of the magnetic configuration and other key plasma parameters. The system is capable of full integrated control of plasma shape, density, pressure, current profile, energy, and toroidal rotation as well as performing feedback stabilization on the neoclassical tearing mode and resistive wall mode.

A substantial number of other support systems are necessary to operate the facility. Prime power for the heating systems is taken directly off the utility grid while the power for the coil system is supplied by a 525 MVA motor generator. A second, smaller 260 MVA motor generator is presently mothballed. The coils are powered by a set of fourteen phase controlled power supplies. In the case of the shaping coils there are high speed switching current regulators (choppers) in series with each; the nonaxisymmetric coils utilize a combination of switching current regulators (0–4 kHz) and higher bandwidth amplifiers (0–20 kHz).

The computer systems for the facility are generally a variety of UNIX and LINUX based systems. The internal network is Gigabit Ethernet with a dual Gigabit link between DIII-D and the main computing center. The Fusion site is a node on ESnet. The data acquisition system routinely acquires over 4 GB per shot and on-line storage permits all present and historical DIII-D data to be available on a 30 TB high speed magnetic disk array for rapid access.

A closed loop, cryogenic system comprised of a 150 l/hr helium liquefier and two compressors provides liquid helium needed to support operation of the neutral beamlines and in-vessel cryopumps. The LHe used for the EC superconducting magnets and the D_2 pellet injector is produced by our helium liquefier, but is used in a once-through system and is not recovered. The LN_2 used for the beamlines and in-vessel cryopumps is purchased and is stored in an 11,000 gallon tank and an 3,000 gallon tank. A set of water conditioning systems provides high purity, low conductivity, deoxygenated water to cool the DIII-D vessel, coils, neutral beams, gyrotrons, power supplies, diagnostics, and other systems.

Operation of the tokamak with deuterium fuel results in significant neutron production. The radiation shield forming the wall and roof of the machine hall reduces the radiation levels to acceptable levels for the public and the staff. Radiation levels at the site boundary are limited to 100 millirem/yr by the state of California regulations and internally to 40 millirem/yr by DIII-D procedures. Radiation levels for staff are limited to 5000 millirem/yr by the state of California regulations and internally to 1600 millirem/yr (400 millirem/qtr) by DIII-D procedures. An active ALARA program keeps radiation doses from facility operation as low as reasonably achievable.

Presently, the radiation dose at the site boundary for a typical week of operation is 0.75 millirem. If the balance in the experimental program between high performance discharges producing high radiation dose and lower dose discharges remains the same, an extension of the typical pulse length by a factor of two would increase the typical weekly dose to 1.5 mrem. Thus the facility can be operated for roughly 26 weeks without exceeding our DIII-D procedures. Based on these average values, the radiation shielding of the facility is adequate to carry out the proposed plan, although the number of long pulse high performance discharges will need to be closely monitored. A more extensive discussion of radiation dose limitations on our proposed long pulse operations is included in Section 10.10.

10.2. OVERVIEW OF FACILITY OPERATIONS AND IMPROVEMENTS

The DIII-D research program has remained vibrant since its beginning in 1986 by constantly enhancing the device capabilities through the upgrades of existing systems and the addition of new systems. In addition, as the number of systems and their complexity has increased, improvements in system reliability and efficiency have been essential in order to maintain high device availability and enable effective use of the full array of systems. A comprehensive system of preventative maintenance and ongoing refurbishments has allowed us to maintain high availability despite the ever-increasing system complexity. The proposals outlined in this section for the next five years continue this legacy of refurbishment, increased efficiency, enhanced operational capability, and innovative upgrades to allow the DIII-D program to maintain a leadership position in the world fusion program. This section will summarize the proposed hardware changes required to achieve the goals described in the research program plan. A discussion of device operation and the proposed operating and upgrade schedule is included at the end of this section.

10.2.1. Tokamak Systems

The upgrades proposed in this area include the correction of the magnetic field error associated with the feedpoint to the TF coil located at 30 deg; the expansion of the water cooling systems to accommodate longer pulses and higher auxiliary heating power; and a helium gas recovery system for the EC superconducting magnets to address an on-going worldwide helium supply shortage.

10.2.2. EC H&CD System

The EC system is a key tool used for off-axis current drive critical to the AT program and for stabilization of NTMs. High system reliability and increased EC power are the main goals for that system. To improve the existing 6 gyrotron system, we propose to upgrade the existing HV supplies PS1 and PS2, enhance our thermal and fault monitoring systems, reduce transmission line power losses, add realtime control of the EC launchers, and expand the control of the gyrotrons by the experimental coordinator. In order to provide sufficient power for off-axis current drive and instability control required by the AT research program, we propose to increase the total source power to 12 MW. We will first increase the system power by 3 MW with the addition of two new high power depressed collector gyrotrons (P = 1.5 MW). We will lead the development of these tubes, leveraging the existing work done in the US and world gyrotron community. This upgrade from 6 to 8 gyrotron systems will require the additional of two full gyrotron systems (including HV tank, electronics, water, transmission lines, dual launcher) and the addition of one HV power supply. The full 12 MW capability will be obtained as we gradually replace the aging 1 MW class gyrotrons with the newer 1.5 MW units.

10.2.3. Neutral Beam Systems

The neutral beam system is presently the primary heating and current drive system used in DIII-D. This proposal will actually increase the role that the system will play in DIII-D research program by adding off-axis current drive to the system capability. Development of the neutral beam system will be focused in three main areas: refurbishment of the existing system including the restoration of the eighth source for additional power, modification of the beam system to enable off-axis injection, and extension of the full 20 MW beam system to a 10 s pulse.

The refurbishment effort consists of the following tasks: restore the eighth ion source, modernize/replace the existing CAMAC systems, rebuild 2 ion sources each year, and rebuild of the local control stations for the NB power supplies.

The extension of the pulse length from 4 to 10 s will require five subsystems to be replaced or upgraded: beamline internal components, drift duct monitoring system, beam diagnostic systems, water cooling systems, and the power supply system. The heart of the beam system, the ion sources, do not require any upgrade to extend the pulse length.

To provide off-axis beam injection, we propose to modify two beamlines to enable the injection angle to be adjusted. This will require significant modifications to the beamline, beamline stand, HV transmission line and support systems (water, cryo, vacuum). Preliminary studies indicate that to provide adequate clearance through the vessel port at the elevated angle, the ion source cross section will need to be reduced slightly and the source operating voltage increased to compensate for the reduced cross section.

10.2.4. Fast Wave Systems

The Fast Wave system is another key element in providing the noninductive current drive for our Advanced Tokamak discharges. The near term objective for this system is to increase the power and reliability of the system with a longer term goal to develop an advanced design for a high power, long pulse antenna. To increase the power, we propose to duplicate the tetrode upgrade on the ABB2 transmitter that was performed previously on ABB1. Recent analysis has shown that to realize the full 2 MW from the transmitter, upgrades of both the intermediate and low power stages will also need to be performed. Reliability issues are already being addressed and this effort will continue with the appropriate refurbishment of data acquisition and control hardware and increased spares inventory. The present FMIT antenna in the 285/300 deg port is not capable of delivering full power for more than 2 s, so a new long pulse design must be developed. We outline an R&D effort that will culminate in the installation of a low-impedance actively cooled wave launcher with an appropriate advanced ELM resilient matching scheme.

10.2.5. In-Vessel Coils

DIII-D has both axisymmetric field shaping coils and nonaxisymmetric coil sets that are used for magnetic error fields studies, feedback stabilization of the Resistive Wall Mode (RWM), and for the creation of a resonant magnetic perturbation (RMP) for ELM stabilization. Two different in-vessel coil sets are proposed to augment this existing coil set on DIII-D. The first is a nonaxisymmetric array of 48 coils to be used to produce an RMP with a flexible mode structure to study the physics of ELM stabilization. The second is an axisymmetric coil set to expand the flux in the divertor region for reduced heat load to the divertor plate. In addition to installing new coils, we propose to enhance the power and control system for the I-coil. A number of different options are proposed to permit stabilization of higher order mode number RWMs with independent feedback control of all 12 coils and simultaneous error field and RWM control using the I-coil.

10.2.6. Fueling and Disruption Mitigation

A higher fueling rate from the inner wall is desirable for pedestal physics studies and for simulating operational scenarios for ITER. We propose to accomplish this on DIII-D by modifying the existing DIII-D pellet injector for higher repetition rate capability and to modify the third gun to be appropriate for inner wall fueling. Steady state, 10 Hz injection capability will also be pursued using a screw extruder presently under development for ITER.

In the area of disruption mitigation, the DIII-D program has had excellent success at reducing high heat loads and halo currents. However, our present method of massive gas injection does not easily extrapolate to full runaway electron suppression for ITER because we have been unable to get a sufficiently high and rapid density increase. In the next five years, we propose to evaluate four different techniques for achieving sufficient density buildup: an modified gas jet that reduces the propagation delay of the injected gas; customized solid pellets consisting of room temperature prefabricated pellets with various layers of materials chosen to optimize density increase and impurity deposition; large cryogenic pellets injected using a newly developed ORNL pipe gun; and a cryogenic liquid jet with sufficient speed to penetrate to the core of the discharge. At least one approach will be pursued.

10.2.7. Divertor/First Wall Modifications

The major goals of the hardware upgrades proposed in this section are to address the research needs for improved pumping effectiveness and to enable the divertor and first wall to handle both higher peak injected power (30 MW) and total injected energy (300 MJ). The hardware changes proposed to enable these include: installation of a new lower inner divertor structure, an improved tile design, enhanced water flow to the divertor structure, upgrades to diagnostic windows and other in-vessel components for the anticipated high heat fluxes, and operation at elevated wall temperature. Thermal analysis has shown that for the highest heat flux areas the use of CFC tiles is essential.

10.2.8. 10 s Upgrade

The upgrade of the tokamak pulse length from 5 to 10 s for a 1.6 MA high performance, advanced tokamak discharge is a relatively modest effort because it primarily involves the utilization of the full thermal capacity already built into the poloidal field (PF) coil systems. This requires an expanded thermal monitoring and interlock system for the PF coils, a new dc power supply and additional current regulators for the PF coils, and upgraded components in the existing regulators. Extension of the heating and current drive systems to 10 s is also part of the plan and is described in separate sections for each system (EC/10.4, NB/10.5, and FWCD/10.6). The prime power requirement for operating the full set of auxiliary systems for 10 s is described in Section 10.11.

10.2.9. Prime Power and Coil Power Systems

The proposed increase in both power and pulse length of the heating and current drive systems for advanced tokamak research requires a significant increase in the site prime power capacity. Two transformers, a 138 kV/12.47 kV and a 12.47 kV/4160 V units, have been obtained through a collaboration with ASIPP from China. Once installed, these transformers will provide sufficient power to operate all heating systems for the full 10 s pulse length and for the additional water pumps required for their operation. In addition, we propose a reconfiguration of the prime power feeds for reduced cost and a refurbishment of the existing PF coil power supply control systems.

10.2.10. Computers, Data Acquisition, and Control

In order to keep pace with the ever increasing demand for more computing power, more storage space for data, more bandwidth with network communication we propose to continue our on-going program of system upgrades. We will increase the use of high-speed multi-processor Linux/Intel computers, investigate the use of Storage Area networks, expand the use of Gigabit Ethernet within the facility and introduce 10 Gigabit Ethernet link between DIII-D site and the main office site. We will also continue the replacement of our aging CAMAC data acquisition with newer, more powerful alternatives and expand this effort to replace the CAMAC-based control systems.

10.2.11. Operations and Improvement Schedule

We propose to provide 21 weeks of operations annually to the experimental program for the years 2009–2013 (an operating week is 5 days of single shift, 8 hours/day). Twenty-one weeks of operation provides an appropriate balance between carrying out the research program, the facility upgrades and improvements outlined in this proposal, system testing and commissioning, and equipment maintenance and repair (Fig. 10-2).

In a typical operating year, the 21 weeks of operation are performed during 31 calendar weeks with alternating periods of 3-4 weeks of experimental operations followed by 2 weeks of maintenance. These short maintenance periods are extremely important since they allow us to perform on-going maintenance and repair to maintain high availability, provide opportunities for installation and testing of new system throughout the year, and allow modification of existing systems to respond to changing experimental needs. Following the completion of the experimental program each year, there is typically an extended maintenance period of a two to three months to enable the performance of longer maintenance and upgrade tasks and permit new system installation and commissioning, both in-vessel and ex-vessel. The annual vessel openings are also used to perform routine diagnostic calibrations and alignments. There is typically a one month ‘cooldown’ period prior to the start of any extended in-vessel work in order to allow radiation levels to decay to levels that will permit useful work periods within the constraints of our radiation guidelines. At the end of an extended maintenance period involving significant in-vessel work or facility modifications/upgrades, there is a six week startup period. This includes a three week period that includes leak checking, high temperature baking, system testing and checkout, and new system commissioning. This is followed by two weeks of plasma cleaning operation and one week for diagnostic calibrations that require plasma operation. Excluding experimental operating weeks and the extended in-vessel work period, the device is typically operated with magnetic fields and/or plasma for an additional 60 days per year for system testing, diagnostic calibration, baking, boronization, plasma conditioning, and new system commissioning. This additional operation requires controlled access to the machine hall and is performed either on weekends, during normal maintenance weeks, on second shift during operating periods, or immediately following or preceding the major vessel openings and upgrade periods. Conditioning of new EC systems or Fast Wave systems into loads external to the machine hall are performed independent to device operation and are not counted in this additional operational time.

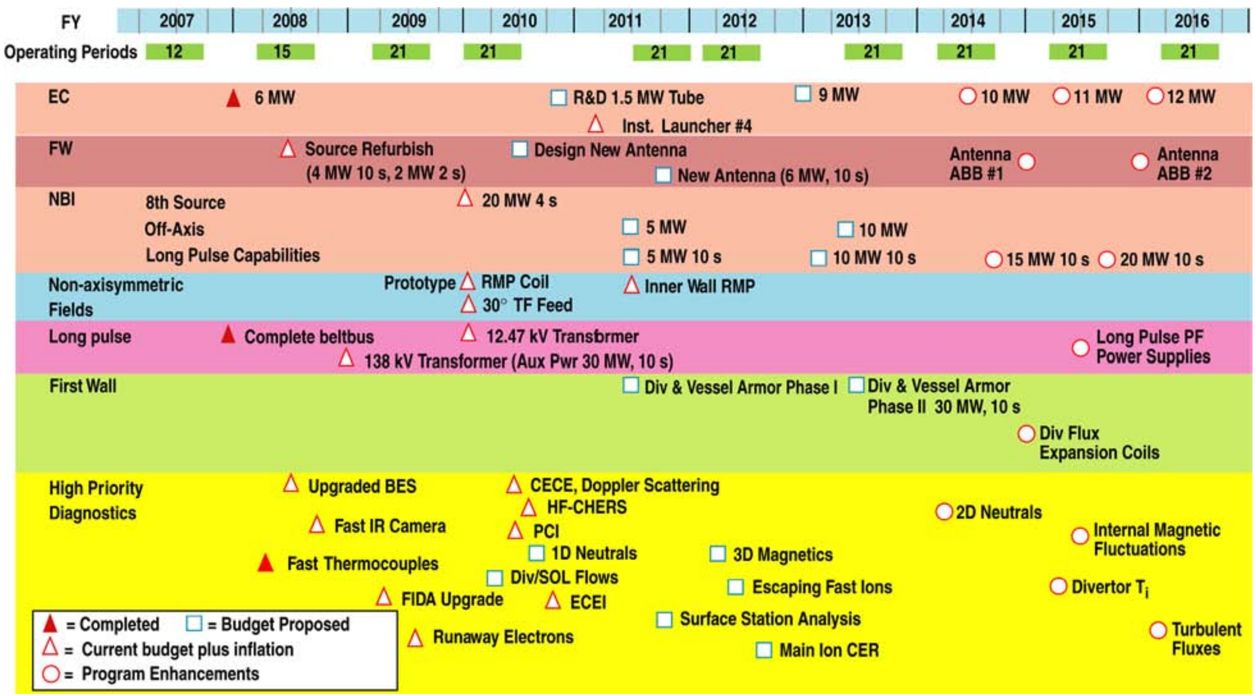

Fig. 10-2. Proposed operations and improvement schedule.

Some of the facility upgrades included in the plan (off-axis NB, long pulse NB, and inner wall RMP coil) require non-operating periods longer than the two to three month periods described above. Figure 10-2 shows the proposed operation and upgrade schedule with two such long maintenance periods. Each is similar to the 2005/06 schedule, during which the full experimental schedule was executed in early FY05, a long maintenance period was held crossing 2005/06, and experiments were again executed during the latter half of 2006. The longest of these periods is the first in 2010-2011 during which time we will install the new RMP coil system, upgrade the in-vessel divertor structures and armor tiles, and rotate one beamline for off-axis injection, and modify two beamlines for long pulse operation. The second period is shorter since the in-vessel work scope is much smaller. To enable us to achieve the desired operating time, sufficient resources are required to enable double-shift work during the upgrade period and extended shift experimental operations before and after the upgrade.

10.3. TOKAMAK SYSTEMS

10.3.1. Tokamak Coil Systems – Reduced Error Field

The DIII-D coil system consists of the axisymmetric main coils [toroidal field coil (TF), ohmic heating coils (E-coil), 18 field shaping coils (F-coil)], and nonaxisymmetric coils (6 external C-coils and 12 internal I-coils)]. The C-coils and I-coils were installed to reduce error fields resulting from nonaxisymmetric features of the main coils.

In this task, we propose to correct a significant fraction of the error field arising from the nonaxisymmetric nature of the 30° Toroidal Field current feed point, one of two such feeds on DIII-D. The TF feed point at 210 deg was redesigned during 2005–2006 and reduced the error field by a factor of ten (Fig. 10-3).

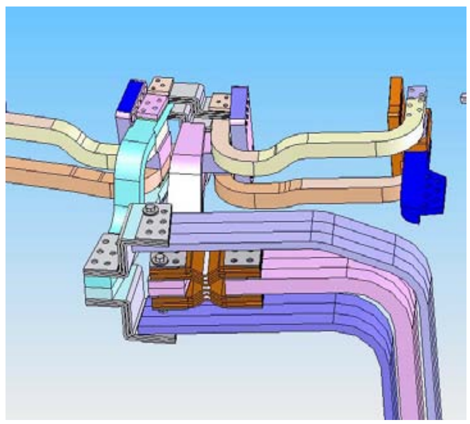

Fig. 10-3. Modified TF feedpoint at 210 deg reduced magnetic error field by a factor of 10.

This has brought significant benefits to the research program: the region of stable low density operation without locked modes was extended from $n_L = 1.2$ to $0.85 \cdot 10^{19}$ m^{-3}, a 30% reduction, and a reduction of external torque has enabled steady plasma rotation at low torque input and thus low velocity. A similar amplitude field error remains at the 30° TF feed point. Unlike these other sources of error fields from the F and TF coils, the feed point is spatially localized, and so it has a slowly decaying spectrum of higher-n Fourier harmonics. If the TF-coil 30° feed error were reduced several-fold, then the remaining DIII-D intrinsic error would be predominantly n=1 and n=2 from coil alignment and spacing errors. Such an intrinsic error is amenable to good correction by the C-coil alone.

We are proposing two improvements to provide significant error field reduction. The conductors in the vertical section of the buswork are spaced widely apart in a dipole configuration that contributes approximately half of the error field from the feed. This section is amenable to correction by reducing the spacing between conductors. The lower section of the feed consists of elaborate buswork that extends from the vertical conductor to the output of the TF-supply coax, and this contributes the other half of the error field. This section is not amenable to any simple realignment of conductors, but may be correctable using a bucking coil.

10.3.2. Cooling Systems

The water cooling system for the DIII-D facility consists of the vessel and coil cooling water systems, high and low pressure components cooling water systems (power supplies, ion sources, neutral beamlines, gyrotrons, and diagnostics), chilled water systems (soft x-ray, Thomson laser and vacuum turbo pumps) and the heat rejection system (cooling towers and heat exchangers) for the operation of the fusion facility. These systems are conditioned for ion and oxygen levels as required.

10.3.2.1. Cooling Tower Replacement. Three cooling towers are used on the DIII-D site. Cooling towers #1 and 2 were replaced in the 2003–2008 contract period. Cooling tower #3 provides cooling for the large motor generator and is over 20 years old. The upper half of the tower is made of galvanized steel and has corroded structural members. Either these members or the upper half of the tower needs replacement. The lower half of the tower is stainless steel and is in good condition.

10.3.2.2. Vessel and Coil Cooling System Expansion. This system supplies cooling water to the DIII-D vessel, the coils and the power supplies for the tokamak coil systems, and the neutral beamlines. Longer pulse operation of the neutral beamlines will require additional cooling for internal components. To meet these requirements and for plant availability reasons, an additional pump is necessary.

10.3.2.3. Low Pressure and High Pressure Systems Expansion. The upgrade to the EC system during the next five-year period (Section 10.4) from the six, 1 MW class gyrotron system to eight, 1.5 MW class gyrotrons will require an expansion to both the low pressure and high pressure cooling water systems.

The low pressure water system provides cooling water for the high voltage supplies for the neutral beams, fast wave, and EC systems. One additional HV supply and two upgraded supplies for the upgraded EC system will increase the water demand from approximately 4000 to 4600 gpm. The present pumping capacity provided by five pumps is sufficient to meet the additional flow requirements. A new heat exchanger for the low pressure water system polishing loop will be required.

The ECH high pressure water cooling system provides cooling water for the gyrotrons. The system in the old gyrotron vault will be renovated to accept two new gyrotrons and a spare socket to permit off-line testing of an additional gyrotron. Each new gyrotron or socket will have its own water cooling manifold equipped with instrumentation. Flow requirements are shown in Table 10-4. The present system consists of four pumps with an installed spare. To accommodate the full buildout of the system with eight higher unit power gyrotrons and a test stand, four new high pressure pumps, with associated PLC control and variable frequency drive motors, will be required. Additional plates will be added to the existing polishing loop heat exchanger.

Table 10-4
ECH High Pressure Water System Requirements

Gyrotrons	Required Water Flow (gppm)	Required Pumps
6–1 MW	3000 gpm	4 + spare
9–1.5 MW	6100 gpm	8 + spare

10.3.3. Cryo System – New Helium Recovery Loop

The cryo system supplies the DIII-D facility with liquid nitrogen and nitrogen gas from two LN_2 storage tanks, and liquid helium produced on-site using two helium compressors and a helium liquefier. The system provides in-vessel cryopumping using three pumps and beamline cryopumping using cryopanels in each of the four neutral beamlines. The cryo system also provides liquid helium and liquid nitrogen for the ECH superconducting magnets, and liquid helium and high-pressure helium gas for the deuterium pellet injector. The helium used in the beamlines and in-vessel cryopumps is recovered and reliquified. Most of the operation is controlled automatically by a PLC.

Supplies of helium in the US are becoming tighter and therefore more costly. Analysis of our use indicates the majority of our helium consumption is for once-through cooling of three of the ECH superconducting magnets. All remaining ECH magnets and new magnets utilize a closed loop refrigeration system. It is possible to develop a closed system for these three magnets to recover the He gas and monitor it for impurities before allowing it to return to the closed loop system. This proposed system will be considered for operational cost savings.

10.4. ELECTRON CYCLOTRON HEATING AND CURRENT DRIVE SYSTEM

The proposals described in this section address the dual goals of improving the operational efficiency, reliability, and power of the present EC system and providing a viable, cost-effective long term path to a significantly higher power EC system required for fully non-inductive high performance discharges.

10.4.1. Enhancements to Present System

The present DIII-D ECH system, consists of five 1 MW class, long pulse gyrotrons operating at 110 GHz, with the sixth gyrotron expected to be operational by early CY08. The six gyrotrons are powered by three high voltage supplies. ECH PS1 and PS2 each power two gyrotrons with a single modulator/regulator (mod/reg) capable of providing regulated power at 80 kV and 80 A. The third power supply, ECH PS3, has three independent mod/regs each providing regulated power at 80 kV, 40 A. Because of the limitation of the transformer/rectifier system on PS3, it is capable of powering two 1 MW class gyrotrons synchronously, or three gyrotrons asynchronously. Independent launchers provide flexible steering over the upper half plane of DIII-D both poloidally and toroidally for each gyrotron. The output power of the gyrotrons can be controlled by preprogramming the time dependence or actively by a signal from the Plasma Control System.

The objectives of the five year plan for the existing gyrotron system address the needs of the research program for higher power, longer pulse operation, real time control of the launch mirrors, and expanded control of gyrotron parameters and timing by the experimental coordinator. The plan also addresses the operational issues of system reliability, improved thermal monitoring diagnostics, more sophisticated fault processing for improved system protection, and expanded automation.

One of the primary goals is to increase the injected power. For reliable long pulse operation, generated power for the six-gyrotron system is expected to be a little above 5 MW, with about 4 MW injected. The output power of the system can be increased by either raising the power to the limits of the gyrotrons and/or by improving the transmission line efficiencies.

10.4.1.1. Increased Output Power. The output power of the two gyrotrons connected to PS1 and PS2 is presently limited by a parasitic oscillation in the tetrode when the two gyrotrons are operated near their maximum power ratings. To enable the gyrotrons to reliably operate closer to their maximum ratings, and to allow independent control for increased scientific flexibility, we propose to retrofit PS1 and PS2 with two mod/regs, one for each gyrotron, duplicating those in PS3.

The transmission line efficiencies for the present lines are estimated at between 75% and 80%. Although the waveguide sections are very efficient, each miter bend has about 1.5% loss due to mode conversion and resistive losses. Each line has about 13 miter bends. We plan to decrease the number of miter bends and reduce the mode conversion losses at many of the miter bends by installing pairs of specially designed couplers. Using both these techniques, the losses in the overall transmission line system can be reduced by almost 50%, potentially gaining over 500 kW of injected power.

10.4.1.2. Longer Pulse Operation. The DIII-D gyrotrons routinely provide 5-s pulses, with limited testing out to longer pulses. Motivated by the physics requirement for 10-s pulses and with the participation of the gyrotron manufacturer, CPI, pulse extension tests will be performed to study the long-pulse thermal equilibrium of the present gyrotron design at full parameters. The goal is to extend the pulse lengths to the design value of 10 s. Vacuum pressures and temperatures will be monitored along the waveguide lines to evaluate the need for additional cooling or vacuum pumping. Also, scaling the present temperature data on the mirrors in the dual launchers to longer pulse lengths indicates that an improved design with active cooling will be required.

10.4.1.3. Expanded Monitoring, Control and Fault Protection. Because of the concerns associated with cyclic thermal stressing of the collectors on the gyrotrons, a collector thermal monitoring system has been implemented for the DIII-D gyrotrons. It has 160 channels for making a detailed thermal map of the collector of any gyrotron, either as part of the commissioning process or as a routine diagnostic. It also provides a modest number of channels for routine collector monitoring of each gyrotron during normal operation. As we gain experience with the new system, it may be expanded to permit routine thermal scans at higher spatial resolution for each gyrotron.

It is desired to be able to move the mirrors in the launchers in real time during plasma discharges to enable more efficient utilization of experimental time, and to open a new regime of experiments by actively steering the rf beams as the plasma equilibrium changes for feedback stabilization of MHD modes and tracking of q-profiles. A drive mechanism for fast motion of the mirrors was designed into a launcher and tested. However, the air turbine motors must be replaced with dc motors to improve performance, and suitable drive algorithms need to be developed to control the motors and protect the apparatus.

As the operational flexibility to support physics was enhanced and new control features were added, it became clear that greatly improved, more sophisticated fault handling was necessary to provide proper protection of the gyrotrons. A new Field Programmable Gate Array (FPGA) fault processing system has been developed and implemented on one of the gyrotron systems. The FPGA system will be implemented on all the gyrotrons systems.

Finally, for increased efficiency, we plan to automate a number of the control functions related to operation of the ECH system for physics that are currently being performed manually. For example, the calculation of desired trajectory of the injected rf is performed between discharges and the necessary control parameters, such as polarization and mirror scan angle are entered by hand into the control system and subsequently into the DIII-D database for storage. This process will be fully automated. In addition, the settings for timing, modulation, and other parameters will be entered directly by the experiment leader from the DIII-D control room into the gyrotron control systems.

10.4.2. EC System Power Upgrade

The research program for the next five-year period will require continuing the growth in gyrotron power and pulse length that led to the current EC system to 12 MW (source) by taking advantage of the worldwide progress in higher gyrotron output power and efficiency. Pursuing the higher power gyrotron is a more cost-effective path to higher system power than doubling the number of 1 MW gyrotron systems because it minimizes the need for additional HV power supplies, gyrotron sockets, transmission lines, launchers, and DIII-D ports. Using 1.5 MW higher efficiency gyrotrons, only one additional HV supply, two gyrotron sockets, two transmission lines, and one dual launcher are required. An expansion based on 1 MW gyrotrons would require six additional sockets, transmission lines, three dual launchers, and three additional high value ports on DIII-D.

Figure 10-4 shows the plan to increase the EC system power based on the 1.5 MW gyrotrons. In the first phase, two additional 1.5 MW gyrotrons would be procured increasing the system power by 3 MW. Then additional 1.5 MW units would be procured as spares eventually bringing the total power to 12 MW as the 1 MW units are phased out.

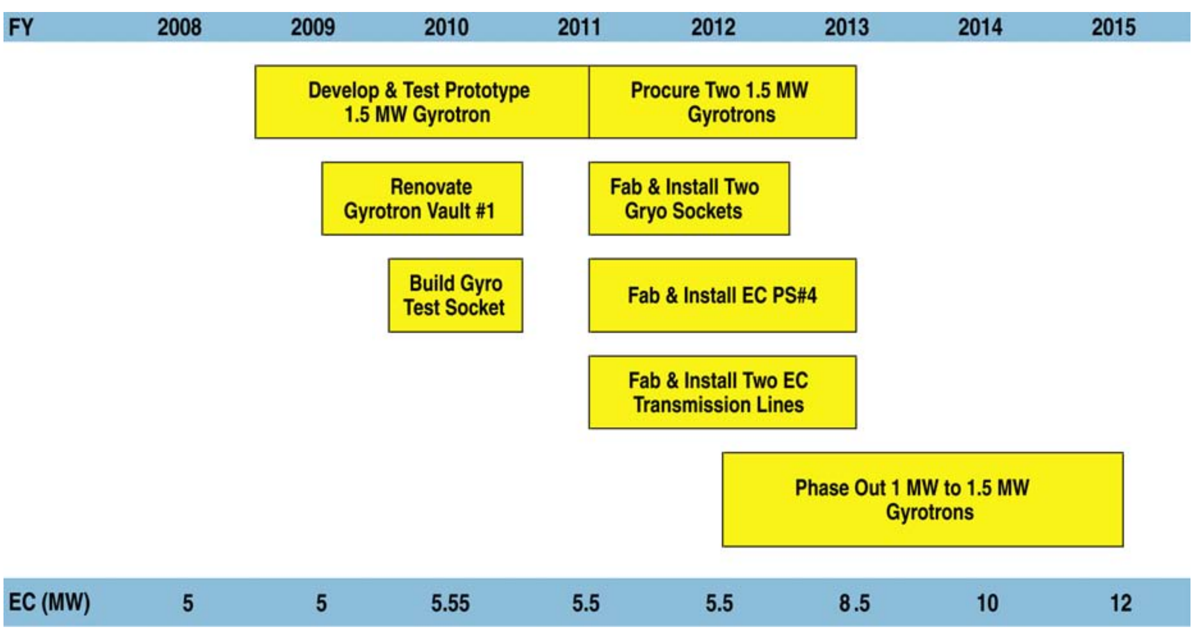

Fig. 10-4. Plan to increase the EC system power to 12 MW.

10.4.2.1. Gyrotrons. GA will lead the development of a higher power, higher efficiency depressed collector gyrotron that is focused on the reliable support of physics. This development will leverage the recent progress worldwide in gyrotron performance with a goal of attaining the highest reliable output power available. A gyrotron capable of generating a robust 1.5 MW long pulse is realizable and is consistent with the capability of our existing power supplies. A prototype will be built and then tested at GA in a gyrotron test socket using the unused third modulator-regulator in ECH PS#3. Upon satisfactory demonstration of the gyrotron performance, two new gyrotrons will be procured. A cryogen-free superconducting magnet and its associated power supplies will be procured with each new gyrotron.

The old gyrotron vault #1 at DIII-D will be renovated to accept the gyrotron test socket and two new gyrotron sockets as shown in Fig. 10-5. Each socket has a high voltage tank, water cooling manifold, and a gyrotron instrumentation and control subsystem, all of which will be essentially copied from the existing DIII-D EC system. The instrumentation and controls for the test socket and two new gyrotron sockets will be l located in the unused control room for gyrotron vault #1.

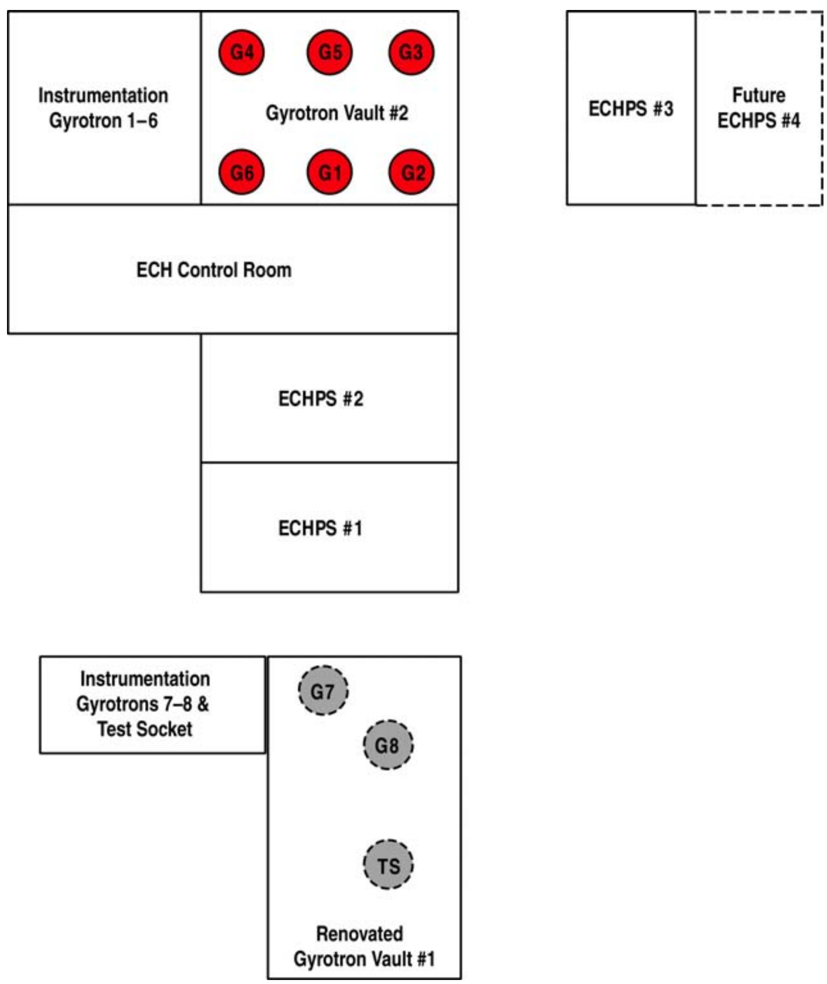

Fig. 10-5. Upgrading EC system towards 12 MW with only eight gyrotrons. Only one additional power supply, ECH PS4 is required.

Additional 1.5 MW gyrotrons would be procured as spares for the existing 1 MW gyrotrons, some of which will be over 10 years old by 2013. The spares would be conditioned in the test socket without interrupting physics support so that ready-to-run replacement gyrotrons would be available. Changing out the 1 MW gyrotrons with new 1.5 MW higher efficiency gyrotrons will increase the EC system power to 12 MW without further investment in costly EC system hardware and without raising operating costs.

10.4.2.2. Power Supplies. The current EC H&CD system has three EC power supplies, one of which has three tetrode-based 80 kV, 50 Amp modulator-regulators in it. A fourth EC power supply, which will have two modulator-regulators, will be built and installed in the location shown in Fig. 10-5. It will use one of the on-site MFTF neutral beam power supplies for the HVDC input to the modulator-regulators. The other two existing EC power supplies are to be refurbished, as described in Section 10.4.1.1, so that each of the two gyrotrons currently connected to these two power supplies will be energized by an individual modulator-regulator. This will give the EC system the capability to run eight gyrotrons independently from four EC power supplies in support of physics while at the same time being able to condition a spare gyrotron without impacting the research schedule.

10.4.2.3. Transmission Lines. Two new transmission lines will be fabricated and installed to route the rf power from the new gyrotrons to the fourth dual launcher to be installed on DIII-D. These lines will use the same components as the existing lines which are capable of safely transporting 1.5 MW.

10.4.2.4. Launchers. A fast-steering dual launcher will be fabricated and installed in a port already reserved for it, expanding the current capability to inject rf power from eight gyrotrons.

10.5. NEUTRAL BEAM SYSTEM UPGRADES

10.5.1. Refurbishment of Present System

The main focus of the refurbishment activity over the next five years will be to maintain reliable operation of the seven operating neutral beam systems and restore operation of the eighth ion source. To maintain reliable operation, several subsystems will require refurbishments. The systems most in need of modernization or replacement include the instrumentation and control system and the Local Control Stations (LCSs) for the HV supplies. Spare parts for the ion sources are needed routinely, and the replacement of several power supply tetrodes will be required.

10.5.1.1. Ion Sources. Six of the seven operating sources have been operating continuously since the beginning of DIII-D operation in 1986. Following a series of water leaks in a number of the sources early in the 2003–2008 contract period, an R&D program was launched to redevelop the technology to produce the necessary accelerator grids. A seventh source was partially rebuilt in 2005 with a modified set of grid rails with a tear shaped cross section replaced by a circular cross section. Tests of the new source demonstrated good performance and that source has been in service for two years. Several accelerator grid modules have been fabricated and assembled into two spare sources, the first of which will serve as the eighth ion source. Two full sets of 16 modules, sufficient for two ion sources, will be produced each year during this contract period to complete the refurbishment of the original eight ion sources. A separate R&D effort is on-going to produce spare Langmuir probes required for proper ion source operation.

10.5.1.2. Restoration of Eighth Neutral Beam Source. The rotation of the two sources on the 210 deg beamline from a co- to a counter-injection beamline increased the need for the restoration of the 30 deg right source as the eighth source and the sixth co-injection source. This task requires the remounting of the spare source described above; the fabrication and installation of a new pole shield for the bending magnet on the 30Rs source; and restoration of the UVC#7 HV power supply that is presently used to power ECH PS3, including modernization of the control interface to the UVC#7 supply. Returning the UVC#7 supply to the NB system requires replacing that system, which is presently used as the HV front end for EC PS#3. This task involves providing a new HV crowbar and HV switch gear and connecting a spare MFTF HV transformer to the EC PS#3.

10.5.1.3. Data Acquisition and Control. As part of the on-going phaseout of all CAMAC-based systems in the facility, the NB data acquisition systems were replaced with modern non-CAMAC equipment in FY08. The NB instrumentation, mode control, and timing system represents the last major NB system requiring CAMAC replacement. The different hardware options available are now being evaluated.

10.5.1.4. Local Control Stations. The interface electronics to the seven high voltage tetrode power supplies currently in use are very old and based largely on wire-wrap vector board technology. To reduce system maintenance and increase reliability these LCS interfaces and distributed circuit boards around the power supplies will be replaced by a modern Programmable Logic Controller (PLC) and a graphical user interface (GUI). As part of the task of restoring operation of the eighth ion source, a prototype LCS will be designed and built for this source. This prototype system will then be duplicated on the remaining seven systems as part of the on-going system refurbishment.

10.5.2. Neutral Beam Long Pulse Upgrade – 20 MW, 10 s

10.5.2.1. Present Status. The DIII-D neutral beam system with seven ion sources on four beamlines has been operated to support physics experiments with high reliability for 20 years. While the sources routinely operate from 75 to 81 kV, three ion sources have operated as high as 93 keV beam energy, much higher than the designed value of 80 keV. With improvement in the operation technique the injected deuterium beam power was increased from the designed value of 2 MW per ion source to 2.5 MW at beam energy of 80 keV. However, beam pulse length was reduced as beam power was increased due to heat handling limitation of the beamline hardware. At 81 kV, the pulse length is limited to 3.8 s. The purpose of this upgrade is to extend the beam pulse length to 10 s while maintaining high beam power of 2.5 MW per source for a total energy injected of 200 MJ from the beam system.

To achieve 20 MW, 10 s deuterium beam operation five neutral beam subsystems need to be replaced with new hardware or upgraded: beamline internal components, drift duct (the section connecting beamline to the tokamak vessel), beam diagnostic system, beamline water cooling system, and power supply system. The ion sources are actively water-cooled and do not require any modifications.

10.5.2.2. Beamline Internal Components. Beam collimators, magnet pole shields and beam dumps, *etc*., were designed and built for hydrogen beam operation of 80 keV, 5 s pulse length with injected power of 1.8 MW per ion source. These components are all passively cooled by de-ionized water and cannot handle the heat generated by the deposited energetic ions or neutrals from 10 s ion source operation at 80 keV. New internal components with active water-cooling need to be designed, fabricated, and installed.

10.5.2.3. Monitoring and Protection Systems. For the long pulses and high power proposed for the upgraded beam system, it is necessary to upgrade beam protection systems for monitoring temperatures of beamline internal components, drift ducts from the beamline into the vessel, and the ion sources.

Most of the energy deposited on the drift duct comes from the reionized energetic neutrals during beam injection into plasmas. A photodiode system monitoring the H_α light emitted from the reionized neutrals serves very well in protecting the drift duct from overheating for short pulse beam operation. For long pulse operation, a more direct method of monitoring the drift duct heating is needed. In addition to a more complete set of thermocouples to monitor drift duct temperature, a water-flow calorimetry system to determine the total energy deposited on the drift duct will be implemented, similar to the system used to monitor energy deposition to internal beam components.

10.5.2.4. Power Supply Systems. The power supplies for the NB systems were provided by two different vendors, UVC and Transrex. The I^2t ratings of the suppressor, arc, and filament supplies from both vendors indicate that these supplies can be operated for 10 s pulses with 20 minutes between discharges at power levels appropriate for our standard 80 kV deuterium beam operation. Similarly, the acceleration supply from UVC also has a sufficient rating for our desired pulse lengths. However, the capability of the Transrex acceleration supplies (for 150 deg and 210 deg ion sources) have a large uncertainty and measurements will be needed to confirm that they have the necessary rating. Tests have shown that the existing gradient grid resistor (voltage divider between the plasma and gradient grids) is limited to 5 s of 80 kV beam operation and will require improved cooling. Preliminary investigation of other equipment in the power supply system such as transformers, rectifiers, and ion source transmission lines indicate they have adequate cooling but this will be confirmed by more detailed analyses.

10.5.2.5. Water Cooling System. Existing beamline internal components are passive cooled by 100 gpm per beamline. Water flow rates will need to be increased to meet the active cooling requirements for 80 kV, 10 s beam operation. Measurements of energy deposition to the beamline drift ducts will be needed to determine if the existing water-cooling system is adequate.

10.5.3. Off-Axis Neutral Beam Injection

DIII-D is equipped with four neutral beamlines (30, 150, 210, and 330), all of which inject horizontally from outside midplane ports. Three of the beamlines, 30, 150, and 330 are oriented to inject in the same direction as the standard plasma current direction (“co-injection”), while the fourth beamline, 210, was reoriented in 2006 to inject in the counter direction (Fig. 10-6). The three co-injection beams provide considerable co-current drive while the counter beam drives current in the counter direction. In both cases, the current is predominantly driven near the plasma axis since the beam is aimed along the midplane and deposition is highest in the central region of the plasma.

Significant off-axis current drive is required for steady state AT discharges and this can be provided by high power ECCD, However, the efficiency of the ECCD decreases as one moves off-axis to lower temperature regions. Theoretical studies and simulations have shown that off-axis neutral beam injection has the potential of supplying substantial off-axis current drive with efficiencies comparable to the EC system. Pursuing off-axis injection by increasing the toroidal rotation of the beamlines in the horizontal plane so the beams are aimed well off-axis is not possible in DIII-D because the TF coils block access beyond their present injection angle of 19 deg off-radial (see Fig. 10-6). Off-axis neutral beam current drive can be achieved by maintaining the present ion source configuration and midplane port injection and rotating the beamline vertically so the source end is raised by up to 2 meters vertically, providing an injection angle of 15 deg (Fig. 10-7).

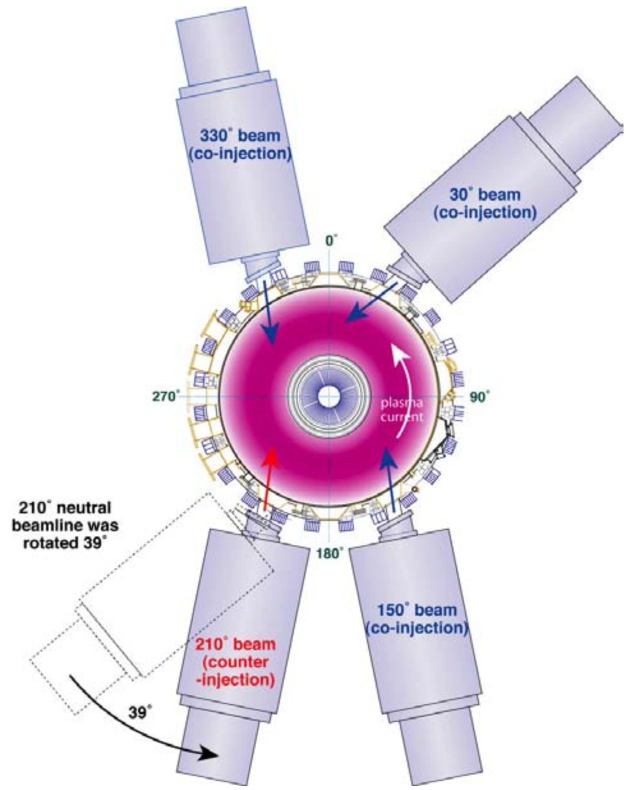

Fig. 10-6. Location around DIII-D of the four neutral beamlines. The 210 deg neutral beamline was rotated for counter-injection in 2006. Two of the other three beamlines are being evaluated for off-axis injection.

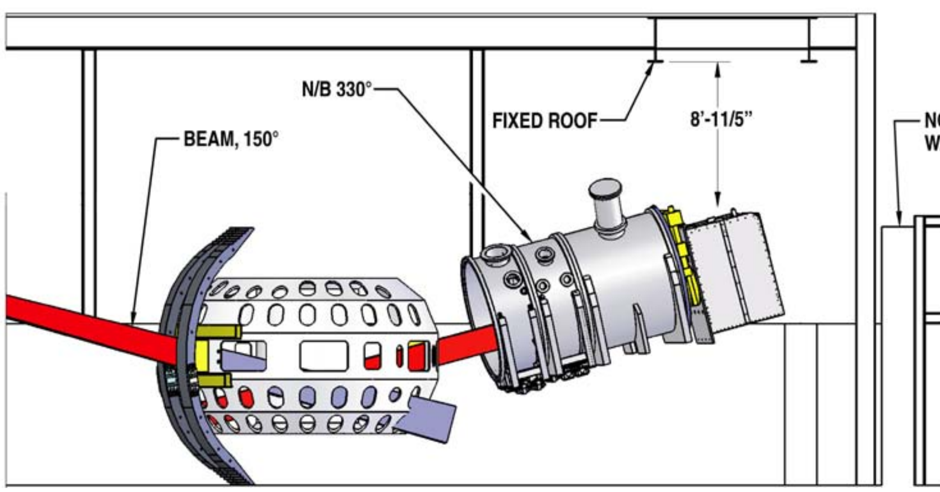

Fig. 10-7. Elevation view of DIII-D machine hall showing two beamlines with ion sources elevated, 150 deg on left and 330 deg on right. There is sufficient room between the beamline and the walls and neutron shield. No details of the beamline-vessel interface is shown.

We propose to modify either one or two of the co-injected beamlines to achieve off-axis injection while retaining the existing on-axis horizontal injection capability. This is a major engineering undertaking impacting several systems of DIII-D, but the recent rotation of one of the beamlines in the toroidal direction provides us with valuable experience. Two different approaches are being evaluated. The first is to provide a fully flexible system in which the beam angle can be adjusted in a relatively short timescale (< 1 day) without the need for venting the machine. This requires the beamline, HV transmission line, and all major elements attached to the beamline to have flexible attachments. The space required for the large beamline bellows requires the beamline to be moved radially away from the vessel and this presents different engineering issues for each beamline being considered. The second approach is to design fixed angle elements that would require more time for removal and reinstallation and would likely require a vent of the DIII-D vessel. This approach would significantly simplify some of the engineering challenges associated with the large flexible elements, but would restrict the experimental flexibility of the system.

A preliminary study was performed to examine the beam layouts with respect to DIII-D vessel port, toroidal magnet coils, and F-coil box beams for various injection angles. The results of the study indicate that it is feasible to inject off-axis beams into DIII-D using the vertically rotated beamline. Figure 10-7 shows the vertically rotated 330 deg beamline with respect to the tokamak and neutron shield roof and walls inside the DIII-D machine hall. The figure indicates that for an injection angle of 14.8 deg, there is sufficient room in the machine hall for the beamline elevation. There is also sufficient room between the beamline and the wall of the machine hall to permit the insertion of a bellows between the vessel and beamline with the resulting radial displacement of the beamline. However, the decreased room between the beamline and the wall increases the difficulty in the design of the flexible transmission line at the 150 and 330 locations.

Assuming the same initial cross section of the beam and the calculated beam divergence, there is a modest interference with the top of port as the beam enters the port and the vessel wall at the bottom of the port as the beam exits the port. A reduction in the beam height by 10 cm, by masking the extraction area of the ion source would provide the same clearance between the beam and vessel surfaces as the existing beam path. Operation of the ion source at 88 kV rather than 80 kV can restore the full beam power of 2.5 MW. The source and power systems have design margins sufficient to operate to this higher voltage and we have successfully operated the DIII-D neutral beam ion sources at 93 kV.

The preliminary study identified all affected systems and hardware. These systems include beamline tank and support stand, transmission lines for powering the ion sources and bending magnets, beam diagnostic systems, beamline cryogenic system, beamline vacuum system, beamline and ion source cooling water system, gas system, instrumentation and control system, tokamak anti-torque structure, in-vessel beam dumps, and physics diagnostic systems. While the solution to many of these issues is common to all of the modified beamlines, the beamline interference with surrounding equipment, the design of a flexible HV transmission line with the limited space, and diagnostic impacts are different for each of the beamlines and will affect the choice of beamlines and the final engineering solutions.

Since presently all beamlines possess one beam-based diagnostic, the primary impact would be the potential loss of a system when the beam is in off-axis (tilted) position. However, in some cases it will be possible to regain the capability by transferring the view to a different (and not tilted) beamline. Presently, no plan exists for any potentially impacted diagnostic to be modified in a way to follow the beam path (*i.e.*, tilted). No capability is lost when the beams are in the untilted (normal) position.

Shown in Table 10-5 is a list of impacted systems for each beamline option. Primary impacts include loss of signal whereas secondary impacts include potential physical interference (*i.e.*, outside the torus) or changes in first wall interface.

Table 10-5
Impacts of Beamline Tilting on Diagnostics for Various Options

30˚ Beamline		
Primary	MSE	Loss of all core chords
	CER	Loss of all core chords
	SPRED	Loss of active CX chords (core)
Secondary	None	
150˚ Beamline		
Primary	BES	Loss of all core chords
Secondary	Divertor Thomson	Increased background
	IR camera	Relocation of external hardware (cryostat)
330˚ Beamline		
Primary	CER	Loss of Edge Tangential chords
	CER	Loss of Core Vertical chords
	B Stark	Loss of all chords
Secondary	210˚ pyrometer	Shine-through potential impact

10.6. FAST WAVE SYSTEM

10.6.1. Refurbishments and Reliability Enhancements for Present System

The goal of the work described here is to increase the reliability of operation of the 5 MW Fast Wave (FW) system into AT-relevant DIII-D plasmas. Each of the three FW systems consists of a HV power supply, a multi-stage transmitter, a transmission line network that incorporates power division, impedance matching and phase control functions, instrumentation and control circuitry, and a four-element phased array antenna. Upgrades to the antennas are discussed in Section 10.6.2; this section describes the work on the other components in the systems.

10.6.1.1. HV Power Supplies. The HV power supplies for the three transmitters are located outside, and some of the problems experienced in operation have had to do with effects of moist air. Efforts to weatherproof the supplies that began in 2004 will continue through the early part of the five-year period. The Programmable Logic Controllers (PLCs) in the power supplies must be replaced with current technology since they are obsolete and can no longer be maintained or repaired.

10.6.1.2. Transmitters. Considerable effort has been invested over the past several years to first recommission the ABB transmitters (ABB1 and ABB2) to operate at power levels above 1 MW, and then to convert ABB1 to use a CPI Eimac tetrode in its final power amplifier (FPA) stage to permit operation at high frequency up to the 2 MW level. Recently, successful operation of the ABB1 transmitter at

~1.5 MW and of the ABB2 source (still using the Thomson TH526 tetrode in its FPA) to ~1.4 MW has been achieved at the desired frequencies of 90 MHz and 88.8 MHz respectively. In arriving at this power level, requirements for upgrades of these systems to enable extension of the operating power to ~2 MW have been identified. To achieve the condition in which the power-limiting elements of the FW systems are only the antennas, the following steps are proposed:

1. The upgrade of the ABB2 FPA to use the CPI Eimac tetrode should be completed.
2. The lower power stages in the amplifier chains, which are the Intermediate Power Amplifier (IPA) and Driver stages, should be redesigned to support higher power operation and the obsolete components replaced.
3. The FPA cavities should be reconfigured to provide lower load impedance to the tetrodes.
4. The control systems should be updated to modern standards.

10.6.1.3. Transmission Lines. The transmission line for the 285/300 system is the world's only ICRF system without tuners, with the only adjustable element being the decoupler stub. Upon installation of a new 285/300 antenna (described in Section 10.6.2), the feed scheme will almost certainly need to be substantially modified, depending on details of the new antenna. At that time, dc breaks similar to the upgraded ones developed for the 0 deg and 180 deg systems will be installed on the 285/300 system.

The 0 deg and 180 deg transmission lines incorporate 10 adjustable tuners in each system. The drive systems for those tuners need to be rebuilt to improve reliability, particularly in the 0 deg system, in which the tuners are installed vertically. We may also consider simplification of both the 0 deg and 180 deg transmission lines in a manner similar to the 285/300 system, which would increase the reliability of the system by removing troublesome moving parts, at the cost of some flexibility in tuning to a range of loading conditions. However, we may decide to upgrade tuners and their controls for reliable operation. In particular, we will consider different ELM resilient matching schemes if it proves to be necessary to optimize the system for use in AT plasmas.

Another increment in reliability could be obtained by completion of the work that was carried out in the early 1990s to upgrade the insulators in the 0 and 180 deg transmission lines to quartz pins — the lines were originally delivered with ceramic pins, which subsequent testing showed are inferior to quartz pins. Consequently, quartz pins were implemented on the most critical sections of the lines. The remaining sections of the lines still use the original ceramic insulators.

Finally, the introduction of a simple transmission line patch panel between the two transmitter's outputs would allow a number of possible configurations (one ABB transmitter feeding both antennas, both ABB transmitters feeding one antenna for power limitation studies, either transmitter feeding either one antenna) which would provide a further increase in reliability by allowing operation of both antennas even if one of the two ABB transmitters is down for repair or upgrade.

10.6.1.4. Instrumentation and Control. The original 25-year-old CAMAC-based data acquisition systems on the 0 deg and 180 deg systems were replaced in FY07 with a modern PXI-based system which has significantly improved the reliability and availability of the FW s systems. A similar upgrade for the 285/300 systems is planned for the near future. In the five-year period, we plan to implement realtime digital control of all three transmitters with further extensions of the PXI-based system, replacing decades-old analog hardware which has begun to fail.

The arc protection system in use on the 0 deg and 180 deg systems was recently replaced with an improved simplified system which is intended to be temporary. Only the most important sensors are used in this simplified system; the full complement of available sensors will be incorporated in the final arc detector system, which should be fully implemented as soon as possible. Incorporation of other inputs (acoustic sensors on the transmission lines, harmonic and subharmonic detectors, optical fibers at the feedthroughs, *etc*.) into the arc detection system will be studied.

The rf phase and amplitude measuring system is based on a module designed in the mid-1980s (on the FMIT/285/300 system) and in the early 1990s (on the ABB1/0 deg and ABB2/180 deg systems). These rather complex systems will be replaced with simplified, compact, modern modules that take advantage of the proliferation of cellular telephone technology. The present complex system is difficult to maintain (many of the required spare parts are unavailable) and keep in calibration; a simpler system would thereby improve reliability.

10.6.2. Advanced Antenna Design for Long Pulse and Effective Coupling

In a series of experiments that began in 1991, DIII-D has demonstrated efficient central electron heating and current drive with Fast Waves (FWs) in the 60–120 MHz range. These experiments have been performed with conventional FW launchers, consisting of four-element phased arrays of loop antennas. The three arrays are of two different designs, dating from 1989 and 1992. The 1989 design (the 285/300 array) is uncooled and hence limited to pulse lengths of 2 s at 2 MW; the 1992 design (embodied in the 0 deg and 180 deg arrays) is water-cooled except for the Faraday screen elements and is capable of 10 s pulse length at 2 MW. Empirically, neither array design is capable of coupling 2 MW to plasmas in H-mode regimes with outer gaps in the 6–10 cm range without unacceptable rates of antenna breakdown (“arcs”).

Since all FW antennas in use on tokamaks everywhere are limited in reliable power handling by the same phenomenon of rf voltage breakdown, the development and demonstration of a vastly improved FW coupler is clearly necessary for both the DIII-D program and for ITER and future machines. In fact, on DIII-D and on all present-day divertor tokamaks with ICRF systems, in H-mode plasmas with moderately large outer gaps the antenna breakdown limit sets the limit on the power that can be coupled, not the transmitters or transmission lines. It should be noted that ITER will require reliable FW coupling at outer gaps in the 15–20 cm range, in a similar frequency and wavenumber regime in which the DIII-D FW experiments operate; recent calculations show that the present ITER FW antenna design will have to reliably operate in a range of rf voltages far above what any experiment has demonstrated.

Three possible approaches to the solution of this problem are:

1. Gaining a radically improved understanding of the physical mechanisms underlying the breakdown seen in all present experiments in order to learn how to raise the reliable operating rf electric fields to the necessary levels.
2. Significantly modifying the plasma parameters in the antenna near-field region to lower the voltage at which the required power levels can be coupled. Examples of such approaches include

localized gas puffing (under study in JET) or the introduction of a localized low-temperature plasma source.

3. Lowering the impedance of the launching structure so that the peak electric field in the tokamak "vacuum" is significantly reduced for a given antenna current. The simplest ways of achieving this are to increase the number of antenna elements, toroidally (as in the "combline antenna"), poloidally (in what has been called variously the "stacked stripline" or "poloidally segmented" launcher design), or both.

Our proposed program would include elements from all three approaches, while concentrating primarily on the third one (low impedance coupler).

1. While the present antennas are in use, high speed electrical diagnostics of those existing antennas will be improved, which can be done at low cost by extending the new data acquisition system that was installed in FY07. Along the lines of the first approach, specific diagnostics for the FW system can be broken down into four categories according to their primary uses: (a) antenna and system protection from damaging arcs, (b) identification of maintenance issues, (c) antenna performance optimization, and (d) improved understanding of antenna/plasma interactions and rf coupling. ORNL proposes upgrades to the FW-specific diagnostics in each of these four areas, including development of advanced arc detection systems, arc localization in the transmission lines, pressure monitoring in the antenna structures, optical diagnostics in and around the antennas, and redeployment of edge reflectometers for high resolution (both in time and in space) of the local density profiles in the near-field zone of the antennas.
2. We will follow the results of the gas puffing experiments on JET in detail, with a possible follow-on experiment on DIII-D if those experiments show promising results, and evaluation will be made of improving pumping in the antenna structures. The antenna diagnostics proposed above are also an essential part of this work.
3. Replace the uncooled 285/300 antenna with a low-impedance, long-pulse antenna. Preliminary work has been performed at ORNL in which a replacement launcher for 285/300 with poloidally subdivided straps and an internal matching capacitor has been studied. Another form of poloidally segmented strap has been proposed for this replacement project by G. Bosia (U. Torino), though only in a very preliminary way so far.

This R&D effort would culminate in the design, construction, and installation of a low-impedance actively cooled wave launcher in the 285/300 port by the middle of the five-year period connected to the transmitters with an appropriate advanced ELM resilient matching scheme. The successful demonstration of greatly improved power handling in H-mode plasmas with moderate-to-large outer gaps would clearly motivate replacement of the other two FW launchers on DIII-D with similar advanced designs, with the eventual goal of being able to couple all of the available transmitter power (6 MW) at 10-s pulse length to such relevant plasmas in a reliable fashion.

10.7. IN-VESSEL COILS

In addition to the set of 18 axisymmetric field shaping coils located external to the vessel, DIII-D has two nonaxisymmetric coil sets: the six C-coil which are located at the height of the vessel midplane external to the vessel and the twelve internal I-coils, located above and below the midplane. The coils are used for correction or enhancement of magnetic error fields, feedback stabilization of the Resistive Wall Mode, and more recently for the creation of a resonant magnetic perturbation for ELM stabilization. A number of different in-vessel coils have been proposed to augment this existing coil set on DIII-D. The first set of coils is a nonaxisymmetric coil set to be used to produce an improved Resonant Magnetic Perturbation for ELM stabilization. The second is an axisymmetric coil set designed to expand the flux in the divertor target region for reduced heat load to the divertor plate. Both coils are discussed below. A common design issue for all the internal coils is the availability of adequate port space to get the large number of electrical and cooling leads in and out of the vessel. A second common design issue is the availability of space under the tiles.

10.7.1. Nonaxisymmetric Coils for ELM Reduction

There are four different concepts that are under consideration for the purpose of providing an improved resonant magnetic perturbation for ELM stabilization (see Table 2-2 and Fig. 2-4). The favored concept, Option 1, is for 48 saddle coils on the centerpost arranged in 4 poloidal rows of 12 coils each. The general arrangement is shown in Fig. 10-8. The coil requirements are for 0–50 Hz with a current of 4 kA-turns and a pulse length of only 3–4 s, long enough to demonstrate the effectiveness of the technique. The low frequency requirement allows the coils to be multi-turn and the short pulse length will permit little or no active cooling. Groups of three of these coils would be connected in series inside the vessel since there are not enough accessible and available vertical ports for the 48 pairs of coil leads.

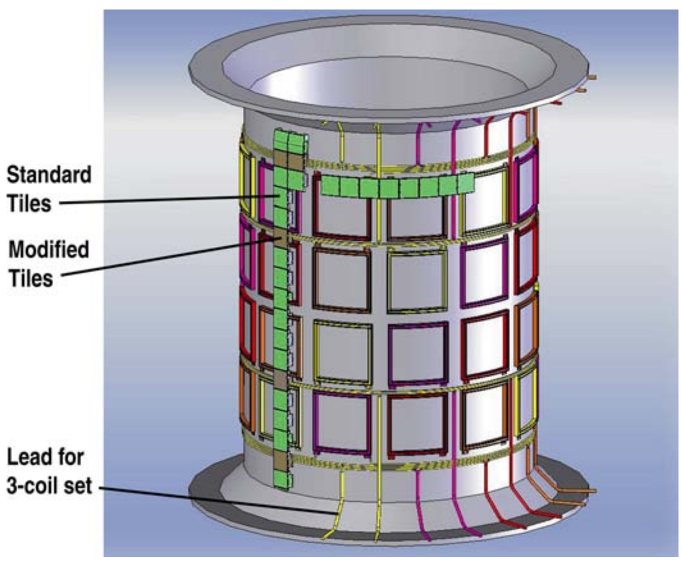

Fig. 10-8. Proposed layout of 48 inner wall coils. Four sets of three coils are shown in different colors (yellow, purple, red, orange) on each row of 12 coils. The green tiles are standard tiles and the gray tiles indicate the rows that would require modification to accommodate intercoil connections. With 16 pairs of leads, the relative phasing of the 4 sets on each horizontal row would be adjustable.

This arrangement provides for operation with n=3 or 6. There is room under the tiles for the coils, but new tiles would need to be designed to accommodate the horizontal coil-to-coil leads on the upper and lower centerpost (shown as gray tiles in Fig. 10-8). A few additional tiles would need to be designed for possible local diagnostic interferences with magnetic sensors, CO_2 mirror, Thompson viewing dump, *etc*.

The small size of the individual coils, ~0.2 m square, will permit the coils to be formed or wound external to the vessel for more efficient and higher quality installation. The present concept is to provide four to eight R2 ports for 16 pairs of leads. The conductors and insulation would be totally enclosed in a stainless steel vacuum can similar to the scheme used for the internal I-coils. Installation would require temporary relocation of divertor cooling plates.

Power supplies, external cabling, patch panels, and an interlock system will be required. Design concepts based on the I-coil will be used since it has similar requirements of a large number of small power supplies with a flexible interconnection system.

10.7.2. Axisymmetric Coils for Divertor Heat Flux Reduction

As described in Section 3.7.4, a new set of axisymmetric coils, referred to as the Divertor Flux Expansion Coils would expand the field lines in the divertor region in order to spread the heat flux over a larger divertor target area. The favored concept is for two to four horizontal, concentric, circular coils mounted under the graphite protection tiles on the top of the lower divertor plate. The coils require less than 10 kA-turns dc in each coil. Fig. 10-9 shows a four-coil set and the resulting flux expansion. It is possible that these coils could be single turn water cooled similar to the I-coils or multi-turn cooled with nitrogen gas between shots. Both concepts require one or two V-1 port for leads and routing space under the divertor tiles. The tiles on the divertor shelf would need to be redesigned to provide the space to route the coil conductor. The space required for the coils impacts present magnetic sensors and Langmuir probes on the divertor plate.

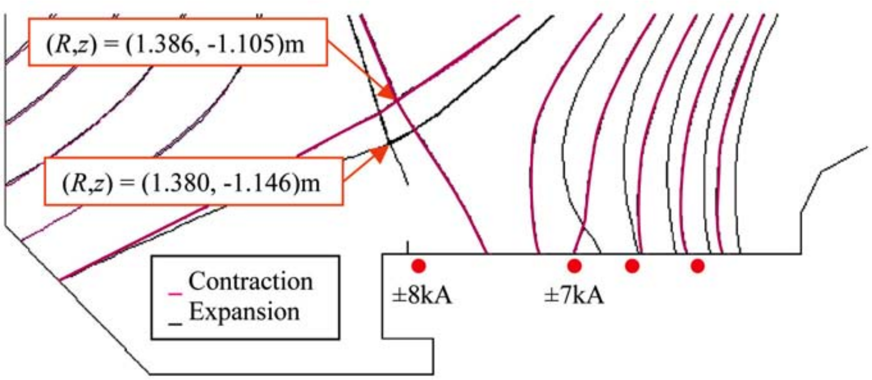

Fig. 10-9. The poloidal cross-sections of the two extremes of the flux expansion cases using a four-element divertor flux expansion coil (red circles). The flux surfaces in the SOL are 10 mm apart at the outer midplane. The ratio of the maximum flux expansion to the minimum expansion is 3.3.

Power supplies, external cabling, patch panels, water cooling, and a monitoring and interlock system will be required for this set of coils.

10.7.3. Error Field and Resistive Wall Mode Control

As outlined in Section 3.6.1, enhancements to the power and control system for the error field and RWM control are desired. The enhancements address two primary requirements: (1) provide sufficient current to enable independent operation of each of the 12 I-coils for control of RWM modes above n=1 and (2) enable the I-coils to be used simultaneously for dynamic error correction and RWM stabilization.

The present system consists of 24 high bandwidth, low current amplifiers (20 kHz, 190 A) and 4 lower bandwidth, higher current switching power amplifiers or SPAs (4 kHz, 5 kA). The two types of amplifiers cannot be operated simultaneously on the same coil set. As such, there is insufficient capability to meet either requirement.

The first requirement can be met by doubling the number of high bandwidth amplifiers from the present value of 24 to 48 or replacing the existing set with an upgraded higher current amplifier. This would double the high bandwidth current capability and enable currents up to 750 A into each of 12 coils (4 amplifiers/coil) or 1500 A into helical pairs. In addition to purchasing additional amplifiers, an additional power feed line, a new 480/208 transformer, and circuit breakers would be required. The existing amplifier patch panel has sufficient room to handle 24 additional supplies.

The second requirement can be met by building a cross-over network that enables the higher current SPAs to be operated in parallel with the higher bandwidth amplifiers. Each 5 kA SPA can be subdivided into three 1.7 kA units, so that the combination of the two amplifier types with the cross-over network would provide 1.7 kA for slowly varying error correction and 750 A for high bandwidth RWM feedback control for each of the 12 I-coils. A raised floor will need to built above the existing power supply area to provide the room for the capacitors in the network. As in the first case described above, the required high bandwidth current can be obtained either by doubling the number of amplifiers or purchasing higher power units.

10.8. FUELING AND DISRUPTION MITIGATION

10.8.1. Fueling

The pellet fueling system on DIII-D is a 3 independent barrel repeating pneumatic injector. Two of the barrels produce slow 1.8 mm pellets at up to 7 Hz for transport through curved guide tubes to reach the plasma from the inner wall or top of the vessel. The third barrel is a higher speed capable gun that is primarily useful for outside midplane injection.

Experiments carried out in the past five years have found that inner wall pellet fueling is much more efficient than conventional outside midplane injection and produces smaller ELMs. The present fueling rate of inner wall pellets on DIII-D, equivalent to 45 Torr-l/s of gas, has been found to be insufficient to produce a significant change in H-mode pedestal density and pressure conditions despite the deep efficient fueling. The large ELMs that are triggered by outer midplane injected pellets have motivated the development of an additional injector, the pellet dropper to produce periodic, small amplitude ELMs. This system, which is to be fully operational in 2008 will only provide ~30 Torr-l/s of very shallow particle fueling and therefore is not useful for fueling studies.

Improved Inner Wall Fueling. A higher fueling rate from the inner wall is desirable for pedestal modification physics studies and for simulating operational scenarios for ITER. Large pellet sizes is one method to increase the fueling rate, but it would be likely to be too perturbative. A more desirable way to increase the inner wall fueling rate is to increase the rep rate of the existing size pellets. This can be achieved by modification of the fast gun to a slow gun and improving the repetition rate of all the guns. The technology to do this for steady state is currently under development for ITER. We propose to use this development to update the DIII-D pellet injector to higher repetition rate capability and provide a third slow gun for inner wall fueling. This would double the inner wall fueling rate to approximately 80 Torr-l/s. Steady state capability can be achieved on the DII-D pellet injector with the addition of a screw extruder. A screw extruder prototype is presently under development for ITER and this could be used on DIII-D for one of the guns to provide steady-state 10 Hz operation.

10.8.2. Disruption Mitigation

For disruption mitigation studies, DIII-D has conducted experiments using two different systems. The first uses the DIII-D pellet injector described above to produce killer impurity pellets; to date, both argon and neon have been used. The second system is the massive gas injection system and this is presently configured as a six-valve injector, each of which is an independent fast acting valve capable of injecting large amounts of gas with a short rise time. The valve design has been optimized to minimize the rise time and the system has been used with D_2, He, Ar, and Ne.

While the killer pellets have been successful at significantly reducing disruptive heat loads and halo current forces, they generated runaway electrons. The massive gas injection valve has had good success with both heat loads and halo currents, but to-date the technique has not been successful at getting enough gas into the plasma to assure collisional suppressions of the runaway avalanche.

In the next five years, the DIII-D program will consider four different techniques to achieve sufficiently high core density to prevent runaway electron avalanching: inverse jet, customized solid pellets, large cryogenic pellets, and liquid jets.

10.8.2.1. Inverse Jet. In this technique, a long cylindrical tube prefilled with gas at high pressure is sealed at the plasma facing end with a rupture disk and placed close to the plasma edge. A fast-acting valve is attached to the other end with a higher pressure gas reservoir behind it so that when the valve is activated, the rupture disk breaks and the gas rapidly reaches the plasma. This technique attempts to avoid the long time delay associated with gas moving through a long guide tube that is inherent in the present massive gas injection system. By reducing the time delay and increasing the fueling rate, the goal is to fuel the plasma more effectively during the thermal collapse phase. A laser could also be used to more rapidly open the rupture disk, but it is likely to increase the complexity of the system. The possibility of metal shards from the rupture disk entering the vessel must be examined.

10.8.2.2. Customized Solid Pellets. This technique refers to using prefabricated solid pellets at room temperature consisting of various layers of materials chosen to optimize core impurity deposition and the fueling deposition profile. These pellets will be fabricated at the GA ICF facility. Coated pellet injection will be done using the GA lithium pellet injector, which will be operational in early CY08. The technique is simple and requires the installation of no new hardware.

10.8.2.3. Large Cryogenic Pellets. This technique utilizes large cryogenic pellets with a diameter of up to 1 cm, and containing the necessary 10^{23} particles. ORNL is developing a “pipe gun” capable of firing these large cryogenic pellets. To avoid penetration through the plasma and damage to the opposite wall, the pellet may be broken up prior to vessel entry.

10.8.2.4. Liquid Jet. This technique utilizes a high pressure jet of liquid helium. As in the inverse jet, the helium is in a tube that is terminated on the plasma facing side by a rupture disk. Various techniques, such as laser triggering, will be considered to rupture the disk and permit the jet to flow. The high speeds required for the jet also require the development of a supersonic injection nozzle. The propagation through the plasma has been studied theoretically and is expected to provide good core penetration without relying on MHD activity for penetration. However, propagation through the vacuum region from nozzle to plasma edge is as yet untested. While the technique avoids the potential damage to the opposite wall associated with the large cryogenic pellets, it is technically the most complex and will require considerable development. Issues of nozzle design, burst disk fragmentation, impurity contamination of the LHe, and vacuum compatibility are among some of the issues to be addressed in this effort.

10.9. DIVERTOR AND FIRST WALL MODIFICATIONS

The present divertor and first wall structures are engineered with the purpose of providing particle control and pumping of plasma particles, dissipation of the high heat loads exhausted from the plasma, and protecting the vessel wall during injection of high power auxiliary heating. To this end, there are three in-vessel cryopumps located under protective divertor baffles, and the divertor structures and most other plasma facing vessel surfaces are covered with graphite armor tile. The major goals of the hardware upgrades proposed in this section are to address the following research needs: improved pumping and thermal performance of the divertor, increased peak injected power as well as total injected energy, reduced erosion of plasma facing components, and enhanced spreading of the energy input to reduce both localized and average heat fluxes. The hardware changes proposed to meet these needs include: installation of a new lower divertor structure (Section 10.9.1); an improved tile design (Section 10.9.2.1), enhanced water flow to divertor structures (Section 10.9.2.2), upgrading other in-vessel components, such as diagnostic windows (Section 10.9.2.3), and operation at elevated wall temperatures (Section 10.9.3).

10.9.1. Lower Divertor Upgrade

The existing lower divertor region provides an inner graphite target region, an in-vessel cryopump for pumping the outer leg of the divertor, and an elevated shelf and baffle plate providing both a pumping plenum for the outer pump and an alternative target region for the divertor strikepoints. It is proposed to add a new lower inner divertor structure that would provide a physical separation between the inner and outer lower divertor target regions. As discussed in Section 3.7.4, this physical barrier would improve particle control and improve the performance of the radiative divertor, and the shape of the structure would provide increased surface area for lowering the peak heat fluxes to the divertor region. Similar to the other three existing divertor structures, the proposed structure will be water cooled and protected by graphite tiles to enable it to handle the full heat load of the divertor strikepoints. However, this lower inner structure is different than the upper inner baffle in a number of ways: the upper curved dome shape is replaced by a flat, horizontal surface and a conical section, and since its purpose is only to provide a physical barrier between the two strikepoint regions, there is no cryopump under the structure and thus, no pumping opening.

10.9.2. Vessel Thermal Upgrade

The goal for the next five years is to be able to be operate up to 300 MJ/shot for a double null divertor and 200 MJ/shot for both upper and lower single null divertors with a 15 minute cooldown between shots. The requirement for the high performance double null discharge represents a five-fold increase in the injected energy we have obtained to date. The injected power will also increase from 20 to 30 MW. The engineering requirements assume an energy deposition pattern based on 30% radiation of the input power, with the remaining 70% conducted to divertors. The conducted power is assumed to be distributed 60% to the outer strike point and 40% to the inner strike point for single null plasmas and 90% to the outer strike points and 10% to the inner strike points for double null plasmas. To provide maximum flexibility for the research program, these high heat fluxes at the divertor strikepoints must be able to be positioned over a wide spatial range. This capability requires modifications to the plasma facing tiles, the upper divertor cooling, and specific diagnostics and other specialized in-vessel components.

10.9.2.1. Vessel Plasma Facing Tiles. The DIII-D first wall and divertor surfaces are fully covered with plasma facing tiles made from Union Carbide TS1792 graphite. These are inertially cooled during the 10 s discharge and cool down during the 15 minutes between shots by water flow in the tile support structure. Union Carbide TS1792 is a low cost graphite with traditionally acceptable, but not high, thermal conductivity. The material has performed well for shot energies up to 60 MJ with a 10 minute cooldown between shots. However, thermal analysis shows that at higher energy loadings, the existing Union Carbide TS1792 graphite tiles have limitations based on excessive peak surface temperatures and thermal stress limits. To avoid these limits, improvements will be required in plasma facing tile thermal conductivity, particularly in the divertors, floor and ceiling tiles. Peaked heat loads in these areas require improved thermal conductivities in all three axes of the tiles. It is expected that carbon-carbon composite materials will be required in these high heat flux areas. The higher input energy and peaked heating profile present in the divertor strike point areas produce thermal stress problems due to high thermal gradients within a tile as well as increased surface temperatures above the acceptable range to avoid enhanced erosion. Carbon-carbon composite materials are structurally more robust and their higher thermal conductivity results in reduced thermal gradients, lower surface temperature due to better utilization of the full thermal mass of the tile, and thus, higher acceptable energy per shot for the same tile geometry.

The upgrade plan is to replace TS1792 tiles with carbon-carbon tiles in specific areas where the heat flux is calculated to exceed the rating of the existing tiles. This is predicted to occur in the areas of the lower divertor shelf tiles, the tiles on upper outer baffle conical section and the upper inner baffle tiles, and the associated floor and ceiling tile rows. The majority of the approximately 3000 tiles on the inner and outer walls will still be acceptable and will remain TS1792. It is planned to replace the flat tiles on the center-post with existing TS1792 material contoured to the radius of the wall with smaller tile-to-tile spacing and improved alignment. These will reduce edge heating and thus reduce both tile erosion and toroidal nonaxisymmetries that make it difficult to interpret diagnostic signals.

10.9.2.2. Vessel Water Flow. The basic water cooling system for the DIII-D vessel is adequate to remove 300 MJ between shots. The upper divertor cooling water flow is significantly lower than the flow in the lower divertor. In order to prevent water boiling in the upper divertor, energy applied to each toroidal row of tiles must be limited to less than 36 MJ/row with present water flow rates, which is insufficient for some of the proposed high injected power scenarios. The water flow rate in the upper outer divertor can be raised with modest changes to the flow configuration in the baffle plate. However, a new inner plate with higher flow capacity is required. Increased water flow will require the use of an existing vessel port.

Additional areas that may require enhancement due to relatively low water flow include the NB drift ducts.

10.9.2.3. Other In-Vessel Components. Other than heat directly conducted to tile surfaces, all other in-vessel components are exposed to heat primarily via plasma radiation. Based on existing measurements, peak radiation levels of 50 W/cm2 are expected to occur in plasmas with highly radiating mantles. While this level is relatively benign for even our existing graphite armor tiles, the thermal impact on diagnostic systems may be significant. Some of the large diagnostic windows that are located close to the plasma surface for enhanced viewing angle will need to be evaluated and may require modification to handle these high heat fluxes.

Other components that will require upgrades due to the high heat fluxes are the ECH launcher assemblies (Section 10.4) and the 285/300 deg Fast Wave antenna (Section 10.6).

10.9.3. Hot Walls

DIII-D is currently operated with 24°C cooling water flowing through the vessel walls and divertor structure. In order to reduce the co-deposition of deuterium or tritium in the redeposited carbon layers, we are proposing to operate DIII-D with elevated wall temperature. Two concepts are being studied:

1. Installation of a closed loop capability to the vessel cooling system to increase water inlet temperature to about 100°C. This provides a relatively uniform temperature to all vessel surfaces and a consistent temperature at the beginning of each discharge due to the large heat removal capability of the water flow. Boiling is avoided at the operating pressure of our present cooling water system.
2. Installation of a closed loop capability to the vessel air cooling system to provide recirculating air with inlet temperature to about 200°C. While this permits a higher temperature, the uniformity of the temperature around the vessel poloidally is poorer due to the poor heat removal capability of an air system. The use of the hot air system may require more than 15 minutes between shots to redevelop acceptable starting temperatures or conversely, the injected power may be limited if temperature uniformity and a reasonable time between shots is desired. The effect of a hot vessel on diagnostic alignment needs to be investigated for each diagnostic system.

10.10. 10 s PULSE UPGRADE

Extending the pulse length of the DIII-D tokamak from 5 to 10 s for a 1.6 MA high performance discharge at full toroidal field requires a number of modifications to the tokamak systems. In this section, modifications to the coil systems and power supplies will be discussed. The required upgrades include an expanded thermal monitoring and interlock system for the poloidal coils, a new dc power supply and additional current regulators for the field shaping system, and upgraded components in the regulators. Radiation dose estimates and the impact on the experimental program are discussed. Successful heat flux reduction techniques, divertor armor tile modifications, and upgrades to the vessel/divertor water cooling system should permit full power, long pulse discharges and were discussed in Section 10.9. These issues are briefly summarized at the end. Modifications to the auxiliary heating systems to enable long pulse at high power were also described in earlier sections (Section 10.4 ECCD, Section 10.5 neutral beams, Section 10.6 FWCD).

10.10.1. Present Status

The present system is capable of operating high performance, double-null discharges with I_p = 1.6 MA, B_T = 2.1 T for 5 s. For 10 s, these levels are limited to slightly less than 1.2 MA at a reduced toroidal field of 1.7 T. The target DND shape is optimized for the high normalized beta, low internal inductance, and good pumping of the outer strikepoints (Fig. 10-10). This shape requires high currents in the F8A and F8B divertor coils and sets the operating limits of the tokamak for this shape. At short pulse length, the plasma current is limited by the peak stress-limited current in the F8 coils and at longer pulses, the thermal capacity of these coil is the limiting feature (Fig. 10-11).

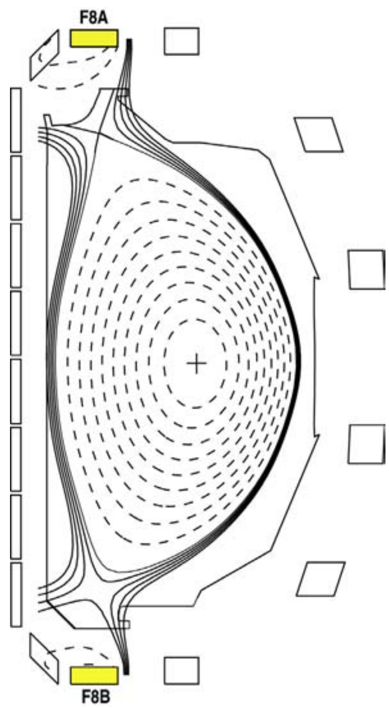

Fig. 10-10. Upwardly biased double null divertor to be used as target shape for high performance, long pulse discharges. Either the F8A or F8B coil are the limiting coils depending on whether the bias is up or down.

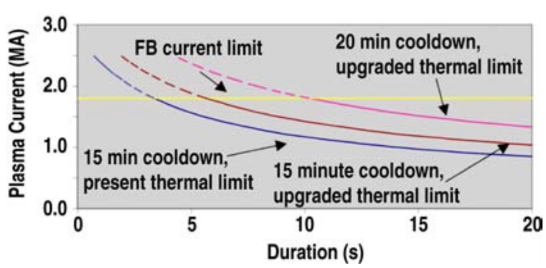

Fig. 10-11. Plasma current flattop duration is limited by the thermal capacity and cooling rate of the F8 coil. Addition of a negative bias current to the inner coils will allow operation of the target shape up to 1.8 MA.

The toroidal field coil is capable of 10 s operation at full field and until recently was limited by the interconnecting bus work between the TF-coil bundles that run along the midplane of the machine on the outside of the coil. Recent upgrades to this bus work have removed that limitation. The pulse length of the toroidal field operation is presently limited to 6.5 s at full field by the SCRs in the TF supply. A modification to the mode of operation of the supply, discussed in Section 10.10.2, should permit operation to 10 s.

The Ohmic heating coil is capable of delivering 7.5 V-s of flux. About 4.0 V-s are required to initiate the plasma and ramp the current to 1.6 MA. With auxiliary heating power in excess of 20 MW, it is possible to heat the plasma sufficiently to lower the one turn voltage to 0.25 V. Thus, the remaining 3.5 V-s is more than adequate to sustain the discharge for 10 s. The use of rf and neutral beam current drive (along with bootstrap current) will further reduce the flux requirement. Thus no upgrade of the Ohmic heating coil is required to support 10 s pulse upgrade. For long pulses, the coil power supplies are powered from the large motor generator MG2, from which 2.25 GJ can be extracted. About 1.6 GJ are required for the coil power supplies for a 10 s plasma. Thus the MG has sufficient extractable energy. Cabling from the MG2 to the power supplies is sufficient and does not need to be upgraded.

The auxiliary heating systems, neutral beams, FW and ECH, are powered from a 138/12.47 kV transformer, which is rated at 9.4 MVA continuous duty and 84 MVA for 1 s. The additional long-pulse ECH sources and the extension of the neutral beam system to 10 s operation at full power require the upgrade of this prime power system.

In 2006, the two old cooling towers were replaced with new, fully stainless steel tower with sufficient heat removal capacity for 10 s operation with the currently planned auxiliary heating power.

10.10.2. Upgrades to 10 s Pulse

10.10.2.1. Coil Systems. The plasma current duration for the target discharge is limited by the thermal capacity and cooling rate of the F8 divertor coils followed closely by the thermal capability of the inner shaping coil power supplies. However, the present system is operated conservatively and we use only half the ultimate thermal capacity of the coils. Upgrades to the instrumentation monitoring the flow rates and coils temperatures, extension of the cooldown time between discharges to 20 minutes, and additional power supplies will permit the use of the full thermal capacity of the coils and permit full 10 s operation for our target 1.6 MA discharge. In order to operate with the shorter cooling time of 15 minutes and reduce the need for additional supplies, the current in the inner coils and, in particular, the F8 coils must be reduced. This can be accomplished by biasing the entire set of inner and divertor shaping coil currents negatively. If the distribution of the bias current is correct, only the nominal flux value at the plasma boundary changes with no change in the discharge shape. Our present mode of operation constrains the sum of the inner coil and divertor coil currents to be equal to zero; however, removing this constraint will permit a bias current and allow the coil currents to be reduced sufficiently to operate with a 15 minute cooldown time. Allowing the sum of currents to be nonzero, increases the total current requirement so additional supplies (a dc supply and four regulators) will be needed for this biasing.

10.10.2.2. Power Systems. As presently operated, the TF pulse length is limited by the SCRs in the TF supply to 6.5 s operation at full field. To permit full field operation for the extended pulse, the supply will have to be operated differently. In the new operation mode, all four modules of the supply would be used during the ramp up. When flattop is reached, two modules will be turned off. After 5 s in flattop, the two modules that have been on will be turned off, while the two other modules will be turned on. This almost doubles the permissible pulse duration of the TF supply. Freewheeling diodes in each supply are used to conduct the current while it is in the off state. A recent study has shown that the present freewheeling diodes have sufficient current capability to permit this mode of operation. Preliminary tests of this mode of operation have shown that improvements to the current regulation system of the TF supply are necessary in order to prevent current transients when the supplies are turned on and off.

The F-coil current regulators also need to be upgraded. The varistors and snubber components for the SCRs in the low voltage regulators need to be replaced, whereas all the other components can support 10 s operation. In the higher voltage regulators, the grid resistor and snubbers need to be replaced. Four long pulse regulators, based on the newer IGBT technology rather than the SCRs used in our present units, will be built. These will allow the removal of the current constraint described above and add considerable flexibility to our shaping ability.

An upgrade to the capacity of our 138 kV transformer is required to provide adequate power for the long pulse and higher power proposed. A new transformer capable of meeting all the future prime power needs of the auxiliary heating systems has been obtained through a collaboration with ASIIP, China. It will be fully installed and operational early in the five year period covered by this proposal. The upgrade is discussed in more detail in Section 10.11.

10.10.2.3. Radiation Dose For Long Pulse, High Power Discharges. Presently, the radiation dose at the site boundary for a typical week of operation is 1.0 millirem. If the balance in the experimental program between high performance discharges producing high radiation dose and lower dose discharges remains the same, an extension of the typical heating pulse length by a factor of two would increase the typical weekly dose to 2.0 mrem. Thus the facility can be operated for roughly 20 weeks without exceeding our DIII-D procedures of 40 mrem/year. Extending our procedural limit to the California legal limit of 100 mrem/year would significantly increase the ability of the research program to focus on longer pulse discharges. Based on these average values, the radiation shielding of the facility should be adequate to carry out the proposed research plan.

More detailed calculations of expected dose rates were performed for the three types of discharges in our advanced scenario research program (Table 10-6). For the highest performance, steady state advanced tokamak discharges injecting 30 MW for 10 s, the number of discharges is limited to approximately 300 per year or approximately 10% of our research program, if the procedural limit is raised to the California limit. The numbers are somewhat higher for the two slightly lower performance hybrid and advanced inductive discharges. Although this permits the research program to proceed, these high dose rates will require prudent operation of these high performance, long pulse discharges.

10.10.2.4. Vessel Armor. The existing vessel armor is sufficient for long pulse operation is total injected power remains unchanged at the present level of 60 MJ. Proposed upgrades to tiles in the divertor and centerpost region described in Section 10.9.2 (replacement of tile material from ATJ graphite to CFC, tile contouring on the centerpost, reduced tile-to-tile gaps) will reduce edge temperatures and increase thermal diffusion into the tile during the 10 s pulse and should reduce the tile temperatures and thermal stresses to acceptable levels. Integration of heat flux reduction techniques already developed on DIII-D (enhanced radiation from impurity seeding in the divertor) coupled with the new coil systems for spreading the heat flux (Section 10.7) should reduce the heat load and provide increasing engineering margin for the tiles.

Table 10-6
Number of High Performance, Long Pulse Discharges Based on Radiation Dose Limits

Discharge Type	Stored Energy	β_N	Number of 10 s Shots/yr (40 mrem/yr limit)	Number of 10 s Shots/yr (100 mrem/yr limit
Steady State AT	3.4 MJ	5	127	319
Advanced Inductive	2.9 MJ	3.2	162	405
Hybrid	2.5 MJ	2.8	202	506

10.10.2.5. DIII-D Water Cooling System. While the heat removal capability of the overall DIII-D plant and the DIII-D vessel is sufficient to enable our proposed high power, 10 s discharges, specific components require additional cooling water (neutral beam drift ducts, EC launcher mirrors, NB internal components, upper divertor structure). Additional water flow is also required for the new EC power supplies and gyrotrons. Additional pumping capability and heat exchangers are planned to be installed to meet the increased cooling water flow.

10.11. PRIME POWER AND COIL POWER SUPPLIES

The aggressive experimental requirements outlined in previous sections will require that there be a significant increase in the site primary power capacity. The majority of coil supplies are fed from the motor generator, which has sufficient capacity to meet the expected demands for the foreseeable future. However, the transformer that feeds the auxiliary heating systems, the 138 kV/12.47 kV transformer, has insufficient capacity to provide the prime power needs of the proposed program. In addition, the current capacity of the 4160 V/480 V substation has reached its limit and is unable to provide the power required for additional water pumps for the new EC gyrotrons, HV power supply, and neutral beam component cooling. Two new transformers, 138 kV/12.47 kV and 12.47 kV/4160 V have been acquired to meet the power needs of the proposed program. This section describes their capabilities, a proposed reconfiguration of the prime power feeds for reduced cost and better power utilization, and the proposed installation schedule. In addition, refurbishment and preventative maintenance of the existing coil power systems is described. A summary of all power supply needs identified in different parts of Section 10 is included at the end of this section.

10.11.1. 138 kV Transformer

A new higher capacity 138 kV/12.47 kV transformer was obtained through the on-going collaboration with ASIPP in China. This new transformer has a pulsed rating of 110 MVA for 10 s with 15 minutes between pulses. A comparison of our existing 138 kV transformer and the new unit is shown in Figure 10-12. The two curved lines (Oil Circuit Breaker Trips) represent the transformer limits and the green and blue curves show the energy demand required for two different heating scenarios. The lower blue curve represents the approximate energy demands of the heating system that we will have in at the end of FY08: 17.5 MW neutral beams (3.8–5.0 s), 6 MW of EC (5 s), and 6 MW of Fast Wave power (2 MW/2 s and 4 MW/5 s). The present transformer is just short of being able to match the capability of the present heating system. The green curve shows the energy demand for the proposed heating upgrades proposed in this plan: 20 MW neutral beams (10 s), 12 MW EC (10 s) and 6 MW of Fast Wave (10 s).

The new transformer is well-matched to meet the energy needs of our proposed system. The 138 kV transformer is presently on the DIII-D site in a temporary storage area. We propose to begin the installation of this unit at the end of FY08 and complete the installation in early FY09.

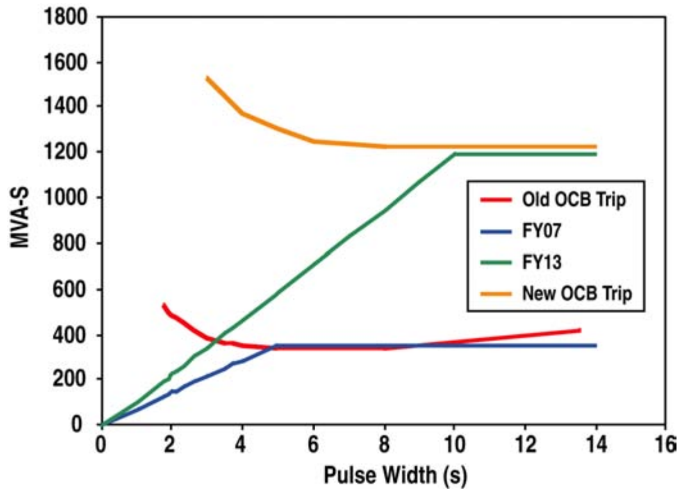

Fig. 10-12. Capabilities of present (red) and upgraded (orange) 138 kV/12.47 kV transformer. Energy demands of existing heating system (blue) and expanded heating system (green) are shown.

10.11.2. 4160 V and 480 V Substations

As presently configured, the 69 kV line to the DIII-D site powers the motor generator and cooling system for the facility. A 69 kV/4160 V transformer feeds the motor generator and a series of 4160 V/480 V transformers feed the water pumps. Since the 4160 V/480 V substation was installed in 1979, there has been substantial growth in the cooling needs for the facility. The anticipated needs of the cooling system for the upgraded auxiliary heating systems cannot be met from the existing 4160 substation. A study was performed to determine possible options for providing the required power for the cooling and other infrastructure as additional heating systems are brought on line. It was determined that if the new installed 138 kV/12.47 transformer powered the motor generator (instead of the 69 kV line), it would free up power from the 4160 V/480 V substation to meet the needs of our future cooling system. As such, the new transformer was specified to provide an additional 12.7 MVA for 90 s every 15 minutes in sequential, not simultaneous operation with the specified 110 MVA pulsed capability. A second transformer, also obtained through the ASIPP collaboration, provides the necessary 12.47 kV/4160 V power to run the motor generator. By powering the motor generator from the 138 kV line, not only will there be sufficient 480 V power to support projected power needs, but there will also be an opportunity for significant energy cost savings resulting from the more favorable billing rate associated with the 138 kV feed when compared to the 69 kV line for the existing 4160 substation.

A pad and appropriate oil containment has already been installed, preliminary design of the associated switchgear has been completed, and the 12.47 kV/4160 transformer has been placed on the pad. The current plan is to complete the switchgear procurement and installation sometime in FY10, with the first MG operation the following year. To take advantage of the released capacity of the existing substation some additional trenching and cabling will be necessary.

10.11.3. Infrastructure Refurbishments

With much of the power systems equipment approaching or exceeding 25 years of service, a proactive refurbishment and inspection program has been initiated. The program will address two primary goals: modernization of power supply control systems and inspections/refurbishment of the many pulsed power transformers used in the auxiliary heating systems. The nine major dc supplies for the various coil systems on DIII-D need modernization of their control systems and this will be pursued on an on-going basis throughout the next five years. A program of detailed inspection of the transformers has already begun. In FY07, weakened insulation in one transformer was identified and the unit was taken out of service and refurbished prior to failing. We have since performed detailed inspections of 13 HV transformers and scheduled two previously failed units for rebuilding. These inspections and refurbishments will continue for the next several years.

10.11.4. Summary Of Coil Power System Upgrades

Table 10-7 summarizes the new or upgraded coil power supplies requirements outlined in this proposal. The section discusses each of the systems is listed in the table.

Table 10-7
Summary of Coil Power System Upgrades

System	New Hardware	Section Reference
ELM reduction coils	16 1 kA dc supply, 5 s	10.7.1
Divertor flux expansion coils	2–4 10 kA dc supplies, 10 s	10.7.2
RWM/error field correction	24 audio amplifiers, cross-over network	10.7.3
F-coils	4 current regulators – 600 V, 6 kA 1 dc supply – 600 V, 20 kA, 10 s	10.10.2.1

10.12. COMPUTER SYSTEMS, DATA ACQUISITION (INFRASTRUCTURE) AND CONTROL (CODAC)

Computer systems, data acquisition, and control systems and instrumentation are integral parts of all aspects of the DIII-D experiment. The plan described in this section is primarily focused on the real-time computing, data, and control system infrastructure but also includes the general computing infrastructure and user support services. Discussion specific to data analysis is the subject of Chapter 9.

10.12.1. Present System Status

During the past five years there have been substantial changes to the DIII-D computing and data acquisition environment. Much of this change is being driven by increases in the amount of data generated and the ever-improving technological capability developed for commercial applications. The largest shot size for a DIII-D discharge has grown from 611 Megabytes in 2002 to 4.15 Gigabytes in 2007 (Fig. 10-13). This increase in shot size has, in part, been driven by increases in the number of diagnostic systems and complexity of the data acquired. In addition, the combination of improved control requirements and more sophisticated analysis between shots and off-line has driven the need for more powerful computers and more data storage.

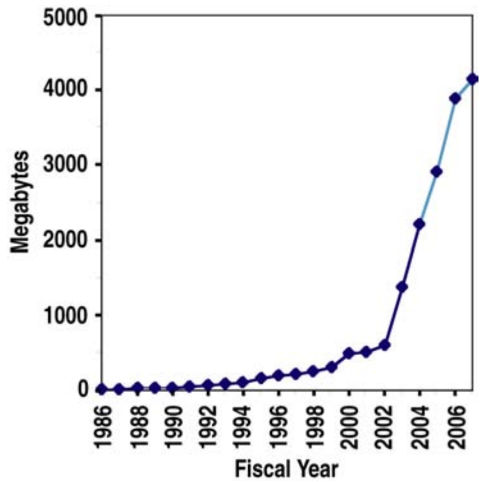

Fig. 10-13. Raw data size over the last five years has grown dramatically (9X). Largest shot per fiscal year.

It is estimated that in five years, raw data shot size will potentially exceed 8.5 Gigabytes and the amount collected per year will exceed 11 Terabytes.

Significant improvements in the past five years have included:

- Expansion of user magnetic disk storage to over 20 Terabytes, processed data storage to 2 Terabytes, and magnetic storage for raw data to over 30 Terabytes.
- The maintenance of all DIII-D raw shot data available on magnetic disk to facilitate access for large shot surveys.
- Upgrades of the tokamak and neutral beam control systems to Linux, the introduction of other Linux data acquisition computers and the replacement of proprietary Unix systems with Linux for general data examination and analysis in the LSF cluster.
- Began transition away from CAMAC-based systems: integration of PCI and cPCI digitizers into the data acquisition infrastructure and PXI-based systems for control and data acquisition.
- Upgrading of the network infrastructure to support Gigabit Ethernet (including a dual Gigabit link between the DIII-D facility and the main computing center) along with modernization of the network router and security firewall.
- Major upgrades of the Plasma Control System using Linux with Myrinet communication links.

Both the control and data acquisition systems have a legacy of using CAMAC based systems.

10.12.2. Major Initiatives for the Next Five Years

A major initiative of real-time computing involves changes to control, data acquisition, and instrumentation systems away from CAMAC. For the past 20 to 30 years, CAMAC hardware has been widely used in the Fusion and High Energy Physics research communities. In recent years, CAMAC systems are no longer supported for most of our applications and as our systems age, the hardware cannot be replaced or repaired. Our first major step away from CAMAC occurred with the purchasing of cPCI and PCI waveform digitizers for the DIII-D plasma control system real time computers. These are installed in CPUs interconnected by Myrinet or Ethernet. An alternative approach has been pursued by both the ECH and FW systems which have installed PXI-based systems for both control and data acquisition. While the FW system is relatively new, the ECH system has been operating reliably for a number of years. We propose to continue this transition away from CAMAC by replacing old systems proactively starting with the oldest and most troublesome systems. As described in Section 10.5.1.3, the neutral beam control and data acquisition system will be the first major system replaced. In addition, all new diagnostic and control systems will use the newer technologies. The actual hardware chosen will depend on an evaluation of the performance history of the new systems that have been installed.

A second major initiative is to investigate the use of Storage Area Networks (SAN) for raw and processed data and user areas. A SAN is a network of storage devices. The use of a SAN can provide more reliability and availability of storage resources and should allow us to maintain our goal of keeping all data available on magnetic disk media for rapid access. It can also provide the capability to expand storage resources transparently to the users as well as enhancing backup capabilities.

The third initiative is the continued expansion of Linux-based computers for control and data analysis. In the control area, the main tokamak control and neutral beam control systems are currently being upgraded to their second generation of Linux/Intel and will be operational for FY2008. The hardware/software combination currently in use is expected to continue through FY2013. Enhancements such as upgrades to faster processors will be made as required, but the overall structure of these systems is expected to remain the same. Some software changes will need to be made as these systems transition to the use of non-CAMAC instrumentation.

For data analysis, the need for additional computing power will continue to increase to analyze data in ever-greater detail, analyze more data, and more sophisticated graphics. A shift was made away from the proprietary Unix operating systems and computing platforms of the past, to the very powerful but less expensive platforms running the Linux operating system. Processors have continued to be upgraded for more computing power with the most recent being a dual quad core computer. Only one HP computer remains and there is still a need to port codes from it to the Linux platform. The Load Sharing Facility (LSF) cluster of computers has proved very useful for data analysis and its use is expected to continue.

The handling of greater quantities of data will also lead to more demands on the networking infrastructure. It is planned to implement 10 Gigabit Ethernet between the DIII-D facility and the main computing center. Increased use of Gigabit Ethernet is expected both for the main computers and file servers used for analysis as well as within the control and data acquisition environment of the experiment. To fully benefit from these improvements will require that upgrades to the network router and firewall be made as well.

Other noteworthy initiatives include:

- The increased usage and more routine use of the web as a means of interacting with control, data acquisition, and analysis systems and to further divorce the user from the details of computing resources.
- Continued security improvements will continue to be of paramount importance to protect DIII-D computing resources.

11. THE COLLABORATIVE NATIONAL PROGRAM

The DIII-D National Fusion Program is a highly collaborative multi-institutional research endeavor with 92 institutional participants worldwide. In large part, the DIII-D research program derives its strength from the diversity and capabilities of its national and international collaborating institutions and associated individuals. The majority of the scientific staff (full time equivalents) are from collaborating institutions; scientific collaborators have significant roles at all levels of the program. Consequently, the DIII-D Program has and will continue to invest significant resources to grow and maintain supportive and effective collaborations.

The DIII-D National Fusion Program maintains close linkage to key elements of the broader U.S. and international fusion science communities as part of its mission to optimize the tokamak approach to fusion energy. The DIII-D program actively participates in the ITER project on many levels and the research plans address issues critical to the success of ITER. The U.S. Burning Plasma Organization (USBPO) coordinates research in support of ITER and potential next-step experimentss; DIII-D scientists serve in leadership positions within the USBPO. The DIII-D Program is strongly coupled to the U.S. Theory Program and to important new computer simulation initiatives such as the Fusion Simulation Project. The DIII-D Program relies upon and contributes to development of enabling technologies such as ECH and rf systems. In addition to institutional topical science collaborations, the DIII-D Program also coordinates its research with other major U.S. and international fusion facilities, such as NSTX, C-Mod, JET, ASDEX-U, JT-60U, and others.

University participation is critically important to the U.S. fusion energy sciences program and to DIII-D. Our university partners bring a unique perspective to fusion research that greatly enriches the research program at major fusion facilities. University programs can maintain a sharp focus on key scientific questions, are able to invest in riskier, though more seminal, concepts, and provide excellent preparation for next generation fusion scientists who will help the U.S. realize the full benefit of our participation in ITER. The DIII-D Five Year Program Plan provides many opportunities for expanded University participation.

11.1. HISTORY AND SCOPE OF THE DIII-D FUSION PROGRAM

The DIII-D National Program evolved from the Doublet III device, which was constructed and initially operated by General Atomics in 1978. During that first year of Doublet III operation, a major collaboration was established with the Japan Atomic Energy Research Institute (JAERI). JAERI invested $84M (FY00 money) in the DIII-D facility and was provided half the run time in the period 1978–1984. This early major collaboration set the Program on the course that has led to the present National Team.

When Doublet III was converted into the DIII-D tokamak in 1986, an expansion of collaborations was sought as a goal of the new DIII-D Program. With the Department of Energy (DOE), GA developed major collaborations with Lawrence Livermore National Laboratory (LLNL), Oak Ridge National Laboratory (ORNL), Princeton Plasma Physics Laboratory (PPPL), Sandia National Laboratory-Albuquerque (SNLA), and the Universities of California at Los Angeles (UCLA) and San Diego (UCSD). The number of collaborations has grown significantly over time; Fig. 11-1 shows their present

US Labs

ANL (Argonne, IL)
LANL (Los Alamos, NM)
LBNL (Berkeley, CA)
LLNL (Livermore, CA)
ORNL (Oak Ridge, TN)
PPPL (Princeton, NJ)
SNL (Sandia, NM)

Industries

ALITRON (CA)
Calabasas Creek (CA)
CompX (Del Mar, CA)
CPI (Palo Alto, CA)
Digital Finetec (Ventura, CA)
DRS (Dallas, TX)
DTI (Bedford, MA)
FAR-TECH, Inc. (San Diego, CA)
GA (San Diego, CA)
Lodestar (Boulder, CO)
SAIC (La Jolla, CA)
Spinner (Germany)
Tech-X (Boulder, CO)
Thermacore (Lancaster, PA)
TSI Research (Solana Beach, CA)

US Universities

Auburn (Auburn, Alabama)
Colorado School of Mines (Golden, CO)
Columbia (New York, NY)
Georgia Tech (Atlanta, GA)
Hampton (Hampton, VA)
Lehigh (Bethlehem, PA)
Maryland (College Park, MD)
Mesa College (San Diego, CA)
MIT (Cambridge, MA)
New York U. (New York, NY)
Palomar (San Marcos, CA)
Purdue U. (W. Lafayette, IN)
SDSU (San Diego, CA)
Texas (Austin, TX)
UCB (Berkeley, CA)
UC Davis (Davis, CA)
UCI (Irvine, CA)
UCLA (Los Angeles, CA)
UCSD (San Diego, CA)
U. Arizona (Tucson, AZ)
U. New Mexico (Albuquerque, NM)
U. Oklahoma (Tulsa, OK)
U. Rochester (NY)
U. Utah (Salt Lake City, UT)
Washington (Seattle, WA)
Wisconsin (Madison, WI)

European Community

CEA (Cadarache, France)
CFN-IST (Lisbon, Portugal)
Chalmers U. (Göteberg, Sweden)
CIEMAT (Madrid, Spain)
Consorzia RFX (Padua, Italy)
CRPP (Lausanne, Switzerland)
EFDA (Belgium)
FOM (Utrecht, The Netherlands)
Frascati (Frascati, Lazio, Italy)
KFZ (Jülich, Germany)
Helsinki U. (Helsinki, Finland)
IFP-CNdR (Italy)
IPP (Greifswald, Germany)
IST (Lisbon, Portugal)
ITER (Cadarache, France)
JET-EFDA (Culham, United Kingdom)
Kharkov IPT (Ukraine)
Max Planck (Garching, Germany)
U. Dusseldorf (Germany)
UKAEA (Culham, United Kingdom)
U. Naples (Italy)
U. Rome (Italy)
U. Strathclyde (Glasgow, Scotland)

Japan

JAEA (Naka, Ibaraki-ken, Japan)
Hiroshima U. (Japan)
NIFS (Toki, Gifu-ken, Japan)
Tsukuba U. (Tsukuba, Japan)

Russia

Ioffe (St. Petersburg)
Keldysh (Udmurtia, Moscow)
Kurchatov (Moscow)
Moscow State (Moscow)
St. Petersburg State Poly (St. Petersburg)
Triniti (Troitsk)
Inst. of Applied Physics (Nizhny Novgorod)

Other International

Australia National U. (Canberra, AU)
ASIPP (Hefei, China)
IPR (Gandhinager, India)
NFRI (Daejeon, S. Korea)
Nat. Nucl. Ctr (Kurchatov City, Kazakhstan)
Pohang U. (S. Korea)
Seoul Nat. U. (S. Korea)
SWIP (Chengdu, China)
U. Alberta (Alberta, Canada)
U. Toronto (Toronto, Canada)

Fig. 11-1. National and international collaborations in support of the DIII-D research program.

geographical and institutional diversity. These collaborations carry out the integrated DIII-D program mission. General Atomics provides most of the operations support.

In the present DIII-D National Fusion Program about 60% of the scientific staff (full time equivalents) are from collaborating institutions. There are a total of 491 users of the facility (as measured by scientific authorship 2007, 2008), 135 from General Atomics and another 356 from:

- 36 national laboratories [U.S. (7), Japan (2), Europe (17), Russia (5), China (2), Korea, India, Kazakhstan]
- 41 universities [U.S. (26), Japan (2), Europe (6), Canada (2), Russia (2), Korea (2), Australia].
- 15 domestic industrial companies.

The team ranges from undergraduates to senior scientists with three decades or more experience in fusion research. This staff has been recognized for its outstanding research: 11 winners of the APS Excellence in Plasma Physics Award and 55 Fellows of the American Physical Society.

11.1.1. The DIII-D National Team

The core of the DIII-D National Team consists of about 90 operating staff and ~120 research scientists (the majority of which are collaborators). Most of the collaborators spend a significant amount of time on-site (2 weeks or more per year participating in experiments). Approximately half of the GA physics effort is associated with building and operating hardware, task coordination, building and maintaining computer codes, and similar service activities.

Many of the research scientists provide operations support. This includes diagnostic and data acquisition maintenance, diagnostic operation, system calibration, and data reduction/analysis. As the sophistication of experiments on DIII-D has increased, diagnostic and data analysis needs have also grown. Many diagnostics require active operation to adjust gains, filters, aiming, and timing to suit the needs of particular experiments. Subsequent to the run day, many data require further specialized reduction and analysis before they can be compared against other data or simulations. Where possible, instruments and data reduction have been automated, but in general, supporting experiments places significant demands on the research scientists who are part of the DIII-D national team.

In addition to GA, there are seven major collaborating institutions that have broad programmatic area responsibilities on multiple topics and may carry management responsibilities. Major collaborators join with GA to form the Executive Committee to guide the programs strategic and near-term directions. The programmatic responsibilities of the major DIII-D collaborators are given in Table 11-1.

Other collaborations involve both universities and national and international laboratories and institutes. Major university collaborations will be covered in Section 11.3. The programmatic roles of other institutions, both public and corporate, and national and international, appear in Table 11-2.

Table 11-1
Programmatic Responsibilities of Major DIII-D U.S. Collaborators 2008

LLNL
- Lead role in edge and divertor physics, diagnostics, and modeling
- Leading role in mass transport and tritium retention studies
- Responsible for MSE diagnostic to measure internal current profiles
- Active role in Advanced Tokamak program
- Diagnostic support

PPPL
- Active stabilization of resistive wall modes
- Advanced tokamak transport barrier studies
- Rotation and momentum transport
- FW and ECH support and profile control studies
- Active role in fast ion physics studies
- Tokamak Operations and Diagnostics support

ORNL
- Leads pellet injection program
- Massive gas puff for disruption mitigation
- Active role in AT program including scenario modeling
- FW transmission line and FW physics
- Boundary physics and diagnostic development

SNLA
- Active role in divertor physics and ELM suppression
- Langmuir probe diagnostics
- Plasma wall interactions

UCLA
- Thrust and ITPA leadership
- Broad spectrum of turbulence measurements
- Anomalous electron transport
- Transport barriers
- Advanced FIR and μwave diagnostics

UCSD
- SOL transport and flows
- ELM control and fast edge probes
- Disruption mitigation studies, runaway electrons, soft x-ray diagnostics
- H-mode physics
- Dust measurements

Columbia U.
- Leading role in resistive wall mode control
- Leading role in high beta AT experiments

Table 11-2
Programmatic Roles of Other Collaborations 2008

ASIPP (China)
Scientist exchanges, Plasma Control System
EAST tokamak collaboration
Integrated modeling

CEA-Cadarache (France)
Coordinated experiments with Tore Supra

CompX
Fokker-Planck modeling

CPI
Gyrotron operation

INEL
Reliability studies

IPP (Germany)
Coordinated experiments with ASDEX-U

JAEA (Japan)
Integrated physics and control of high performance steady state tokamaks

JET (UK)
Coordinated experiments with JET
MHD modeling and disruption studies

KFA Julich (Germany)
QMB and tritium retention studies

KSTAR/NFRC (Korea)
Advanced tokamak physics integration
Divertor design and Plasma Control System

Kurchatov Institute (Russia)
DINA disruption modeling for ITER

LANL
MHD modeling

MAX PLANCK (Germany)
Fast divertor IRTV

NREL/Calabassas Creek Research (SBIR)
Amorphous silicon R&D (microwave irradiation)

SWIP (China)
Integrated modeling

Troitsk (Russia)
Disruption modeling

U. Wisconsin
BES diagnostic
Turbulence, zonal flows and transport

11.1.2. International Collaborations

The DIII-D international collaboration program continues to provide a broad source of innovative ideas and opportunities which support the DIII-D research program. Throughout, the DIII-D Program has benefited from the activities in many foreign collaborating institutions. The guiding principle of the DIII-D international collaboration program is to enhance the DIII-D research program through a combination of a detailed exchange of scientific information with foreign researchers and participation in experiments on other fusion facilities which complement current experiments carried out on DIII-D. The present and planned collaborations are closely related to the research areas of prime interest on DIII-D, namely the thrusts and topical research areas. International collaborations are covered more fully in Chapter 12.

11.2. NATIONAL LEADERSHIP ROLE AND PROGRAM LINKAGES

A key element of this DIII-D Program Plan is to provide national program leadership arising from the DIII-D mission: ***optimization of the tokamak approach to fusion energy.*** The responsibility of the DIII-D National Team extends beyond the conducting research on DIII-D to contribute to six additional entities or groups:

1. ITER: The DIII-D program will continue to work to ensure the success of ITER and to enable the U.S. ITER Project Office to fulfill its commitments to the international ITER project. ITER is the highest priority new facility within the DOE Office of Science. The DIII-D National Program is addressing key issues related to the design, construction, and operation of ITER. DIII-D capabilities allow researchers to simulate many aspects of ITER operation and research on DIII-D has led to expansion of ITER capabilities. The International Tokamak Physics Activity (ITPA) provides coordination of the physics input to the international ITER Organization and is covered in Chapter 12.
2. U.S. Burning Plasma Organization: The creation of the U.S. Burning Plasma Organization provides a framework for organizing research related to next-step experiments in the U.S. and abroad. Many DIII-D Program scientists participate in this organization (including serving as Deputy Director) and the DIII-D facility is featured prominently in its research plans.
3. Theory: The DIII-D Program will continue to work with and stimulate theory development throughout the broader U.S. Theory Program. Experiments on DIII-D provide an important opportunity to validate theory and simulation; conversely, new theoretical understanding motivates significant components of the experimental program.
4. Enabling Technology: DIII-D will continue to participate in developing enabling technologies critical to the future of the tokamak in burning plasma experiments (e.g., ECH systems, rf systems, and plasma facing components).
5. U.S. Fusion Experiments: DIII-D will maintain its active participation and collaboration with other domestic programs. We will continue to coordinate research plans with other tokamak facilities. The many international collaborations supported by the DIII-D Program are covered separately in Chapter 12.
6. Broader U.S. Scientific Community: The DIII-D Team will continue to set a high standard of scientific research. The DIII-D team will make available, when appropriate, the well-diagnosed DIII-D high temperature plasma facility for nonfusion plasma physics research, and will

communicate the excitement and progress of fusion energy science. DIII-D will provide a rich training ground for next generation fusion researchers conducting research on ITER.

11.2.1. DIII-D Research in Support of ITER

The DIII-D Research Program is committed to the success of the ITER experiment; by far the largest share of experimental time in 2006 was focused on ITER-relevant research. As this is written, the ITER agreement is in place, the U.S. ITER Project Office located at ORNL is functioning, the ITER International Team (IT) is forming, people are relocating to Cadarache (France), and a major design review activity is nearly complete. The DIII-D National Fusion Program supports the ITER Project by participating in the work of the International Tokamak Physics Activity and the U.S. Burning Plasma Organization, providing physics and engineering analysis in response to specific requests from the U.S. ITER project office, and by conducting experiments suggested by ITER scientists.

DIII-D is an ideal research facility for conducting research in support of ITER. Operational flexibility allows similarity experiments with other tokamaks in the U.S. and abroad to validate fundamental physics understanding, and makes it possible to simulate the ITER shape and collisionality to anticipate future results once ITER begins operating. DIII-D has a state-of-the-art comprehensive diagnostic set maintained and operated by a national team of scientists that provides a strong basis for model validation. The DIII-D heating and current drive systems, coupled with its advanced plasma control systems, provides further ability to isolate and elucidate underlying physics. Finally and most importantly, the DIII-D national team is highly motivated to help ITER succeed by providing relevant data, analysis, and expertise.

In the near term the ITER Project is seeking help addressing urgent design questions related to items such as plasma startup scenarios and control, first-wall materials and maintenance, ELM control, and feedback control of instabilities (e.g. neoclassical tearing modes and resistive wall modes). The DIII-D program has responded to requests from ITER by allocating run time to conduct targeted experiments, providing support to address specific questions, sending people to the ITER site to work with the ITER team on specific issues, and working through the U.S. Burning Plasma Organization to define and carry out high priority research tasks. The DIII-D program plans to spend nearly 5000 man hours in CY07 working on tasks related to the ITER design review.

In FY08, half the run time on DIII-D is directed to ITER-urgent research topics, as shown in Fig. 11-2. Furthermore, the DIII-D Experimental Science Division was reorganized in early FY07 to better manage the conduct of ITER-relevant research. The development of ELM control via application of resonant magnetic perturbations (RMP) is one example of the close connection between the DIII-D and ITER programs. RMP provides a means for reducing the size of ELMs to acceptable levels in ITER, or eliminating them altogether. The ITER design team is considering how to implement such control coils and is looking to the DIII-D team for physics-based design requirements. Other ITER focus areas in FY08 operations include control of neoclassical tearing modes (NTM) by ECH, disruption mitigation experiments, tritium retention and plasma wall interaction, pedestal studies, and high performance hybrid scenario development. Two-thirds of the DIII-D FY08 and FY09 program milestones proposed to DOE relate to ITER R&D.

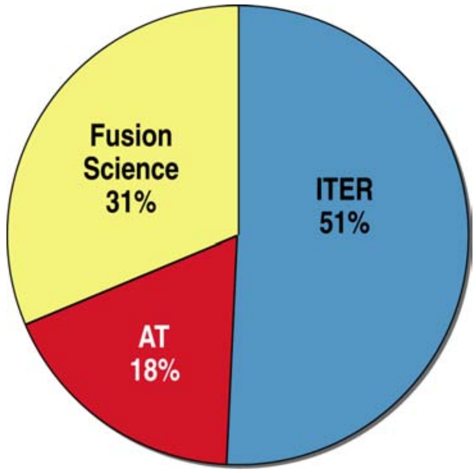

Fig. 11-2. DIII-D run time allocation for FY08 showing balance between major program elements.

Members of the DIII-D National Team are actively engaged with the international fusion community in conducting ITER-related R&D. Table 11-3 lists existing collaborations between DIII-D scientists and others related to ITER research. These collaborations leverage the capabilities of the DIII-D facility in significant ways: e.g., U.S. research teams gain experienced international experts who bring fresh perspectives and new ideas with them, DIII-D data can be integrated into international databases more effectively, and U.S. scientists gain access to international facilities with the corresponding ability to conduct more comprehensive experiments. In addition to individual international collaborations related to ITER, DIII-D team members are active in the International Tokamak Physics Activity (ITPA) and lead several of the working groups (see Table 12-4). The DIII-D program is well represented at the biennial ITPA meetings and experimental proposals are developed that are aligned with ITPA research goals.

In the future, we see ITER's needs shifting from design-related issues to operational issues. We anticipate that DIII-D will help develop startup scenarios, develop experience operating in hydrogen, and simulate operation of the ITER control system. Hydrogen operation is important because the ITER research plan will include significant operation with hydrogen to minimize activation, whereas all high power, high performance tokamaks have operated exclusively in deuterium for the past 20 years. We believe that, as a result of these R&D activities in support of ITER, the DIII-D facility will provide excellent training for the next generation fusion scientist in the U.S. who will assume responsibility for conducting fusion experiments on ITER.

11.2.2. DIII-D Support for the U.S. Burning Plasma Organization

The U.S. Burning Plasma Organization (USBPO) was created in FY06 to coordinate relevant U.S. fusion research with broad community participation "to advance the scientific understand of burning plasma and ensure the greatest benefit from a burning plasma experiment." DIII-D scientists were instrumental in setting up the organization. Dr. T.S. Taylor, the DIII-D Program Director served as the first Deputy Director of the USBPO, and Dr. C.M. Greenfield serves in this capacity now. Drs. S.L. Allen of Livermore, G.A. Navratil of Columba U., C.C. Petty and T.S. Taylor of GA, and Prof. G. Tynan (UCSD), all part of the DIII-D program, serve on the USBPO Council. Drs. R. Nazikian (PPPL), E. Doyle (UCLA), W. Heidbrink (UC Irvine), and D.A. Humphreys, R. Boivin, J. Kinsey, and T.C. Luce (from GA) serve in leadership of the Topical Groups and on the USBPO Research Committee.

Table 11-3
DIII-D Collaborations Related to ITER Physics 2008

Topic	Collaborating Institution	Key Collaborator	DIII–D Contact
Disruption database	MIT	R. Granetz, S. Angelini	A. Hyatt
	NSTX	J. Menard	A. Hyatt
	JT-60U	Y. Kawano	A. Hyatt
	ASDEX-U	G. Pautasso	A. Hyatt
	Compass-D	T. Hender	A. Hyatt
	JET	P. deVries, V. Riccardo	A. Hyatt
	TCV	J.-Y. Martin	A. Hyatt
Disruption mitigation	UCSD	E. Hollmann,* J. Yu, A. James, V. Izzo*	E. Strait
	MIT	R. Granetz	E. Strait
	ORNL	T. Jernigan*	E. Strait
Disruption modeling	Kurchatov Institute (Russia)	V. Lukash	D. Humphreys
Edge fluctuations	UCSD	D.L. Rudakov*	W. West
Edge modeling	U. Toronto	P. Stangeby, D. Elder, A. McLean, Y.-R. Mu*	W. West
ELM control	UCSD	R. Moyer*, I. Joseph*, S. Mordijck*	K. Burrell
	LLNL	M. Fenstermacher*	E. Strait
	Greifswald	R. Schneider, A. Runov, M. Jakubowski	T. Evans
	Helsinki U.	J.S. Lonnroth	T. Evans
	CEA-Cadarache	M. Becoulet, E. Nardon, P. Thomas, P. Garbet	T. Evans
	SNLA	J. Watkins	T. Evans
	FZ Jülich	K.H. Finken B. Unterberg, O. Schmitz, H. Frerichs, M. Lehnen, Y. Liang, H.R. Koslowski, D. Reiser, D. Reiter	T. Evans
	IST, Portugal	M.F. Nave	W. West
	JET	V. Parail, R. Buttery	T. Evans
	SWIP	L.W. Yan	T. Evans
	Heinrich-Heine Univ. (Düsseldorf)	A. Wingen	T. Evans
Erosion	IPP Julich	V. Phillips	W. West
Heat flux	LLNL	C. Lasnier*	W. West
	Max Planck	M. Jakubowski	C. Lasnier
Langmuir probes	SNLA	J. Watkins*	W. West
Magnetic diagnostics	KSTAR	J.G. Bak	E. Strait
Magnetic error fields, locked modes	HBT–EP/Columbia Univ.	G. Navratil	M. Schaffer
	PPPL	J. Menard, J.-K. Park	M. Schaffer
	IFS	R. Fitzpatrick	M. Schaffer
	Culham	T. Hender, D. Howell	M. Schaffer
	JET	R. Buttery	M. Schaffer
Neoclassical tearing mode control	MIT/Alcator C-Mod	S. Wolfe, R. Granetz	R. La Haye
	ASDEX-U	S. Gunter, M. Maraschek, H. Zohm	R. La Haye
	JET	R. Buttery, T. Hender, R. Howell	R. La Haye
	IFS	F. Waelbroeck	R. La Haye
	Lehigh University	G. Bateman	R. La Haye
	LANL	A. Glasser	R. La Haye
	PPPL	F. Perkins	R. La Haye
	Columbia Univ.	S. Sabbagh	R. La Haye
	PPPL	S. Gerhardt	R. La Haye
	U.Wisconsin	C.P. Hegna	R. La Haye
	U. Tulsa	D.P. Brennan	R. La Haye
	IPR	A. Sen, D. Raju	R. La Haye
	Tech-X Corp.	S. Kruger	R. La Haye
	PPPL	R.A. Ellis, J. Hosea	J. Lohr
	ORISE	F. Volpe*	E. Strait
Pellet injection studies	ORNL	L. Baylor, T. Jernigan*	K. Burrell
Plasma control	TRINITI	R. Khayrutdinov	D. Humphreys
	MIT	D. Whyte	D. Humphreys
RWM control for ITER	Columbia	H. Reimerdes*	E. Strait
	ORISE	F. Volpe*	E. Strait
Sawtooth control	CRPP-Lausanne	O. Sauter	R. La Haye
	UKAEA-Culham	R. Buttery	R. La Haye
Surface erosion	SNLL	R. Bastasz	C. Wong
	MIT	D. Whyte	W. West
	IPP Julich	A. Litnovsky	W. West
	SNLA	W. Wampler	W. West

*On-site personnel.

The USBPO Research Committee meets regularly by teleconference to discuss and develop R&D plans for the national programs. The USBPO identified 14 high priority research tasks for 2006–2008 relevant to ITER that the U.S. fusion community can address using DIII-D and other U.S. facilities; 12 scientists affiliated with the DIII-D program are serving as PIs on these tasks. Over 70 scientists who participate in DIII-D research have joined the USBPO as members of the various Topical Groups and work with the Topical Group leaders to assess progress on burning plasma research and to update research plans. This year, we specifically recruited members from the USBPO Research Committee to help facilitate discussions at the C-Mod, NSTX, and DIII-D National Tokamak Planning Workshop at MIT in September 2007 to ensure that DIII-D plans address issues relevant to research on Burning Plasmas.

Over the longer term, the USBPO expects to address a broad set of issues related to the development of the science of burning plasmas that extends beyond the ITER mission. Future fusion experiments will likely study steady-state fully noninductive operation at high plasma pressure, such as considered in the ARIES-RS and AT reactor studies and the proposed Fusion Development Facility (FDF). Such devices will require integration of many components related to present-day fusion research, such as maintaining 100% noninductive current drive along with highly radiating boundaries while using actively controlling MHD modes to avoid disruptions. At present, DIII-D Program scientists are participating in many collaborations related to such Steady-state Integration issues relevant to the long-term development of fusion energy, as shown in Table 11-4.

11.2.3. DIII-D Research and U.S. Theory Program

The DIII-D program prominently features the close interaction between theorists and experimentalists. Theory motivates and guides formulation of experimental proposals and, conversely, DIII-D experimental observations are often used to guide development of theory and computational tools. Theorists are included in DIII-D near-term and long-term program planning and serve on the DIII-D Research Council. They are actively involved in the planning, execution and analysis of DIII-D experiments. This interaction together with detailed comparison of theoretical predictions with experiments have led to the identification of a great deal of important new physics.

The GA Theory Group focuses on five areas of research:

- Stability and MHD,
- Confinement and Transport,
- Energetic Particles, Heating and Current Drive,
- Integrated Modeling,
- Innovative Concepts.

One third of the 15 theorists are APS Fellows and two have won the Rosenbluth Prize for Fusion Theory. Besides GA staff, the Theory Group has benefited from close collaboration with other senior theorists. Past and present long-term visitors and consultants have included M.N. Rosenbluth (UCSD), F.W. Perkins (PPPL), M.S. Chance (PPPL), X. Garbet (CEA-Cadarache), S. Günter (IPP-Garching), H.R. Wilson (Culham), F.L. Waelbroeck (IFS), A. Bondeson (Goteborg Sweden), A. Hassam (UMd), W.M. Nevins, L.L. Ledestro, and X. Xu from LLNL, L. Chen (UC-Irvine), J. Callen (UW), and A.M. Popov (Moscow State).

Table 11-4
DIII-D Collaborations Related to Steady-State Integration Physics 2008

Topic	Collaborating Institution	Key Collaborator	DIII–D Contact
Active mode control	IFS	R. Fitzpatrick	R. La Haye
	PPPL	M. Okabayashi, A. Nagy,* R. Hatcher	E. Strait
	FAR-Tech, Inc.	Y. In, N. Bogatu,* J. Kim,* J.S. Kim*	T. Luce
AT control modeling	CEA-Cadarache	V. Basiuk, F.P. Imbeaux, M. Schneider	T. Luce
	Lehigh U.	E. Schuster, Y. Ou	J. Ferron
AT modeling	LLNL	T. Casper	T. Luce
AT scenario development	UCLA	E. Doyle*	T. Luce
	UKAEA	C. Challis	T. Luce
	JAEA	S. Ide	T. Luce
	IPP	G. Sips	T. Luce
AT scenario modeling	ORNL	M. Murakami	R. Prater
	ORNL	J.-M. Park*	H. St. John
ECH/NBCD	JAEA/JT-60U	T. Suzuki	C. Petty
Fast wave	PPPL	J. Hosea, E. Fredd, N. Greenough, J.R. Wilson	R. Pinsker
ICRF modeling	ORNL	M. Murakami	R. Prater
ICRF technology	ORNL	F.W. Baity	R. Pinsker
MSE measurements	LLNL	M. Makowski,* C. Holcomb*	T. Luce
	U. Arizona	R. Chipman	S. Allen
RWM Physics	Columbia University	G. Navratil, A. Garofalo,*J. Bialek, H. Reimerdes,* M. Lanctot*	T. Luce
	PPPL	M. Okabayashi	T. Luce
	PPPL	M. Chance	M. Chu
	Chalmers Institute	Y. Liu	M. Chu

*On-site personnel.

Through its interactions with the GA Theory program, the DIII-D program maintains close connection to the SciDAC (Scientific Discovery through Advanced Computing), FSP (Fusion Simulation Project), and other OFES Theory Program Initiatives. There are several SciDAC FSP prototype Centers and fusion SciDAC projects which feature strong connection to the DIII-D program: Center for Simulation of Wave Particle Interaction with Magnetohydrodynamics (SWIM), Center for Plasma Edge Simulation (CPES), and Petaflops for Gigawatts (FACETS), Gyrokinetic Particle Simulation Center (GPSC), Center for Simulation of Wave-Plasma Interactions (CSWPI), and Center for Extended MHD Modeling (CEMM). In the following, we will give a brief description of our relationships with these SciDAC projects.

- GYRO simulations challenge and complement work carried out in the SciDAC Center for Gyrokinetic Particle Simulations of Turbulent Transport in Burning Plasmas (GPSC) project. The data contained within the GYRO transport database will prove valuable for benchmarking purposes. With regard to core transport, we also emphasize that the GLF23 and TGLF codes are targeted as core transport modules for FACETS

- Exploratory work by E. Belli using the “rapid prototype” EGK code is done in support of the Edge Simulation Laboratory (ESL) project, which is itself complementary to Center for Plasma Edge Simulation (CPES).
- ORBIT-RF is being further upgraded for the TORIC and AORSA RF codes. This work is expected to make a significant contribution to the Center for the Simulation of Wave Interactions with Magnetohydrodynamics (SWIM) project.
- The GA Theory group, working with the NIMROD team and the Center for Extended Magnetohydrodynamic Modeling (CEMM) project, put a large effort into linear and nonlinear ELM simulations.
- NIMROD simulations of ELM stabilization by resonant magnetic perturbations (RMPs) will be performed by V. Izzo in direct collaboration with the CEMM effort. In addition, NIMROD will be applied to simulate mitigated and unmitigated disruptions using improved radiation and pellet/gas jet penetration models.

Serving in its role as a national fusion facility, DIII-D data is made available to theorists worldwide via a number of collaborations targeting some of the most challenging issues confronting fusion energy science. Table 11-5 lists DIII-D collaborations related Integrated Modeling. The GA Theory group is uniquely placed in this regard with its past history of leadership in this area. Both the GA theory staff and the experimental research staff are committed to helping collaborators in the U.S. Theory Program with access to the data for validation of theory. Infrastructures have been set up to facilitate this interaction, which produce a continuous dialogue between theory and experiment. Here as examples we highlight four major research topics featuring close coupling between theory and experiment:

Edge Pedestal and Control of Edge Localized Modes (ELMs). Gaining an understanding of ELMs, including onset conditions and dynamic evolution, has come to the fore as a critical issue for ITER and other burning plasma experiments, both because of the potential impact of ELM pulses on material surfaces, and because ELM onset places an effective constraint on the pressure at the top of the edge barrier (the ”pedestal height”), which strongly impacts core confinement and overall fusion performance. The GA Theory group has made important breakthroughs in physics understanding of ELMs and ELM-free operation over the past few years. In particular, the peeling-ballooning model of ELMs, pioneered by GA in collaboration with Culham, has continued to be quantified, elaborated, and extensively and successfully tested against experimental data from DIII-D and other tokamaks. This model serves to guide proposed ELM-control experiments.

Rotation and Wall Stabilization of Global MHD Modes. In the next generation of steady-state advanced tokamaks, high-performance plasmas must remain stable over time scales long compared to the flux diffusion time τ_w of the external resistive wall. It is also desirable to have $\beta_N = \beta/(I/aB_T)$ as high as possible for optimal performance. If possible, it should exceed the maximum β_N^{NW} stable to global modes in the absence of a conducting wall. This can be achieved by placing a conducting wall close to the plasma. However, when β_N exceeds β_N^{NW}, the perturbation flux due to the unstable kink mode can diffuse through the external conducting wall on a time scale longer than τ_w, resulting in the resistive wall mode (RWM). To fully realize the advantage of the conducting wall, the RWM must be stabilized. The goal of research on the RWM is to find methods for its stabilization and to develop a quantitative model to predict its stability in present and future advanced tokamaks.

Table 11-5
DIII-D Collaborations Related to Integrated Modeling 2008

Topic	Collaborating Institution	Key Collaborator	DIII–D Contact
3D MHD	ORNL	S. Hirshmann	L. Lao
	PPPL	M. Zarnstorff	L. Lao
	Auburn University	S. Knowlton, J. Hanson	L. Lao
	ORNL	E. Lazarus*	L. Lao
	CRPP-Lausanne	A. Cooper	A. Turnbull
	Columbia University	A. Boozer	A. Turnbull
Edge stability	JAEA/JT-60U	Y. Sakamoto, Y. Kamada, N. Oyama	L. Lao
	York University	H. Wilson	P. Snyder
	LLNL	X. Xu	P. Snyder
	Euratom/IST	F. Nave	P. West
Edge modeling	LLNL	M. Groth,* G. Porter	R. Prater
	UCSD	S. Krashenninikov, Pigarov	W. West
Energetic particle stability	UC-Irvine	L. Chen, Z. Lin	M. Chu
Equilibrium reconstruction (EFIT)	MIT/Alcator C–Mod	S. Wolfe	L. Lao
	Culham/MAST	L. Appel	L. Lao
	KSTAR/NFRC	K.I. You	L. Lao
	Columbia, PPPL NSTX	S. Sabbagh	L. Lao
	JET	V. Drozdov, E. Solano	L. Lao
	ASIPP	C. Zhang, S. Wang	L. Lao
	SWIP	J. Dong	L. Lao
	CEA Cadarache	W. Zwingmann	L. Lao
Gyrokinetic simulation	Columbia	C. Estrada-Mila	J. Candy
ICRF, ECH physics	MIT	M. Porkolab	R. Prater
Integrated Modeling	ASIPP	G. Li,* Q. Ren,* W. Guo,* C. Pan*	L. Lao
	SWIP	A. Wang	L. Lao
	ORNL	J.M. Park*	L. Lao/R. Prater
MHD analysis	University of Wisconsin	J. Callen	A. Turnbull
	UC Berkeley	X. Li	A. Turnbull
MHD mode analysis	FAR-Tech	J.S. Kim	A. Turnbull
Neoclassical tearing modes	Univ. of Wisconsin	J. Callen, C. Hegna	R. La Haye
	JT-60U	N. Hayashi, A. Isayama	R. La Haye
Neutral modeling	ORNL	L. Owen	R. Prater
Nonlinear MHD simulations	SAIC	D. Schnack	M. Chu
	LANL	A. Glasser	M. Chu
Nonlinear MHD stability	NYU/Courant Inst.	P. Garabedian	A. Turnbull
Pellet ablation	Ukraine	R.V. Samulyak	P. Parks
Pedestal	ASDEX-U	C. Maggi	R. Groebner
	NSTX	R. Maingi	T. Osborne
Pedestal modeling	Germany	K. Hallatschek	J. Candy
Pedestal, neutrals	Georgia Tech.	W. Stacey	R. Groebner
Resistive MHD code development	FAR-TECH	S. Galkin	A. Turnbull
	LANL	A. Glasser	A. Turnbull
Resistive and edge stability	MIT	L. Sugiyama	L. Lao
Resistive stability	Culham	Y. Liu	M. Chu
	PPPL	M. Chance	M. Chu
	U. of Tulsa	D. Brennan	L. Lao
RF modeling	CompX	R. Harvey, A.P. Smirnov	R. Prater
Simulation of turbulence parameters	U. Alberta	R. Sydora	T. Rhodes
	PPPL	D. Mikkelsen	R. Waltz
Test of theory-based transport models and turbulence simulations	U. Texas	K. Gentle	R. Prater
Tokamak modeling	ORNL	M. Murakami, J.M. Park	R. Prater/T. Luce
Transport modeling	PPPL	R. Budny	R. Waltz

*On-site personnel.

With balanced beam injection, it was discovered that RWMs in DIII-D can remain stable at very low rotation. This was also corroborated by experimental results in JT-60U. The overall rotation profile, although not zero, is very close to zero across the whole plasma cross section. Also, throughout the discharge history, the outside 40% of the rotation profile is constant. The MARS-F code with kinetic damping has been employed to study the stability of these plasmas and found to agree with the experimental observation.

Sheared Flows and Turbulent Transport. The theory group at General Atomics (GA) has always approached the problem of transport in tokamak plasmas with a four-part strategy: First, we aim to advance the most physically comprehensive but tractable numerical simulation of the ion and electron gyroradius scale turbulence possible. It is now practical to do full ***gyrokinetic simulations.*** Second, computationally faster ***theory based transport models*** accurately fitted to these simulations are developed in order to predict plasma profiles self-consistent with sources. Third, we ***compare transport simulations and models with experiment.*** This work is greatly enhanced by our close relationship to the DIII-D National Tokamak Program. Finally, ***analytic theory*** is used to extend existing formulations and provide a qualitative understanding of simulations and experiments. In collaboration with the wider fusion theory community, these elements are cycled to advance our understanding of transport in tokamak plasmas.

The gyrokinetic equations provide a fundamental theory of low-frequency drift instabilities in tokamak plasmas. Recent groundbreaking work with GYRO has focused on the simulation of short-wavelength ETG transport including kinetic ion physics. After a decade of successful use, the limitations of the GLF23 model are now well known. The remedy is a more accurate, and therefore more computationally intensive, gyrofluid model. Considerable effort has gone to the development of the TGLF model (Trapped Gyro-Landau Fluid) to replace GLF23. We expect the TGLF model will be our principal tool for experimental analysis, discharge scenario development, and ITER core performance projection. The focus of our TGLF transport model applications will be testing the new physics compared to GLF23 using DIII-D data (shaped geometry, low aspect ratio, and modeling the pedestal region).

Heating, Current Drive, and Energetic Particles. The DIII-D fast wave current drive (FWCD) experiments on neutral beam (NB) heated plasmas have demonstrated that the presence of beam ion cyclotron harmonic resonances may lead to partial or strong damping of FW on beam ions, resulting in a significant reduction of FWCD efficiency. Since FWCD provides a good control of the central q-profile and an improvement of off-axis electron cyclotron current drive (ECCD) efficiency by increasing β_e locally, conditions for strong damping of FW on energetic particles at high harmonics should be well understood for high power FWCD on DIII-D and future devices. Validation of the FW damping mechanism on energetic beam ions will be critical for evaluating the efficiency of FWCD in burning plasmas dominated by alpha particles.

The finite orbits of the energetic particles and full 2D structure of the rf-wave electric field have to be accounted for when developing a quantitative theoretical model rf current drive. For this purpose, the generalized quasi-linear (Q-L) diffusion operator was implemented using a Monte-Carlo technique in the ORBIT-RF code and modifications to the code to add MHD perturbations to the guiding center motion have been proposed. The ORBIT-RF code will be capable of calculating modifications in the distribution function of energetic particles in both configuration and phase spaces through the resonant interactions with high frequency ICRF waves and low frequency MHD waves. Results will be validated with measurements from advanced multi-channel fast ion D_α (FIDA) diagnostic on DIII-D. Our proposed studies in this area will provide the scientific basis for reliably achieving steady-state advanced tokamak (AT) regimes in existing tokamak experiments and future burning plasma experiments such as ITER.

11.2.4. Role of DIII-D Research for Enabling Technologies, Contributions and Needs

Progress in fusion has been closely coupled to advances in enabling technology. Success in many advanced tokamak studies to date has been achieved transiently; e.g., using heating to control the resistive evolution of the ohmic current. Further progress needs active control tools to maintain the desired profiles of current density and plasma pressure for several resistive times. DIII-D has identified electron cyclotron heating and current drive for off-axis profile control, fast wave heating, active nonaxisymmetric MHD mode control, divertor pumping for density control, and active real-time plasma feedback control as near term control needs for advanced tokamak development. Where possible, the DIII-D group develops, tests, and applies new control systems as a natural part of its research program. DIII-D also supports longer-range work related to development of new plasma facing materials. Table 11-6 lists active collaborations between the DIII-D program and other institutions related to Enabling Technologies for Plasma Control, Operations, and Technology for ITER and beyond.

The key DIII-D enabling technology need is for reliable, long-pulse high power (P_{tube} > 1 MW) gyrotrons at 110 GHz. Long pulses are essential to control the current density profiles due to the long current current diffusion times of the plasma. DIII-D supported testing of a new, 1.5 MW 110 GHz depressed collector tube for the Virtual Laboratory for Technology (VLT). CPI manufactured the depressed collector tube, and the DIII-D program provided the requisite supporting infrastructure and manpower to operate the system. In the future, higher power tubes will save money on ECH systems since fewer power supplies and control systems will be needed. The Advanced Tokamak program also will benefit from development of improved launchers to allow fast tracking for MHD mode control.

In addition to ECH, improved rf antenna technology is needed for Advanced Tokamak research. There are three rf heating and current drive systems on DIII-D operating at 60–90 MHz. Reliable high power rf coupling (multi-megawatt) into AT discharges provides increased control flexibility by providing increased electron heating to improve current drive efficiency. Key to improving coupling is development of new current strap configurations to reduce operating voltages and improved feedthrough designs. New rf design software exists which can be used to aid in this work. The DIII-D program currently supports research related to developing new antenna designs.

Real-time plasma profile control and feedback stabilization of instabilities is another area where further development is needed and where DIII-D is playing an active role. The DIII-D team has exported its plasma control system (PCS) to the new EAST and KSTAR tokamaks. The active control of plasma profiles necessitates internal measurements with feedback to heating, fueling, and current drive systems; future burning plasma experiments will add new challenges due to the presence of internal alpha heating and limited diagnostic capability. Disruption avoidance and mitigation is also needed on next generation experiments such as ITER; DIII-D is presently working with the fusion community to develop new techniques.

Plasma facing materials will be key to development of fusion energy over the long term. New materials are needed which are compatible with high performance tokamak operation and which exhibit long lifetime in a nuclear environment with minimal tritium uptake. The DIII-D DiMES and MiMES materials sample exposure systems, coupled with its inherent shaping flexibility and comprehensive set of diagnostics, provides the means to effectively and efficiently test new materials for fusion.

Table 11-6
DIII-D Collaborations Related to Plasma Control, Operations, and Technology 2008

Topic	Collaborating Institution	Key Collaborator	DIII–D Contact
2D MHD control simulation	TRINITI	R. Khayrutdinov, V. Dokouka	D. Humphreys
	EFDA	M. Cavinato	M. Walker
	LLNL	T. Casper, W. Meyer, L. LoDestro	D. Humphreys
	Lehigh U.	E.J. Schuster	T. Luce
Amorphous silicon crystallization	NREL	D. Young	J. Lohr
	CCR	J. Nielsen	J. Lohr
EAST physics operator training	ASIPP	Q. Yuan, B. Xiao	A. Hyatt
EAST plasma control system	ASIPP	J. Luo, H. Wang, B. Shen, B. Xiao, Q. Yuan, J. Qian	D. Humphreys
Equilibrium control	U. Rome	L. Zaccarian	M. Walker
	Lehigh U.	E. Schuster, M. Alsarheed	M. Walker
	PPPL	C. Rowley	M. Walker
KSTAR plasma control system	NFRC	H. Jhang, S.H. Hahn, S.H. Seo, J.-Y. Kim, M. Kwon, Y.K. Oh	D. Humphreys
MAST plasma control system	UKAEA-Culham	G. McArdle	J. Ferron
Nonaxisymmetric control	CREATE-Italy	A. Pironti, F. Villone	D. Humphreys
NSTX plasma control analysis/design	PPPL	D. Gates, D. Mueller	J. Leuer
NSTX plasma control system	PPPL	D. Gates, D. Mastrovito	J. Ferron
Pegasus plasma control system	U. Wisconsin-Madison	M. Bongard	D. Humphreys
Plasma control for ITER	EFDA Garching	M. Cavinato	D. Humphreys
	EFDA-Barcelona	A. Portone	D. Humphreys
	NFRC	J.-Y. Kim, H. Jhang	D. Humphreys
Profile control	CEA-Cadarache	D. Mazon, E. Joffrin	P. Gohil
	Lehigh U.	E. Schuster, Y. Ou	M. Walker
Real-time scanning ECH launcher	PPPL	R.A. Ellis, J. Hosea	J. Lohr
Silicon annealing	U. Wisconsin	K. Thompson	J. Lohr

11.2.5. Collaboration with other U.S. Fusion Experiments

To serve the U.S. Fusion Program more fully, the DIII-D Program has established and will seek to expand linkages to other magnetic confinement experiments in the U.S. Program. Other experiments can supply tests of concepts and supporting information that might be utilized in later stages of the DIII-D Program. The DIII-D Program, representing a large collaboration of institutions, will seek to assist linked programs to succeed in their research endeavors. Joint experimental work falls into three categories:

1. Collaborative – teams may work together to transfer experience from one facility to another to expand overall capabilities and expertise. Research teams may travel to other facilities to achieve specific goals.
2. Complementary – similar experiments performed on separate facilities to expand the parameter range and improve understanding. Close coordination on subsequent data analysis.
3. Confirmatory – a key element of scientific research involves reproducing key results on different experiments. This is especially valuable when developing understanding and testing new models. We describe below some of the present and past linkages that incorporate one or more of these types of collaborative research.

ALCATOR C-Mod is one of the three major tokamak facilities in the U.S. Program. Located at MIT, it complements the research on the DIII-D tokamak with its high toroidal magnetic field and accompanying high density operating capability. Because of the high field, ALCATOR C-Mod pursues Lower Hybrid current drive for current profile control. The lower field in DIII-D makes it more appropriate to use electron cyclotron current drive in DIII-D. Both machines have strong programs in divertor physics with theory and code support for divertor research and edge transport. DIII-D features all-carbon PFCs while C-Mod operates with high-Z boronized walls. C-Mod has recently added divertor pumping, allowing direct comparison with DIII-D particle control experiments. Both machines have active research programs in disruption mitigation. Due to the wide range in size and plasma parameters, DIII-D and Alcator C-Mod play important roles in dimensionless scaling experiments with the larger European JET and Japanese JT-60U tokamaks and scientists from each program participate in focused joint experiments.

National Spherical Torus Experiment (NSTX) is a major spherical torus (ST) at Princeton Plasma Physics Laboratory, which began operation in 1999. NSTX is a proof-of-principle experiment investigating, in depth, many of the important scientific issues for the ST. Scientists from GA and DIII-D collaborators participate in many of the NSTX working groups, such as those in the areas of plasma control, coaxial helicity injection and high harmonic fast wave physics. Aspect ratio scaling experiments in the core and the pedestal have been performed between DIII-D and NSTX and are planned for the future. NSTX is planning on adding a liquid lithium divertor module and is planning on participating in joint experiments on particle control in FY09. Both programs support significant activity related to feedback stabilization of resistive wall modes.

High Beta Tokamak — Extended Pulse (HBT–EP) is a small tokamak at Columbia University studying the issue of wall stabilization. The research is pursuing both the "smart shell" approach to making a resistive wall look superconducting and the use of nonaxisymmetric coils to force plasma rotation in the core plasma. The idea of forcing plasma rotation by rotating magnetic islands in the plasma with external coils ("magnetic stirring") originated at GA by T.H. Jensen. That idea and the smart shell approach are being tested on HBT-EP. Columbia staff has played leading roles in DIII-D high performance and resistive wall mode stability experiments, and development of the VALEN code for feedback system design.

PEGASUS is a small spherical torus experiment at the University of Wisconsin. Its research program is focused on noninductive startup and operation at high beta with high plasma elongation. GA provided design engineering and analysis help for the PEGASUS vacuum vessel and provided the port extensions in order to assist in a more rapid startup of this device. GA has assisted in providing DIII-D Plasma Control System technology to the PEGASUS team.

Madison Symmetric Torus (MST) is a proof-of-principle scale reversed field pinch experiment at the University of Wisconsin. The program is focused on understanding and optimizing energy confinement, understanding and controlling the plasma dynamo that sustains the RFP configuration, developing current drive techniques, and deploying state-of-the-art plasma diagnostics to support physics studies.

The **Fusion Facilities Coordinating Committee (FFCC)** was established in 1998 to facilitate improved coordination between the three major U.S. magnetic fusion facilities (DIII-D, NSTX, and C-Mod) and between the major U.S. facilities and major international facilities. Representative program leaders from the U.S. facilities meet together at least once per year at the OFES with the relevant DOE program managers, the corresponding facility PAC chairs, U.S. BPO representatives, and ITER managers, usually immediately preceding the annual OFES Budget Planning meeting in March. Topics for discussion include operating schedules, research goals, national and international collaboration activities, and ITER-related research activities. Other FFCC meetings take place either by phone or in person throughout the year as needed. Dr. Richard Hawryluk from PPPL has served as chair since the FFCC's inception. Most recently, the FFCC organized a National Tokamak Planning Workshop at MIT in September 2007 to discuss Five Year Program Plans for each facility.

Joint Facility Level-1 JOULE Milestones. In recent years the OFES has established the practice of identifying one high level milestone for the OFES based on conducting coordinated research among the three major U.S. facilities: DIII-D, NSTX, and C-Mod. Each year OFES and program representatives attending the Fusion Facilities Coordinating Committee meeting discuss potential research topics which could best provide important high visibility results in a timely manner through coordinated research activities. Topics and quarterly targets are developed and chosen that reflect expected facility capabilities and OFES/facility research priorities. Each program then adjusts its programmatic milestones to support the joint milestone and takes appropriate action to complete the work. For the most recent budget year (2008), approximately 20% of the experimental run time for DIII-D supports the FY08 Joint Facility Milestone.

11.3. COLLABORATIONS WITH THE BROADER SCIENCE COMMUNITY

The DIII-D program conducts research that advances fusion science across a broad front and involves students, post docs, and scientists from a wide range of national and international research institutions. The DIII-D Team takes seriously its role as a steward of plasma physics, its responsibility to maintain a world class scientific research facility, and its duty to help recruit and train tomorrow's fusion scientists. The Team supports collaborations with universities and laboratories that address fundamental issues related to fusion science. At the end of this section, Table 11-11 summarizes existing collaborations. Within the U.S., the Transport Task Force (TTF) provides a highly visible and effective framework for organizing fundamental research related to fusion energy, much of which is conducted by faculty and students at universities. We now summarize the relationship between the DIII-D program and the universities and the TTF.

University Participation in DIII-D. The active, on-site participation of university scientists is an important part of the DIII-D fusion research program. Throughout the years, many major universities have taken part in the DIII-D program, funded by, direct grants from the OFES as part of the DIII-D program, subcontracts from GA and other DIII-D collaborators, and other direct grants from DOE. This participation has added an important breadth to our research and has provided a mechanism by which we have been able to quickly and cost effectively involve scientists with unique and specific experience and capabilities in our efforts. The experience, which they gain by working with a major fusion research facility, in turn, enhances their ability to contribute to their home university's programs both in teaching and research. Table 11-7 lists present major university collaborations on DIII-D.

Table 11-7
Programmatic Roles of Major DIII-D University Collaborators (2008)

Columbia U.
- Leading role in resistive wall mode control
- Leading role in high beta AT experiments

Georgia Tech
- Pedestal fueling and transport analysis

Lehigh U.
- Profile control modeling

MIT
- Fast wave analysis
- Phase contrast imaging long-wavelength fluctuation diagnostic

Purdue University
- Divertor erosion

U. Arizona
- Polarization analysis for MSE diagnostic

U. Maryland
- Vertical and horizontal ECE measurements

U. Texas
- Transport experiments and modeling
- Fine spatial/temporal scale ECE temperature measurements

U. Toronto
- SOL simulation and analysis
- Plasma wall interaction
- Hydrogenic retention
- Oxygen bake code point removal

U. Wisconsin
- Neoclassical MHD
- Zonal flows and BES fluctuation diagnostic
- LH transition physics

UC Irvine
- Neutron and fusion reaction diagnostics
- Charge exchange diagnostics
- Fast ion confinement and energetic particle physics

UCLA
- Thrust and ITPA leadership
- Broad spectrum turbulence measurements
- Anomalous electron transport
- Transport barriers
- Advanced FIR and μwave diagnostics

UCSD
- SOL transport and flows
- ELM control and fast edge probes
- Disruption mitigation studies, runaway electrons, soft x-ray diagnostics
- H-mode physics
- Dust measurements

New opportunities for expanded university partnerships arise regularly, often through OFES initiatives. The DIII-D program welcomes new ideas from university programs across a broad range of topics. Table 11-8 defines three broad categories of University participation and lists needs that could be addressed by new or expanded university collaborations. The list is meant to be suggestive of possible topical areas and is neither exclusive nor all-inclusive.

Table 11-8
Areas of Potential Additional University Collaboration

Activity	Needs/Opportunities
Diagnostic Instrumentation	SOL flows and particle transport (impurity/main ion)
	Runaway electrons and nonthermal ion/electron distributions
	Turbulence and magnetic fluctuation measurements
	Erosion/Redeposition
	Internal magnetic field measurements
	RF antenna diagnostics
	Plasma rotation (impurity and main ion)
	3D effects
	ELM effects
Experiment and Analysis	Pedestal width
	Main ion particle transport
	Sawtooth physics
	Tearing mode control
	ELM losses
Theory and Modeling	Synthetic diagnostic development
	Scenario modeling
	SOL drifts
	Fast-ion transport by energetic particle modes
	Core/edge coupling
	Extended MHD
	Intrinsic rotation
	Error field screening effects
	ELM losses
	Pedestal width

Opportunities for Training Ph.D. Students. Universities which participate in the DIII-D program use the facility as a training ground for graduate students. Fifty-three students have performed research at the GA fusion facility, which has led to the award of an advanced degree. Some students may be full time at the DIII-D site designing, installing, and using diagnostic systems or analyzing DIII-D data (Fig. 11-3). All students are provided opportunities to present their work at science meetings. Others may work at their university writing analysis codes or developing theories, explaining plasma phenomena observed on DIII-D. Table 11-9 shows a list of past and present Ph.D. candidate students at DIII-D.

Fig. 11-3. DIII-D hosts many graduate students, providing them with a wide range of research experience.

Postdoctoral and recent Ph.D. graduates participate in the DIII-D experimental and theoretical fusion research. Scientists holding postdoctoral fellowships at universities have also furthered their scientific training at the GA fusion facility. Table 11-10 summarizes past and present participants.

General Atomics is also actively involved in the National Undergraduate Fellowship Program in Plasma Physics and Fusion Engineering which is administered by the Science Education program of the Princeton Plasma Physics Laboratory. This program provides about 25 summer internships to outstanding undergraduate students. Typically, DIII-D hosts four to six students each summer. From the results of student evaluations, it is clear that DIII-D has provided these students with a valuable learning experience that has proved to be a strong motivator for their pursuit of advanced degrees in fusion science. The program is a valuable tool for recruiting sharp new graduate students who may one day lead research on ITER.

Relationship with the Transport Task Force. The long-term goal of the U.S Transport Task Force (TTF) is to develop a fundamental understanding of plasma transport in magnetic confinement devices so that transport in these machines can be predicted and controlled. The TTF seeks this understanding through a multi-faceted approach. Plasma transport is characterized through power balance studies and through modulated transport studies. A major emphasis is placed on the study and understanding of the underlying plasma turbulence, which is widely accepted as being the source of the transport. These turbulence studies are made possible by dramatic improvements in the ability to measure turbulence properties. The most complete understanding of plasma transport is obtained by an integration of theory and experiment, which is often accomplished by comparisons of experimental data with the predictions of theory-based simulation codes. These comparisons are often done for macroscopic phenomena, such as temperature profiles. However, there is an increasing effort to compare turbulence measurements on microscopic scales with the theoretical predictions, a comparison that is considered the most fundamental test of theory that can now be done.

The work of the TTF requires the integration of many individuals, groups and machines, spread among a large number of institutions. The main vehicle for this integration is the annual meeting of the TTF at which participants present results, identify issues and discuss future plans. The primary way that the TTF affects research in the U.S. fusion community is that individuals return to their home institutions and implement ideas gained from the TTF interactions. For work that requires experimental time, researchers propose their ideas and compete with other interests to obtain machine time and to implement their ideas. This process leads to a significant alignment of the science programs on several machines and the goals of the various TTF working groups.

Table 11-9
Past and Present Graduate Students at DIII-D

Researcher	Affiliation	Topic
Students		
N. Antoniuk-Pablant*	UCSD	B-Stark diagnostic
Q. Boney	Hampton University	Divertor impurity diagnostic
E. Carolipio	UC Irvine	TAE mode studies
S. Coda	MIT	CO_2 phase image interferometer
K. Comer	U. Wisconsin	MHD studies
D. Content	Johns Hopkins	Bolometers and visible bremsstrahlung
R. Deranian	U. Wales	Plasma control
M. Donales	Hampton University	Divertor impurity diagnostic
J. Dorris*	MIT	Phase contrast imaging
H. Duong	UC Irvine	Fast ion bursts
D. Elder*	U. Toronto	OEDGE modeling of C^{13} experiments
C. Estrada-Mila	UCSD	Turbulent transport simulations
D. Finkenthal	UC Berkeley	He transport
J. Fitzpatrick	UC Berkeley	TAE mode analysis
C. Fransson	Chalmers U.	Plasma control
H. Frerichs*	FZ Julich	3D fluid modeling of RMP
Z. Friis*	Georgia Tech.	Thermal instabilities
R. Gatto	UC Berkeley	Heat pinch modeling
S. Harrison	U. Wisconsin	Plasma surface interactions
J. Hillscheim*	UCLA	Multi-frequency Doppler reflectometry
W. Howl	UCSD	MHD reconstruction
D. Hua	UC Berkeley	ITG modes and energy confinements
M. Jakubowski	U. Wisconsin	Beam emission diagnostics
A. James*	UCSD	γ-ray spectroscopy of disruptions
S. Janz	U. Maryland	ECE diagnostic bolometers
O. Katsuro-Hopkins	Columbia U.	RWM feedback control modeling
F. Kelly	Georgia Tech	Radiation modeling
K.W. Kim	UCLA	Fast density profiles reflectometry
S. Kruger	U. Wisconsin	Flow shear effects on MHD
M. Lanctot*	Columbia U.	RWM feedback control
T. Le Hecka	UCLA	Microwave reflectometry
J.H. Lee	UCLA	Fast wave studies
B. Leslie	U. Wisconsin	Beam emission spectroscopy
Y. Luo	UC Irvine	Beam ion studies
A. McLean*	U. Toronto	Plasma surface interactions
B. Modi	UC Berkeley	Turbulence modeling
S. Mordijck*	UCSD	2D modeling of edge transport
Y. Mu*	U. Toronto	Hydrocarbon fragmentation modeling
C. Muscatello	UC Irvine	Fast ion transport
E. Nardon*	CEA-Cadarache	ELM control by stochastic fields
Q. Nguyen	UC Berkeley	UEDGE development
Y-S. Park*	Seoul National U.	NTM detection and control
M. Perry	Johns Hopkins	Impurity transport
D. Pretty	Australia National U	Stochastic edge mag. field studies
Chuang Ren	U. Wisconsin	Plasma rotation
C. Rettig	UCLA	Microturbulence studies
R. Rubilar*	Georgia Tech	Radiation modeling
G. Sager	U. Illinois	Data analysis program
M. Shafer*	U. Wisconsin	Beam emission spectroscopy
D. Schlossberg*	U. Wisconsin	Beam emission spectroscopy
R. Stockdale	Princeton U.	Perturbative transport experiments
W. Wang	UC Irvine	Neoclassical transport studies
G. Watson	UC Irvine	ICRF Studies
A. White*	UCLA	T_e fluctuation diagnostic
B. Zaniol	U. Padova	Impurity ion flow in divertor
A. Zwicker	Johns Hopkins	Multi-layer mirror spectrometer

*Indicates current graduate student (all or part of FY08).

Table 11-10
Past and Present Post-Doctoral Fellows at DIII-D

Researcher	Affiliation	Topic
Post Doctorates		
M. Austin	U. Maryland	ECE diagnostics
E. Belli*	ORISE	Edge gyrokinetic simulations
D. Brennan	ORISE	MHD
A. Brizard	UC Berkeley	Transport analysis
A. Cole	U. Wisconsin	Nonresonant field error effects
N. Eidietis*	ORISE	Plasma control
D. Ernst	Princeton	Transport studies
C. Fenzi	France/U. Wisconsin	Beam emission spectroscopy
A. Garofalo	Columbia U.	Wall stabilization
G. Garstka	U. Maryland	ECE diagnostics
T. Gianakon	U. Wisconsin	MHD theory and modeling
D. Gray	UCSD	Disruption and coherent mode studies
M. Groth	LLNL	Boundary physics
D. Gupta	U. Wisconsin	Beam emission spectroscopy
E. Hollmann	UCSD	Disruption and coherent mode studies
C. Holcomb	LLNL	MSE diagnostic
C. Holland	UCSD	Turbulence studies
B. Hudson	ORISE	Edge current measurement
Y. Jeon*	ORISE	Integrated modeling
I. Joseph*	UCSD/LLNL	RMP ELM control
J. Kinsey	Lehigh	Transport modeling
M. Kissick	U. Wisconsin	Heat pulse propagation
S. Kruger	U. Wisconsin	MHD studies
K. Kupfer	ORISE	RF current drive
T. Kurki-Suonio	UC Berkeley	Transport analysis
R. Lehmer	UCSD	Divertor
R. Maingi	ORISE	Divertor physics
G. McKee	ORNL	Divertor spectroscopy
S. Muller*	USCD	Edge turbulence
P. O'Shea	MIT	Phase contrast imaging
D. Ponce	ORISE	Thomson scattering
J.M. Park	ORNL	Integrated modeling
H. Reimerdes	Columbia	Resistive wall mode stabilization
C. Rost	MIT	Phase contrast imaging
D. Rudakov	UCSD	Edge turbulence and transport studies
H. Schmitz*	FZ Julich	Divertor DiMES camera
W. Solomon	PPPL	CER diagnostics
A. Sontag*	ORNL	MHD studies
J. Squire	ORISE	X–ray diagnostic
Z. Unterberg*	ORNL	Edge spectroscopy
M. VanZeeland	ORISE	CO2 interfermeter
F. Volpe*	ORISE	Structure of magnetic islands
M. Wade	ORISE	Helium transport
G. Wang	UCLA	Transport studies/diagnostics
Dennis Whyte	CCFM/Canada/UCSD	Divertor physics
L. Zeng	UCLA	Transport studies/diagnostics

*Indicates current Post-Doctoral Fellow (all or part of 2008).

Table 11-11
DIII-D Collaborations Related to Fusion Science Research 2008

Topic	Collaborating Institution	Key Collaborator	DIII–D Contact
Aspect ratio scaling	PPPL	S. Kaye	C. Petty
Atomic physics modeling	U. Strathclyde	M. O'Mullane	T. Evans
Beam emission spectroscopy; transport	U. Wisconsin	G. McKee*, D. Schlossberg, M. Shafer	C. Petty
Beam emission spectroscopy; analysis	UCSD	G. Tynan, C. Holland*	G. McKee
CER diagnostics	NFRC	W.-H. Ko	K. Burrell
Dimensionless scaling	JET	D. McDonald	C. Petty
	MIT	M. Greenwald, S. Wolfe	C. Petty
	UKAEA	M. Valovic	C. Petty
DiMES	USCD	D. Rudakov*	C. Wong
Divertor physics	LLNL	S. Allen*	C. Petty
	MIT	B. Lipschultz	C. Petty
Divertor spectroscopy	ORNL	D. Colchin, R. Isler, Z. Unterberg*	W. West
ECE diagnostic	U. Maryland	R. Ellis	C. Petty
	U. Texas	M. Austin*	C. Petty
Edge current density	ORISE	B. Hudson*	C. Petty
Edge modeling	UCSD	A. Pigarov	W. West
	UCSD	S. Mordijck	T. Evans
Edge turbulence	UCSD	J.A. Boedo,* S. Muller	C. Petty
Fast-ion physics	UC Irvine	W. Heidbrink, Y. Luo, Y. Zhu, C. Muscatello	C. Petty
	PPPL	R. Nazikian	C. Petty
	PPPL	N. Gorelenkov, G. Kramer, G. Fu	A. Turnbull
FIR scattering, hi-k backscattering	UCLA	T. Rhodes*, L. Zeng*, X. Nguyen*, G. Wang*, L. Schmitz, A. White,* J. Hillscheim	J. DeBoo
Fluctuation diagnostics	U. Tokyo	S. Kado, T. Oishi	G. McKee
ICRF fast ions	U.C. Irvine	W. Heidbrink	R. Pinsker
L-H transition physics	UCSD	D. Rudakov*, R. Moyer*	K. Burrell
	U. Wisconsin	G. McKee,* D. Schlossberg	K. Burrell
Leader of UCLA effort; member of DIII-D Executive Committee	UCLA	W.A. Peebles	T. Taylor
Midplane and X-point Langmuir probe	UCSD	J. Boedo,* R. Moyer,* D. Rudakov*	W. West
	SNLA	J. Watkins*	W. West
Neoclassical tearing mode physics	York University	H. Wilson	R. La Haye
	CRPP-Lausanne	O. Sauter	R. La Haye
	U. Tulsa	D. Brennan	R. La Haye
	Tech-X Corp.	S. Kruger	R. La Haye
Neutral effect on L-H transition	ORNL	L. Owen	R. Groebner
Phase contrast imaging	MIT	C. Rost*, J. Dorris*	K. Burrell
Plasma ExB shear	Ukraine	M. Shats	K. Burrell
Plasma flows	UCSD	J. Boedo*	W. West
Plasma rotation	PPPL	W.M. Solomon*	K. Burrell
	U. Wisconsin	J. Callen, A. Cole	R. La Haye
Reflectometry L-H and core barrier physics	UCLA	E. Doyle*, G. Wang*	J. DeBoo
Sawtooth physics	ORNL	E. Lazarus*	C. Petty
	CRPP-Lausanne	O. Sauter	R. Pinsker
SOL transport	UCSD	S. Krasheninnikov	W. West
Theory of transport barrier formation and fluctuation suppression	UCSD	P. Diamond	Many people

*On-site personnel.

The DIII-D program has provided strong support for the TTF since its inception. This support includes the presentation and supply of experimental data, including profile and fluctuation measurements, regarding core and edge transport and energetic particle physics. In recent years, an average of about ten DIII-D scientists has attended the yearly TTF meeting. This participation has helped to develop or strengthen productive collaborations with other institutions in terms of experiments that have been proposed or experiment-theory comparisons that have been undertaken in DIII-D. DIII-D intends to maintain its participation in the TTF during the next five years.

12. INTERNATIONAL COLLABORATIONS

12.1. INTRODUCTION

International collaboration is a key component of the DIII-D research program. The DIII-D program is enhanced in multiple areas through its international collaborative activities. These areas include: (a) the validation of fusion plasma physics through joint experiments; (b) testing and advancing the capabilities of plasma control techniques through applications on multiple machines; (c) broadening the knowledge base and experience of the research staff aimed towards further innovation in fusion physics and, primarily, towards advancing the DIII-D research program. The detailed collaborative exchanges, which are carried out with our international partners, provide the building blocks for advancing fusion energy science worldwide, particularly in preparing for and supporting international next step experiments, such as ITER. The range of scientific exchanges is very broad and covers a large number of plasma physics topics. This can be clearly seen in Table 12-1, which shows the recent exchanges for the 2006–2007 period.

The DIII-D international collaboration program is actively engaged with multiple international partners to enhance the world-leading expertise and capabilities within the DIII-D research program as well as sharing these capabilities with other fusion facilities. Figure 12-1 shows the interaction between the DIII-D research program and the major fusion facilities. The main premise of the international collaboration program at DIII-D is to advance the science of fusion plasma physics through interaction with its international partners. This includes continuing to perform pioneering work and increasing the relevance and validity of the DIII-D research program through application of the results on other fusion devices. As part of these collaborative efforts, the DIII-D program is heavily involved with the ITER organization through: (a) addressing ITER high priority research needs by the planning and execution of joint experiments with other fusion facilities proposed through the International Tokamak Physics Activity (ITPA), and; (b) work on advisory committees, such as the Science and Technical Advisory Committee (STAC) and the Technical Advisory Group (TAG).

The present and future collaborative activities planned in the DIII-D international collaboration program are detailed in this section and are highlighted in Table 12-2. The list of planned collaborative activities covers a broad range of topics, of which prime examples are: personnel exchanges to prepare and perform joint experiments; the development of software and hardware components for specific applications, such as plasma control and auxiliary heating systems; the development of remote participation capabilities; the development of tools for data analysis and modeling; work on technical and advisory committees. These collaborations will continue to expand on activities with long established fusion facilities, such as EFDA-JET, JT-60U and Axisymmetric Divertor Experiment Upgrade (ASDEX-U).

Table 12-1
A Broad Range of Scientific Personnel Exchanges Enhance International Collaborations and Joint Experiments

2006–2007

To DIII-D

Plasma Control System Development
S.-H. Han (KSTAR)
S.H. Seo (KSTAR)
B. Xiao (EAST)
Q. Yuan (EAST)
J. Qian (EAST)

NTM Stabilization
R, Buttery (UKAEA)
F. Volpe (MPI Gesellschaft)
A. Isayama (JAEA)

RWM Stabilization
M. Takechi (JAEA)
G. Matsunaga (JAEA)

Pedestal and QH-mode Studies
C. Maggi (IPP-Garching)
Y. Kamada (JAEA)
Y. Sakamoto (JAEA)

ELM Studies
N. Oyama (JAEA)

AT Scenarios
T. Suzuki (JAEA)
S. Ide (JAEA)
Y. Sakamoto (JAEA)

Li Beam Development
K. Kamiya (JAEA)
A. Kajima (JAEA)

Resonant Magnetic Perturbation Studies
K.-H. Finken (FZ-Julich)
M. Jakabowski (FZ-Julich)
M. Lehnen (FZ-Julich)
B. Unterbert (FZ-Julich)
O. Schmitz (FZ-Julich)
H. Frerichs (FZ-Julich)
D. Reiter (FZ-Julich)
D. Harting (FZ-Julich)
D. Schega (FZ-Julich)
E. Nardon (CEA-Cadarache)
M. Becoulet (CEA-Cadarache)
V. Paraail (UKAEA)
J. Lonnroth (U. Finland)
R. Koslowski (FZ-Julich/JET)
Y. Liang (FZ-Julich)
S. Pinches (IPP Garching/JET)
R. Buttery (UKAEA)

Transport Physics
L. Vermare (IPP-Garching)

Boundary Physics
S. Brezinsek (IPP-Garching)

From DIII-D

Boundary Physics (ASDEX-U)
M. Groth (LLNL)

ITB Physics and Real Time Control (JET)
P. Gohil (GA)

NTM Studies (JET)
R. La Haye (GA)

QH-mode Plasmas (JT-60U)
P. Gohil (GA)
L. Lao (GA)

AT Scenarios (JET)
T. Luce (GA)
J. Ferron (GA)
M. Murakami (remote)

Ergodic Divertor Studies (TEXTOR)
T. Evans (GA)

Remote Participation on ELM Control Studies (JET)
T. Evans (GA)
P. Gohil (GA)
R. Moyer (UCSD)
T. Osborne (GA)

Pedestal (JET)
T. Leonard (GA)
T. Osborne (GA)

Divertor Hardware Development (EAST)
M. Schaffer (GA)

Neutral Beam Development (EAST)
R. Callis (GA)

Plasma Control System Development (EAST)
D. Humphreys (GA)
M. Walker (GA)
J. Leuer (GA)
G. Jackson (GA)
D. Piglowski (GA)
B. Penaflor (GA)
A. Hyatt (GA)

Plasma Control System Development (KSTAR)
M. Walker (GA)
B. Penaflor (GA)

ITER COD AC Design (ITER Cadarache)
M. Walker (GA)
D. Humphreys (GA)

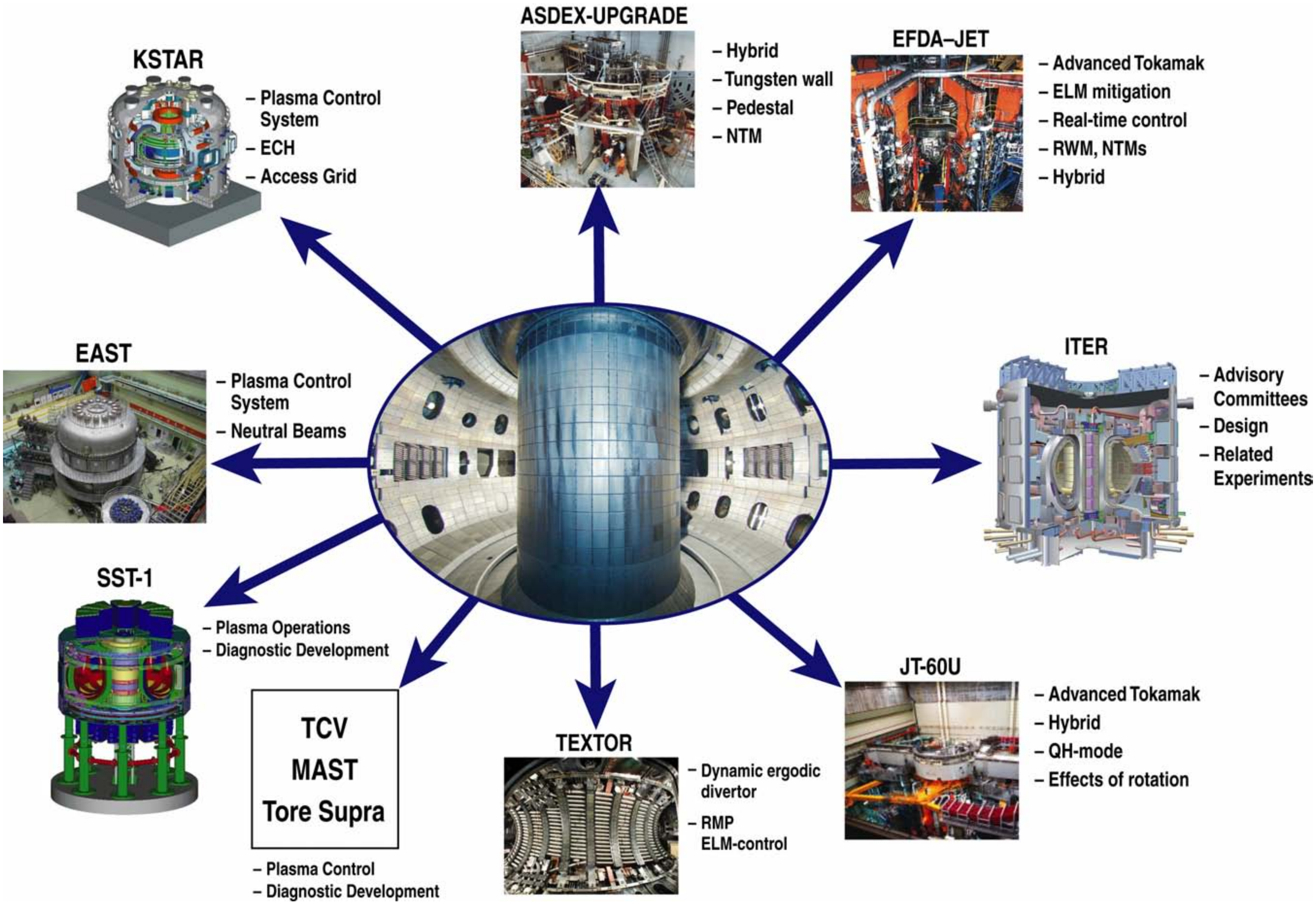

Fig. 12-1. The DIII-D research program is actively collaborating with numerous fusion devices world wide.

Table 12-2
Collaborative Activities Described in this Section

Section	Collaborative Activity	Status and Plans
12.2	**International tokamak physics activity (ITPA)**	• **Active involvement of DIII-D personnel in ITPA topical groups** — Propose and execute joint experiments with other fusion facilities — Perform data analysis and prepare reports of scientific results
12.3	**Collaboration with other tokamak facilities**	• **Scientific personnel exchanges for performing joint experiments data analysis and modeling** • **Hardware and diagnostic development of prime areas of research**
	⇒ JT-60U	— Joint experiments in resistive wall mode (RWM) instabilities, neoclassic tearing modes (NTMs), quiescent H-mode (QH-mode) and hybrid plasmas, pedestal and ELM characteristics, momentum transport studies, high performance steady state plasmas — Diagnostic development — Edge current profile measurements
	⇒ EFDA-JET	— Joint experiments on high performance steady state plasmas, NTM and RWM studies, hybrid plasma development, realtime profile control and ITB studies
	⇒ ASDEX-U	— Joint experiments on NTM stabilization, pedestal studies, hybrid plasma development, QH-mode plasma and divertor/scrapeoff layer (SOL) studies
	⇒ TEXTOR	— Joint experiments, modeling and code development on resonant magnetic perturbative (RMP) techniques for ELM control
	⇒ MAST/TCV/ Tore Supra	— Develop plasma control systems — Diagnostic development
12.4	**International Cooperative Agreements**	• **Develop framework for carrying out collaborative activities**
12.5	**Web access to the DIII-D facility**	• **Develop and enhance capabilities for interaction with the DIII-D research program through web-based tools**

12.2. INTERNATIONAL TOKAMAK PHYSICS ACTIVITY

The DIII-D research program is actively engaged with the workings and plans of the International Tokamak Physics Activity (ITPA). The ITPA organization is presently supported through the International Fusion Research Council (IFRC) and is a joint activity of fusion programs in U.S., EU, Japan, Korea, China, India and Russia. The ITPA aims to provide cooperation on the tokamak physics R&D activities in order to develop the physics basis for burning tokamak plasmas. The internationally coordinated research activities within the ITPA are separated into topical physics groups and are performed on a voluntary basis. The purpose of these topical groups is to: (a) propose joint experiments to advance the understanding of fusion plasma physics; (b) assimilate data from these experiments and coordinate the analysis and prepare reports on the results; (c) organize, manage and update qualified databases in the different areas of fusion physics; (d) develop theoretical models and simulation codes to explain and reproduce experimental results; (e) integrate the R&D results towards improving the plasma performance and developing the operational scenarios for burning plasmas; (f) identify and resolve the key diagnostics issues associated with the control and analysis of burning plasma experiments.

The DIII-D research program is closely involved with the ITPA and makes strong contributions on many ITPA tasks, particularly with regard to the proposal and execution of ITPA joint experiments. For the DIII-D experimental program for 2007, out of the 53 experiments performed, roughly 39 were allocated for ITPA related experiments. Table 12-3 shows the breakdown of the experiments according to the research topics. The close involvement with the ITPA is also reflected in the U.S. membership of the ITPA Topical groups, which is shown in Table 12-4. The names highlighted in red are closely involved with the DIII-D program and reflect the strong contribution of the DIII-D program to this important international activity.

Table 12-3
General and ITPA Related Experiments Performed on DIII-D in 2007

Thrust or Topical Science Area	Total Experiments	ITPA Experiments
Thrust IT-1: ELM control for ITER	7	6
Thrust IT-2: ITER hybrid scenarios	5	4
Thrust IT-3: NTM central for ITER	3	2
Thrust IT-4: RWM control for ITER	7	5
Thrust AT-1: Advanced scenario development	9	5
Thrust SC-1: Pedestal width physics	1	1
Confinement and transport	6	6
Stability	7	5
Boundary	6	5
Heating and current drive	2	0
	53	39

Table 12-4
U.S. Members of the ITPA Topical Groups

Coordinating Committee	E. Oktay R. Stambaugh N. Sauthoff	Steady State Operations	P. Bonoli C. Kessel T. Luce R. Prater M. Murakami
Transport Physics	E. Doyle P. Gohil J. Rice J. Kinsey E. Synakowski	MHD and Disruptions	E. Strait R. Granetz J. Menard E. Lazarus G. Navratil W. Heidbrink
Scrapeoff Layer and Divertor Physics	B. Lipschultz P. Stangeby S. Krashennikov M. Fenstermacher D. Whyte	Confinement Database and Modeling	W. Houlberg S. Kaye J. DeBoo J. Snipes R. Budny
Pedestal and Edge Physics	T. Leonard P. Guzdar A. Hubbard M. Wade T. Rognlien	Diagnostics	D. Johnson R. Boivin G. Wurden G. McKee A. Peebles

The DIII-D program will continue to place a high emphasis on performing ITPA tasks within its research program and further adapt to the high priority tasks, as well as the evolution of the ITPA organization itself, in light of the greater requirements and influence of the ITER organization.

12.3. COLLABORATION WITH OTHER TOKAMAK FACILITIES

12.3.1. JAEA/JT-60U

The collaboration with the Japan Atomic Energy Agency (JAEA), formerly known as the Japan Atomic Energy Research Institute (JAERI), has been the longest and most extensive in the history of the DIII-D research program. The collaborative agreement started in 1978 during the first year of operations on Doublet III and over nearly the last 30 years has provided a source of both financial contributions and scientific manpower that has significantly enhanced the DIII-D research program. With the advent of the JT-60U fusion facility, the areas of collaboration with DIII-D were greatly increased, particularly in the fields of Advanced Tokamak (AT) science and steady state, integrated performance optimization. Presently, collaborative exchanges between DIII-D and JT-60U are performed under the framework of the JAEA/DOE DIII-D agreement and the IEA Large Tokamak Implementing agreement. Over the course of the last five years, the major areas of collaboration have included Advanced Tokamak research, the development of ITER hybrid scenarios, the suppression of Neoclassical Tearing Mode (NTM) and Resistive Wall Mode instabilities (RWM), vessel activation studies, particle control, edge stability physics and edge current density measurements, transport barrier studies, and ECH, NBI and advanced control technology between DIII-D and JT-60U. Of particular importance have been the collaborative efforts on QH-mode plasmas in which a DIII-D team was instrumental in producing QH-mode plasmas on JT-60U and which led to many joint experiments been performed at both DIII-D and JT-60U. These efforts have increased the feasibility of QH-mode plasmas, which were first developed at DIII-D, as a viable operating scenario for next step fusion devices.

For future collaborative work, the QH-mode studies also highlight the capabilities in both DIII-D and JT-60U to perform experiments in which dramatic changes in the beam torque (i.e. co-, counter-, balanced) can be applied to affect the plasma rotation and the plasma behavior. Using these similar capabilities under different plasma conditions and parameters in DIII-D and JT-60U will advance collaborative studies in the following areas: (a) studies of resistive wall modes — this collaboration is especially important in that it feeds directly into the design for JT-60SA; (b) studies of neoclassical tearing modes; (c) pedestal and H-mode studies (including QH-mode) examining the effects of plasma rotation on the threshold power, pedestal characteristics and momentum transport; (d) characteristics of grassy ELMs with respect to plasma rotation; (e) study of the effects of plasma rotation on the momentum transport in L- and H-mode. Other areas of mutual interest are high performance steady state scenario development and diagnostic development, especially for measurements of the edge current density. All these studies aim to establish the commonality of the physics between these machines in an effort to provide greater confidence in extrapolation to the next step fusion devices.

12.3.2. EFDA-JET

The administration of the JET fusion device is organized under the European Fusion Development Agreement (EFDA) and is operated primarily as a user facility for member associations within EFDA. As such, experiments to be performed on JET are first proposed through the various association institutes and then prepared and executed by specific task forces comprised of visiting scientists from the association institutes. The main collaborative activities between DIII-D and JET have been centered on performing joint experiments at both facilities although collaborations have also been extended to diagnostic development and modeling efforts. The large majority of joint experiments are proposed through the International Tokamak Physics Activity (ITPA) organization and the collaborative exchanges between DIII-D and JET are primarily performed under the framework of the IEA Large Tokamak Implementing Agreement.

The main areas of interaction between DIII-D and JET have been: (a) development of high performance, steady state operating scenarios; (b) NTM and RWM studies; (c) hybrid plasma development; (d) ELM mitigation and pedestal studies; (e) real time profile control and ITB studies. The DIII-D research program benefits from these collaborations through validation of plasma physics, operational scenarios and control techniques developed at DIII-D, especially since JET is the largest fusion device in the world and offers collaborative opportunities for multi-machine comparisons such as size scaling with similar plasma shapes and aspect ratios. These collaborative efforts will continue in coming years with the addition of studies in the area of plasma facing components since JET plans on installing the ITER-like wall (Beryllium first wall with a tungsten/carbon divertor) in 2009.

12.3.3. ASDEX-U

The main areas of collaboration between DIII-D and ASDEX-U have recently been in the areas of NTM stabilization by ECCD, pedestal studies, hybrid plasma development, QH-mode plasma studies and divertor/scrapeoff layer (SOL) studies. The latter has become more prominent recently as ASDEX-U has installed a tungsten clad first wall and divertor hardware. The walls are comprised of carbon tiles with tungsten coatings of typical thicknesses of 1–2 μm with the divertor region having tungsten coatings with 4 μm thickness. A key issue will be how the plasma performance and operating scenarios (such as hybrid plasmas) are affected as a result of the new PFCs. Comparative studies between DIII-D (an all carbon PFC device) and ASDEX-U in the above named research areas will endeavor to resolve the virtues of the various PFC materials used. The DIII-D research program has benefited particularly by the interaction with visiting scientists from ASDEX-U in the areas of hybrid plasmas, pedestal physics and NTM stabilization and these collaborations will be continued and further strengthened.

12.3.4. TEXTOR

The primary collaboration with TEXTOR involves the development of numerical models needed to scale resonant magnetic perturbation (RMP) ELM and pedestal control techniques, originated at DIII-D, to burning plasmas. The details of the collaboration are described in a collaboration agreement entitled: "Joint Collaboration agreement on Resonant Magnetic Perturbation Research between DIII-D and TEXTOR in the frame of the IEA Implementing Agreement on Plasma Wall Interaction in TEXTOR". During the initial period covered under this agreement, DIII-D scientists traveled to TEXTOR in order to participate in RMP experiments using the Dynamic Ergodic Divertor (DED) coils and TEXTOR scientists have traveled to DIII-D in order to participate in RMP ELM control experiments as well as to work on various aspect of the modeling codes. These exchanges included 2 FTE weeks of DIII-D staff at TEXTOR and 33 FTE weeks of TEXTOR staff at DIII-D in FY07 as well as the exchange of specialized RMP diagnostic hardware needed to validate the numerical models. A near term goal of this collaborations is to validate a poloidally diverted version of the 3D fluid transport code EMC3 using DIII-D experimental data.

Over the next five years, we will develop and validate numerical models needed to describe the 3D magnetic topology of the pedestal plasma in DIII-D including effects due to screening of the external magnetic perturbation in rapidly rotating, high beta, plasmas. This involves coupling GA's TRIP3D field line integration code to the Kikuchi resonant field screening two-fluid code developed at TEXTOR. Results from the TRIP3D-Kikuchi code will be coupled to the DIII-D version of the EMC3-ERIENE code in order to calculate heat, particle and momentum transport. Results from these codes will be compared to DIII-D RMP experiments while stochastic transport studies in TEXTOR will be used to develop basic physics models needed in these codes. In addition to the stochastic transport modeling codes, a longer term goal of this collaboration is to develop a model describing the self-consistent behavior of the electric field in across the pedestal plasmas during RMP pulses and to couple a poloidally diverted symplectic field line mapping code, the ATLAS code, developed at TEXTOR to the DIII-D resonant field spectrum. The ATLAS code will be used to model field line diffusion coefficients and the fractal structure of the field produced by the RMP in DIII-D.

12.3.5. MAST/TCV/Tore Supra

The DIII-D research program is actively engaged with many other fusion programs around the world, which cannot all be mentioned in detail here. However, of these the most notable are the interactions with MAST, TCV and Tore Supra which involve the work on plasma control systems with MAST and Tore Supra and the development of the charge exchange diagnostic system at TCV.

12.4. INTERNATIONAL COOPERATIVE AGREEMENTS

The collaborations between DIII-D and its international partners are carried out within the context of international cooperative agreements, which provide for the legal and administrative framework for performing the scientific research. These agreements range from those based on specific topics or machines (e.g. the IEA Large Tokamak implementing agreement, the IEA implementing agreement on Poloidal Divertor tokamaks, the IEA plasma-wall interaction in TEXTOR implementing agreement, bilaterals) and those based on scientific exchanges between specific countries (e.g. bilaterals). For example, the IEA Large Tokamak Implementing Agreement involves collaboration between the JT-60U, EFDA-JET, DIII-D, C-Mod and NSTX. The IEA Implementing Agreement on "a co-operative program for the investigation of toroidal physics in, and plasma technologies of, tokamaks with poloidal field divertors" involves collaboration between ASDEX-U, DIII-D, C-Mod, NSTX and KSTAR (with the proposed addition of EAST and SST-1). Also, scientific exchanges with work performed by JT-60U staff at the DIII-D facility are also carried out within the framework of the JAEA/DOE DIII-D agreement. Proposals are underway, for example, to consolidate The IEA Large Tokamak, Poloidal Divertor and TEXTOR agreements into one encompassing agreement. The DIII-D program is participating in this process of consolidation and will be an active partner in the resultant agreement.

The DIII-D research program will continue its involvement with its present agreements as well as being open and prepared to enter into new agreements that enhance fusion research worldwide, such as with ITER, KSTAR, JET, JT-60U, etc. This is a fundamental premise in the development of our remote collaboration capabilities for interaction with these other fusion facilities.

12.5. WEB ACCESS TO THE DIII-D FACILITY

The web site of the DIII-D National Fusion Facility (http://fusion.gat.com/) is a critical tool for successfully conducting the program's mission by a geographically dispersed research team. Traditionally this tool has been used for communication and as an historical information archive. From the daily experimental plan to the publications repository, the web has allowed for rapid worldwide communication. However, as with all information technology, the evolution of web-related technology has been rapid and the DIII-D web site will take advantage of new capabilities to better serve the scientific community and to improve the interaction of international collaborators with DIII-D.

To allow for easier web authoring by the DIII-D National Team, the DIII-D internal web site is being transitioned to a Wiki that allows editing by the scientific staff. This capability allows more scientists to add web content and to do so faster by eliminating the need to learn HTML and by eliminating the bottleneck of waiting for the web master to make required changes. This transition will continue and more scientists will be given accounts and educated on this new capability. To ease access, web-related security such as viewing access, Wiki editing, Bugzilla, and web-based applications will be unified to present a single security interface to the international DIII-D team.

Given the ubiquity of web browser clients on all operating systems, more client software will be transitioned away from custom applications to web-based systems. Moving to a web interface provides easier access, is more reliable, and can allow for usage by a DIII-D team member located worldwide including our international colleagues. An example of such a change is the Session Leader checklist that will allow for some control of between-shot data analysis on the STAR cluster. Details on security and web-based applications are provided in Chapter 9.

13. DIII-D GOVERNANCE

Effective governance is an essential component of the DIII-D National Fusion Program, both for efficient management and for supporting a world-class multi-institutional fusion energy research program. Governance includes defining overall roles and responsibilities, establishing an open program planning process that nurtures efficiency and innovation, coordinating research activities among the partners, and reporting and publication of results. Professional development is an important consideration here, since the strength of the DIII-D program resides in the motivated creativeness of the participants.

The processes outlined here represent snapshot of what is a dynamic organization that began with the Doublet-III project, which featured a major collaboration with JAERI in Japan. DIII-D participants provide continuous feedback and suggestions for improvement in what is a very open and collaborative environment. The present way of doing business builds upon this past experience and reflects the broad-based input provided by team members.

13.1. ROLES AND RESPONSIBILITIES

General Atomics is the host institution for the DIII-D National Fusion Facility. The Director of the DIII-D National Fusion Program is an employee of General Atomics and is responsible for safe operation of the facility and for oversight of the DIII-D National Fusion Program. The present Deputy Director is an LLNL employee and a member of the Livermore-GA collaboration team The Director of the DIII-D Experimental Science Division (formerly a member of the ORNL-GA collaboration team) is responsible for the execution of the DIII-D research program.

The DIII-D Team. The DIII-D Program is an open program with extensive national and international collaborations. The DIII-D National Team consists of collaborations among 7 national laboratories, 26 universities, and 15 industrial companies. International collaborations are conducted with 29 national laboratories and 15 universities. Presently, the near full time scientific staff consists of approximately 40% General Atomics scientists and 60% collaborating scientists.

DIII-D Executive Committee (DEC). The DIII-D Executive Committee meets monthly to advise the DIII-D Director on a broad range of programmatic issues such as long-range program planning. The DEC also addresses institutional issues related to managing the DIII-D team. DEC membership consists of 14 senior representatives from General Atomics and the major collaborators, including, Princeton Plasma Physics Laboratory, Lawrence Livermore National Laboratory, Oak Ridge National Laboratory, Columbia University, University of Texas, the University of California at Los Angeles, and the University of California at San Diego.

DIII-D Program Advisory Committee (PAC). The DIII-D Program Advisory Committee is composed of 15 experts in the field not directly involved in the DIII-D Program. It reports to the Vice-President for Magnetic Fusion Energy. The PAC meets openly at least once per year, responding to specific charges, which generally seek their comments on the Experimental Plan for the coming year and other issues prominent at the time. Their report is given to the Vice-President for Magnetic Fusion Energy but is regarded as a public document.

Research Council. The Research Council (RC) is a large multi-institutional advisory group (29 members in FY08) chaired by the DIII-D Deputy Program Director. It is composed of scientists at all levels representing the various research areas, program management structures, and major collaborators. Its principal role is to plan the experimental program each year. Table 13-1 lists the members of the 2008 Research Council.

Table 13-1
FY08 Research Council Members and Affiliations

Chair: David Hill (LLNL)	**Vice Chair:** Chuck Greenfield (GA)	
Experimental Coordinator: Chuck Greenfield (GA)	**Assist. Experimental Coordinator:** Max Fenstermacher (LLNL)	
Keith Burrell (GA)	George McKee (U Wisc)	Mickey Wade (GA)
Edward Doyle (UCLA)	Masanori Murakami (ORNL)	Phil West (GA)
John Ferron (GA)	Peter Petersen (GA)	Mike Van Zeeland (GA)
Mathias Groth (LLNL)	Craig Petty (GA)	
Eric Hollmann (UCSD)	Ron Prater (GA)	
Dave Humphreys (GA)	Holger Reimerdes (Columbia)	Ron Stambaugh[(a)]
Tom Jernigan (ORNL)	Dimitry Rudakov (UCSD)	Vincent Chan[(a)]
Arnie Kellman (GA)	Phil Snyder (GA Theory)	Tony Taylor[(a)]
Lang Lao (GA Theory)	Wayne Solomon (PPPL)	
Tim Luce (GA)	Ted Strait (GA)	

[(a)]Ex Officio

Experimental Science Division and Physics Groups. The Experimental Science Division of the DIII-D program is composed of five Physics Groups:

1. Integrated Steady State
2. ITER Physics
3. Fusion Science
4. Integrated Modeling
5. Plasma Operation and Control

The scientific content and expertise within these groups is quite broad (topically and institutionally), with activities being differentiated by particular application for all the groups except Fusion Science. The Fusion Science Group (which is the largest group) maintains primary stewardship of five topical science areas: Stability, Transport, Heating and Current Drive, Plasma Boundary, and Energetic Particles. All scientists and students participating in on-site research are assigned to one of these Physics Groups.

The Experimental Science Division is responsible for executing the overall DIII-D Research Plan. Specific research activities are organized and executed by **Topical Working Groups** and **Task Forces**. Topical Working Groups are closely aligned with the organizational structure of the Physics Groups within the Experimental Science Division and may be more enduring than Task Forces, which report directly to the Division Director and address more near-term high priority research that is cross-cutting in nature, which is best managed by a team of experts specifically assembled for the task. Leadership of Task Forces and Working groups constitutes a significant programmatic responsibility, which often leads to increased leadership opportunities for DIII-D program scientists, including those from universities and other collaborating institutions. The organization of Task Forces and Working Groups for 2008 experiments appears in Fig. 13-1. The highly collaborative nature of the scientific leadership should be noted.

DIII-D Operations Division. This group is responsible for the safe and efficient operation and maintenance of the DIII-D facility. They oversee all the major hardware systems on DIII-D, including the auxiliary heating and current drive systems, the DIII-D vessel and coil systems, all major power supplies, vacuum systems, water systems, and cryo systems. The Operations Division is responsible for modifications and upgrades to these systems. This division is organized into six groups: Tokamak Operations, Tokamak Systems Engineering, Neutral Beam Operations, Electron Cyclotron Operations, Fast Wave Operations, and Electrical Systems Engineering. Staff from the major collaborators (PPPL, LLNL, and ORNL) serve in key leadership and support positions within the Operations Division.

Energy Group Theory Division. The DIII-D program relies on and benefits from close connection to theory. DIII-D scientists participate in a broad range of collaborations with theorists from around the US and abroad. The DIII-D program provides data for validating new theory and models; conversely, theory motivates and guides planning for many DIII-D experiments. The Theory Division within the Energy Group at General Atomics works closely with the DIII-D program, providing not only general theory support, but also extending to key data analysis codes such as EFIT and the ONETWO profile analysis codes. GA theorists serve on both the DEC and the Research Council.

Working Group and Task Force Leaders. DIII-D research activity is organized by the Physics Groups and by Task Forces. High priority research within the Physics Groups is usually coordinated by one or more long-lived topical Working Groups. Working Groups may draw participants from across the organization (including the Theory group), and are responsible for experimental planning, execution, and data analysis. Working Group leaders report to the Physics Group Leaders. Task Forces are formed for the specific purpose of addressing near-term high priority, high-visibility cross-cutting research tasks for which the effort is best organized outside the Physics Groups. Task Forces report directly to the Director of the Experimental Science Division.

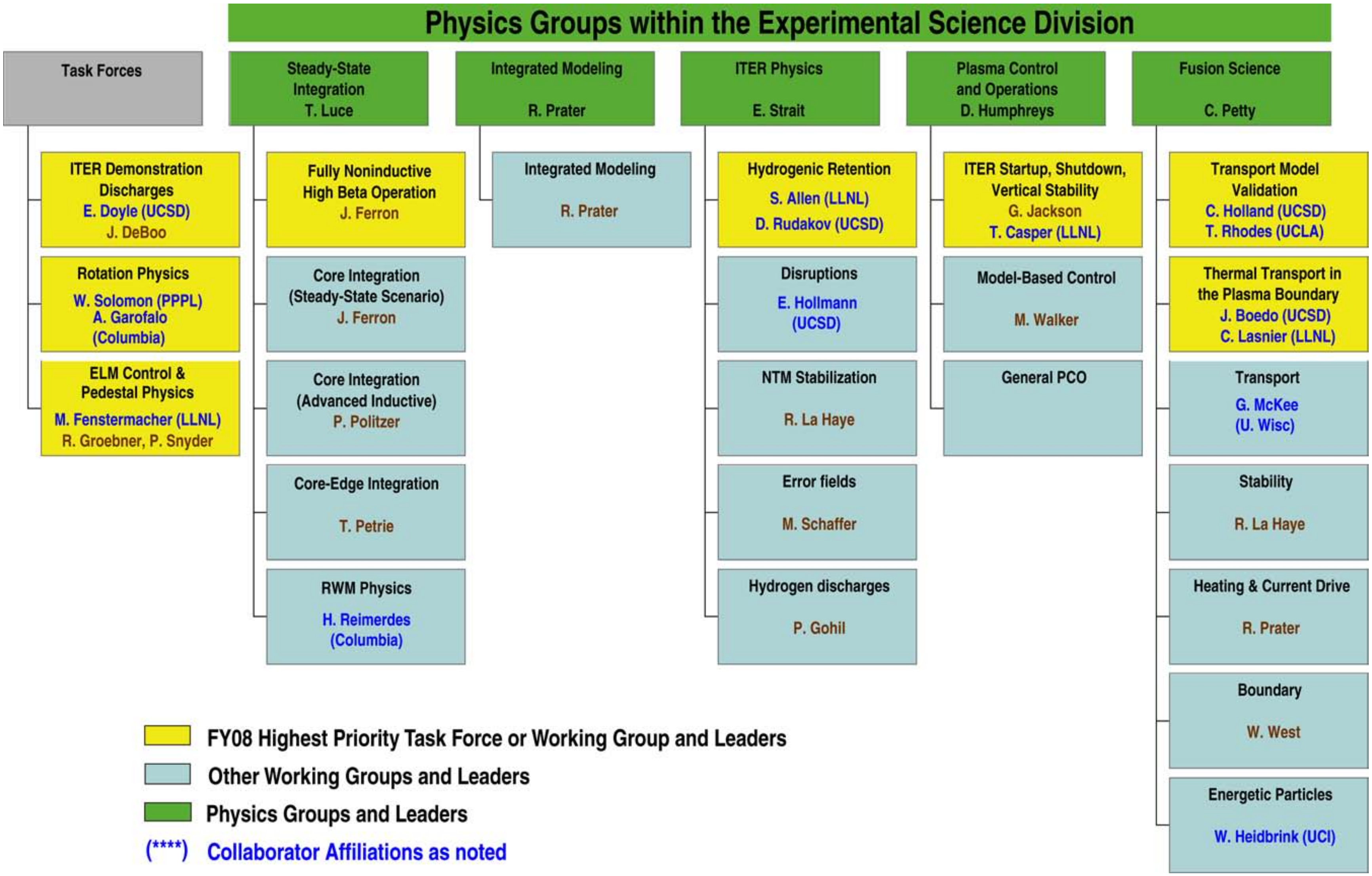

Fig. 13-1. Organization of Experimental Science Task Forces and Working Groups for 2008 DIII-D operation.

13.2. PROGRAM PLANNING

Planning for the DIII-D National Program is carried out in partnership with DIII-D management, DIII-D collaborators, and the DOE Office of Fusion Energy Sciences, with input provided by the broader (national and international) fusion community. Program plans range from daily experiments to this Five-Year Program Plan. Both General Atomics and its DIII-D collaborators provide regular reporting to DOE. In this section, we outline the planning process for these various program plans, starting from the longer-term perspective.

1. A **Five-Year Program Plan** is prepared every five years through open interaction of the DIII-D team with the international fusion community and the Office of Fusion Energy Sciences (OFES):
   a. A draft Five Year Plan is prepared by the DIII-D National Team. Multi-institutional teams are formed to flesh out various possible program elements for inclusion in the Five Year Plan.
   b. The draft plan is presented at a National Tokamak Planning Workshop conducted jointly with the NSTX and Alcator C-Mod programs. The workshop solicits broad participation nationally and internationally. International technical experts take the role of facilitator or summarizer in various main topical areas. Based on the workshop output, the draft plan is modified.
   c. GA proposes the five year research program to OFES. Companion documents from the major collaborators, which lay out the planned contributions of the collaborator, are also submitted to provide a complete view of the proposed five year DIII-D National Program to the OFES.
   d. The GA proposal based on the Five-Year Plan, is reviewed by a panel appointed by OFES.
   e. This long range plan may be updated to make it consistent with evolution of the national program priorities and with technical developments in the international fusion effort.
2. An Annual Experimental Plan is prepared as follows:
   a. A review of the previous year's results presented in a DIII-D Year-End Review, open to the community, provides the technical basis to begin developing the experimental plan for next fiscal year's operation.
   b. Research topics are identified by the DIII-D Research Council (Table 13-1).
   c. An international Research Opportunities Forum provides the opportunity for the community to propose experiments within the thrust and topical areas. Remote interactive participation is provided via the Internet.
   d. Based on the proposals, Task Force and Working Group leaders work with groups interested in the specific research area to prepare plans. These plans are presented to the Research Council, which then develops a draft experimental plan allocating machine time to each research topic.
   e. The DIII-D Executive Committee (Section 13.1) and the international DIII-D Advisory Committee (Section 13.1) review the draft experimental plan.
   f. Based on feedback received, the DIII-D Program Director issues the final experimental plan, and allocates experimental time, consistent with available resources.

g. The annual Experimental Plan is reviewed on a monthly basis, taking into account changing hardware availability and DIII-D or national program priorities as provided by the OFES, the US Burning Plasma Organization, or the ITER Project Office.

3. Monthly and Daily Experimental Planning is managed by the DIII-D Experimental Coordinator in consultation with the Director of DIII-D Operations, the Director of the Experimental Science Division, and program scientists.

13.3. FUNDING OF RESEARCH ON DIII-D

1. The major participating laboratories in the DIII-D team (GA, PPPL, LLNL, ORNL, SNLA) receive their funding directly from OFES.
2. Major university participants on DIII-D (UCSD, Texas, Columbia, Wisconsin, Georgia Tech) also receive their funding directly from OFES.
3. GA subcontracts with some universities and industries for specialized diagnostics and technical services.
4. The people who make proposals at the annual research opportunities forum meetings can apply to OFES for funding if their proposals are included in the experimental plan.
5. Universities, laboratories, and private industry may (and have) apply for DOE or NSF funding to conduct research at the DIII-D facility in response to specific calls for proposals that occur on a regular basis (e.g., SBIR, diagnostic competitions, joint projects between DOE and other government agencies, and various awards such as Young Investigator Awards or Faculty Startup awards).

13.4. REPORTING DIII-D PROGRAM ACTIVITIES

1. DIII-D issues weekly highlights to the broader fusion research community on program activities. These highlights are available on the Web at http://fusion.gat.com/diii-d/Weekly.
2. DOE conducts quarterly reviews of the program. GA and major collaborators report on facility operations, technical accomplishments, budgets, safety matters, and outstanding issues.
3. The annual Experiment Plan is submitted to DOE.
4. The DIII-D program activities are discussed at the Fusion Facilities Coordinating Committee (FFCC) meetings for coordination with other major U.S. facilities (C-Mod and NSTX).
5. The DIII-D program planning is reported at the annual OFES Budget Planning meeting at Germantown, which is open to the U.S. fusion community.
6. DIII-D program activities are discussed extensively at meetings of the US Burning Plasma Organization. Often, DIII-D results form the core technical content of USBPO reports and recommendations.
7. DIII-D is a major contributor to national and international fusion and plasma physics meetings and conferences including APS, EPS, IAEA, PSI and to many special workshops.

8. DIII-D research results are reviewed and published in many scientific, technical, and engineering journals. An extensive bibliography of DIII-D publications resulting from the previous Five Year Cooperative Agreement appears in Section 15 at the end of this document.
9. The GA DIII-D Website at http://fusion.gat.com/diii-d/Home provides an extensive collection of public information about the DIII-D program.

13.5. SAFETY

The DIII-D program places high value on safe operation of the facility by GA personnel, visitors, collaborators, and on-site contractors. GA has organized and maintains a very strong safety program for the DIII-D facility. The DIII-D Safety Officer, working in concert with the Fusion Group Safety Committee and CAL-OSHA, has developed a comprehensive safety plan and set of safety procedures to be followed by everybody working at DIII-D. This safety plan defines requirements for worker training and supervision for both GA employees and collaborators. The plan also covers corrective actions and general reporting requirements. All collaborators working on DIII-D are assigned a GA host who acts as the on-site supervisor. The on-site supervisor, working with the DIII-D safety officer and representatives from the collaborator's home institution, will ensure that the collaboration has the appropriate training to perform the assigned work. All new personnel (GA and collaborators) are first given appropriate training and documentation and must be escorted by trained personnel until their supervisor is confident that they are familiar with the DIII-D safety procedures and can work safely in the DIII-D environment.

13.6. MANAGEMENT OF THE COLLABORATIVE NATIONAL TEAM

13.6.1. General Principles of Collaboration

The following principles serve as guidelines for conducting institutional collaboration on the DIII-D Program:

1. Advancement of the DIII-D Program is held by all participating institutions to be essential for advancement of U.S. fusion energy science and to be in the interests of all DIII-D program participants.
2. Collaborators will accord high priority to their DIII-D commitments, both in the use of resources and in the assignment of personnel. GA will recognize that some collaborating personnel assigned to DIII-D activities may have additional responsibilities in their home programs.
3. In support of the DIII-D Program objectives, collaborators will be accorded lead responsibilities in defined areas and participation in other areas as spelled out in institutional Memoranda of Understanding (MOU). "Lead responsibility" does not imply sole responsibility. In those areas where it does not hold a lead, a party may elect to retain significant minority participation sufficient to develop and sustain expertise in the area. These lead or support roles will be based on consensus assessments of capability and party needs by the program leadership and the DIII-D Executive Committee. Individuals or groups which wish to collaborate on DIII-D should negotiate with the institution who has lead task responsibility. Cases of disagreement should be called to the attention of the director and Executive Committee. Institutions having lead responsibility for a task are not to delegate responsibility to another party without approval of the director.

4. GA will have sole responsibility for operating the DIII-D tokamak, though it counts on support from collaborating institutions. If a collaborator has a lead role involving an auxiliary hardware system on DIII-D, they may undertake the responsibility to operate that system. The scope of the collaborators responsibility in design, construction, and operation of systems will be defined in the individual MOU between GA and collaborating institutions.
5. In order that the DIII-D Program accomplish its programmatic objectives and the individual researchers have the opportunity to pursue rewarding research, it is generally expected the participants will spend roughly half of their time carrying out program-related support tasks (e.g., leading research thrusts, operating a diagnostic, or acting as a physics operator) and spend the other half of their time pursuing an agreed-upon research program.
6. All data, raw or analyzed, will be considered the property of the DIII-D Program and will be accessible to others in the program. The rights of first authorship and lead responsibility will be respected. It is expected that GA staff and collaborators operating diagnostics or doing specialized analysis will provide data into defined DIII-D databases on a routine basis and to other members of the program when requested.
7. Subject to DOE’s technical data rights and patent rights, all data and results from the DIII-D Program will be freely shared and acknowledged between the collaborating parties. In general, all publications or reports must go through the standard GA DIII-D review cycle. However, in the case that the work reported on is principally done by collaborating personnel using collaborators equipment and codes, the publication or report may be submitted through the collaborating institution’s review process. In such cases, a copy must be provided for timely courtesy review by the responsible DIII-D research area coordinator and division director. DIII-D division directors will make the determination of the appropriate review channel. Publications and reports will clearly identify that the work was done on the DIII-D tokamak and acknowledge DOE funding support. Detailed requirements for presentation formats, use of logos, and issues related to invited talks and papers will be managed by the DIII-D Program Director with review by the DIII-D Executive Committee.
8. DOE data and patent rights as specified in GA’s contract with DOE will take precedence in all work done on or derived from DIII-D.
9. All GA data, which GA identifies as proprietary, will be protected by individual collaborators and collaborators’ institutions.
10. Collaborating institutions are expected to participate in all DIII-D related DOE and community reviews.

13.6.2. Documents Governing Active Collaborations

MOUs are written between GA and major collaborators. MOUs generally cover the historical background that has led to the collaboration, the institutional goals and requirements of both parties for participating in the collaboration, the principles and agreed upon procedures for the collaboration, and a definition of lead and participatory roles for the collaborator. The MOU is signed by the program leaders of GA and the collaborating institution.

13.6.3. Approval Process for Project Activities

Project Management Plans are developed for facility modifications or upgrades approved by the DEC. DIII-D participants, as well as outside technical specialists, may review project plans and provide advice. Progress, costs, and schedules for special projects are reported at DOE Quarterly Reviews. Both GA and collaborator DIII-D Program tasks will be summarized in common master schedules and milestones. A manual describing the work procedures for DIII-D tasks and projects is available for all DIII-D personnel and collaborators. It describes a sequence of procedures (WP-01 through WP-14) which establish a uniform approach to developing and maintaining new capabilities at DIII-D including designing, engineering, fabricating, installing, and maintaining hardware and equipment on the DIII-D tokamak or any of its related systems. Procedures are also included to guide the performance of work in the machine pit and within the facility. Procedure WP-01, "Initiation, Review And Approval of DIII-D Task Proposals," describes the process for gaining approval of new tasks at DIII-D. Procedure WP-02, "Implementation and Completion of DIII-D Tasks," and others following, cover the whole span of engineering development from the inception of an idea through the different approval cycles to the point where the product is operational. These work procedures is available on the DIII-D local web at http://web.gat.com/support/procedures/D3WPpdfs/D3WP_tofc.pdf

13.6.4. Budget Planning for DIII-D Projects

Budgets for program tasks are generated by all tasks managers working with the DIII-D Planning and Control Group and submitted to the DIII-D program director for distribution to the Executive Committee and the DOE. Task priorities are set by the DIII-D program director in consultation with the DIII-D Executive Committee and in accordance with GA's contractual requirements with the DOE. Resource disbursements are made with input from collaborating DIII-D program leaders. The Executive Committee will also recommend on priorities of collaborators budgets. Disagreements between parties will be arbitrated by DOE when they cannot be resolved by the Institutional Leadership.

13.6.5. Program Reporting

GA will submit all required plans and reports identified in its contract with the DOE. GA will prepare a DIII-D Experimental Plan each year that details all planned experiments for that year including those to be performed by collaborators. It will be reviewed quarterly in conjunction with the DOE Quarterly Contract Review and updated as needed. The Plan will be prepared by the DIII-D Research Council, which includes representatives from the major collaborators as well as GA. Before submission to DOE for approval, it will be reviewed by the Executive Committee and approved by the DIII-D program director. Technical program reports will be submitted quarterly as part of the DOE Quarterly Review or as needed. An Annual Technical Report and Final Contract Technical Report will also be submitted. An overall Management Plan will be submitted after contract award. At the beginning of the contract and on a quarterly basis thereafter, GA will submit management status and summary reports. Annually, GA will submit a milestone schedule plan, cost plan, and milestone schedule status report.

14. ACCOMPLISHMENTS AND HISTORY OF THE DIII-D PROGRAM

14.1. SECTION OVERVIEW

The DIII-D Tokamak program is recognized to be one of the most productive in the world as measured in terms of both progress toward the practical realization of tokamak-based fusion energy production and in plasma science output. The present five-year program (2003–2008) is shaping up to be one of the most productive, enabled by new hardware capabilities that have come on line during this period. The past is prologue to future DIII-D success in fusion research and in being able to deliver on the proposals and plans detailed in this document.

14.2. ACCOMPLISHMENTS

The DIII-D Tokamak program at General Atomics has made many scientific contributions to the worldwide fusion effort. The prescient pursuit of shaped plasmas drove pioneering shape control techniques that were rewarded in record plasma beta values and reactor-relevant fusion triple products, $nT\tau$; a stable reactor core can exist. The hallmark of the DIII-D Research Program is the integration of good scientific research into advance operating scenarios aimed at optimizing the tokamak – the Advanced Tokamak. In the present five year plan the DIII-D program has moved on from establishing the viability of the AT and guiding the present ITER design toward providing critical physics solutions for key issues in order that ITER be a successful harbinger of the tokamak reactor. These particular ITER contributions are detailed in Section 14.8.

The DIII-D program pioneered not only plasma shaping but also profile control as a means of improving the plasma performance. <u>The DIII-D program has made important contributions to the present ITER design</u>, including the shape, disruption mitigation systems, MHD stability control, and advanced scenarios for ITER. <u>Stability research</u> is laying the foundation for operation at high beta, above the free boundary limit and routine operation with stabilization of the tearing mode with electron current cyclotron current drive. The important role of sheared ExB flows in forming <u>transport barriers</u> was discovered, with the result of achieving neoclassical ion confinement across the entire plasma cross section. With the recognition of the importance of the edge pedestal in achieving high performance, the DIII-D program performed pioneering research of the edge pedestal and <u>validated that the peeling/ballooning mode set the limit for the Edge Localized Mode (ELMs)</u>. Additionally, the DIII-D program pioneered the use of nonaxisymmetric coils in controlling the edge pressure gradient and <u>suppressing the ELMs</u>. The DIII-D program pioneered both the physics and technology of electron cyclotron heating and current drive, <u>validated the current drive theories</u>, and developed the use of <u>electron cyclotron current drive for current profile control and instability control</u>. The DIII-D program <u>co-discovered toroidally induced Alfven Eignenmodes</u> (TAEs) and continues to carry out detailed experiments to validate theoretical models in support of ITER. Recent advances in simulation, detailed profile measurements and turbulence measurements have made <u>validation of transport models</u> a more important and visible part of the DIII-D program. New capabilities to control the plasma rotation with co-plus counter NBI on DIII-D has emphasized the <u>importance of plasma rotation for both transport and stability</u>. DIII-D has carried out <u>pioneering work in particle and heat flux control</u> with divertor geometry, and radiative divertors. A key element and strength of the DIII-D program is <u>pioneering work in plasma control</u>, to isolate and elucidate the underlying physics, for precise control of plasma instabilities, and access and control of advanced performance regimes. The DIII-D digital plasma control system is now used on many tokamaks worldwide.

14.3. HISTORY OF THE DIII-D PROGRAM

The General Atomics Tokamak Program has a history of creative concept development. The program began in 1968 with the Doublet I device, the first tokamak with a highly noncircular cross section, using solid copper walls to shape the plasma. Experiments on this device showed the doublet configuration to be magnetically and dynamically stable. These successes led in 1971 to the larger Doublet II device, also with solid copper walls. This device was reconfigured in 1974 to use external coils to replace the copper walls. The new device was named Doublet IIA, and it pioneered the use of external coils to shape a wide range of highly noncircular plasmas and maintain them in nondecaying magnetic configurations.

The success of these experiments led to construction of the Doublet III device, completed in 1978. In the first years of operation, it was the largest operating tokamak in the world and attained the highest plasma current levels recorded at that time (2.2 MA). Experiments with a broad range of plasma configurations demonstrated the importance of elongation and shape control. Dee-shaped plasmas proved easiest to form and were projected to reach β values adequate for viable power plants. Diverted dee-shaped plasmas were also effective in achieving reduced impurity levels and enhanced confinement.

These successes led to the reconstruction of the Doublet III tokamak into a large dee-shaped cross section capable of a wide range of plasma shapes and divertor configurations. The device was renamed DIII-D in 1986. DIII-D rapidly reached currents of over 3 MA and achieved superior levels of confinement and β. DIII-D set and still holds the record of 13% beta for a conventional aspect ratio tokamak. Another significant numerical achievement was reaching a value of the fusion triple product $nT\tau$ of 7×10^{20} keV-s/m^3 corresponding to an equivalent fusion gain of 0.3.

In the late 1980s and early 1990s, DIII-D contributed, with other world tokamaks, to developing an understanding of routine tokamak performance that projected to a successful burning plasma experiment. The main parametric dependences of plasma confinement were found to be common among the various tokamaks, allowing the development of confinement scaling laws which implied a common underlying physics for the results and which allowed empirical extrapolation to burning plasma experiments. The limits to the stable operating space were understood and the empirical beta limit was in accord with Troyon scaling and also in agreement with theory. These developments allowed the definition of the standard tokamak operating space as given by an H factor of 2 for conventional ELMing H-mode operation and a normalized beta of 2 for the beta limit. This physics basis was and is adequate to project to burning plasma experiments and ultimately fusion reactors.

However, the DIII-D Team saw that the tokamak as a magnetic confinement configuration had potentially much more to offer than this nominal performance. In the early 1990s, modes of enhanced confinement considerably above the nominal H=2 scaling were being realized. Theory calculations implied normalized beta values up to perhaps 6 might be possible with wall stabilization, strong shaping, and broad pressure profiles. The DIII-D Team coined the term “Advanced Tokamak” to capture that package of research issues aimed at finding out just what the limits of the tokamak configuration could be as a magnetic confinement device. Since that time a major emphasis of the DIII-D Program has been Advanced Tokamak physics. With the stabilization of the resistive wall mode, operation above the free boundary limit has been realized. Advanced Tokamak research is also closely aligned with the requirements for steady state, since a high bootstrap current fraction requires a high normalized beta and enhanced confinement at lower plasma current than is given by H=2. Discharges with 100% noninductive current have been obtained that project to Q=5 in ITER for durations in excess of several energy confinement times, and 90% noninductive discharges have been obtained for approximately a current redistribution time of 2 s. We are confident that sustaining these "steady-state" discharge at 100% noninductive plasma current for several current redistribution times can be accomplished with additional co-injected NBI and additional off-axis current drive. Advanced Tokamak research is now a major effort in many of the world tokamaks and is the main goal for a number of planned tokamaks, EAST (China, first plasma 2006), KSTAR (Korea), and JT-60SA (Japan).

The importance of ITER in DIII-D research is clearly evident, especially in the research results of the last Five-Year Program Plan. Innovative solutions to the ELMs in ITER have been developed with the QH-mode and ELM suppression with resonant magnetic perturbations. New high performance operating scenarios have been developed for ITER, the advanced inductive and hybrid scenarios, that potentially provide higher Q in ITER or longer pulse duration for exploitation in the Phase II of ITER, for example evaluating test blanket modules.

The purpose of the following sections is to offer some perspective on DIII-D research in the last five years 2003–2007, not to give a detailed recounting of the research results. We will also discuss what was accomplished with respect to what the previous five-year plan foresaw. This material is meant to set a basis point for the forward looking Five-Year Plan that has been presented in previous sections of this Plan. The present five-year period is 2003–2008 so all of the research campaign in 2008 remains to be completed as of this writing.

14.4. CHANGES IN THE U.S. FUSION PROGRAM

The U.S. and six other international parties have agreed to build ITER, an international Tokamak "to demonstrate the scientific and technological feasibility of fusion energy." ITER is a presidential initiative. The intent for the U.S. to move forward with ITER came in 2003, and the ITER Organization became a legal entity in November 2007. ITER design and construction is now a reality, and this is the most significant change in the U.S. Fusion Program over the last decade. The U.S. ITER Project Offices was formed at ORNL to be responsible for the U.S. contributions to ITER construction. The U.S. Burning Plasma Organization was formed to advance burning plasma science and provide the coordination of the U.S. scientific efforts for ITER and burning plasmas. An appropriate interface with the new U.S. organizations and the international structure is an important consideration of the DIII-D Program. ITER is the most important element of the U.S. fusion effort, and the success of ITER is the highest priority of the DIII-D Program.

14.5. EVOLUTION OF THE DIII-D MISSION STATEMENT FROM THE 2003–2008 PLAN

The DIII–D Program mission statement adopted in this Five-Year Plan is

To establish the scientific basis for the optimization of the tokamak approach to fusion energy production.

This is the same mission statement as in the 2003–2008 program plan. We strongly feel that this statement captures the essence of the DIII-D Program intent, maintaining a strong focus on excellent science focusing on innovation and optimization, and with the goal of attractive fusion energy. In the 2003–2008 plan the primary focus was on Advanced Tokamak research with a strong emphasis on advancing fusion science and concepts. With the emergence of ITER construction, we now articulate this mission statement in three main research goals, or themes, for 2009–2014.

1. Enable the success of ITER by providing physics solutions to key issues.
2. Establish the physics basis for steady-state high performance operation for ITER and beyond.
3. Advance fundamental understanding of fusion plasmas along a broad front.

The first of these goals recognizes the importance of the success of ITER in the U.S. and international program. The near term DIII-D program focus will be heavily influenced by the research needs of ITER. The second goal reaffirms our commitment to Advanced Tokamak research, to find the optimization of the tokamak as an attractive energy source. The idea of optimization carries with it a commitment to creativity and innovation. The third goal expresses the desire of the DIII-D research staff and the responsibility of the DIII-D program to continue to be a leader in developing the fundamental understanding of fusion plasmas. We believe building on a sound and tested scientific foundation is the most reliable and most effective way to advance fusion energy. This embodies doing “science with a purpose.”

14.6. REORGANIZATION OF EXPERIMENT PLANNING

The DIII-D program planning was reorganized in 2007 to focus more tightly around these three restated major goals of the research program, ITER, Advanced Tokamak, and fusion energy science. The Experimental Science Division was reorganized into five physics groups: ITER Physics, Steady-State Integration, Fusion Science, Integrated Modeling, and Plasma Control and Operations. The first three of these physics groups are organized along the three main DIII-D research goals. The integrated modeling group was formed to develop the tools for, and carry out a more focused research effort on model validation. The plasma control group was formed in recognition of the important role control plays in performing well-characterized fusion science experiments, the opportunity for excellent science research that control provides, and the importance control plays in steady-state advanced tokamak research. A matrix structure was adopted for implementing the research program as shown in Table 14-1. The five rightmost columns are the five physics groups as just described, and the rows are the annually established task forces and working groups. The task forces and working groups shown are those adopted for the 2008 experiment campaign. The five physics groups recognize the long-term focused commitment of the program toward these research themes. The Fusion Science Group contains the five topical science areas (transport, stability, energetic particles, heating and current drive, and boundary/divertor physics) that provide a continuing multi-year long-term effort in basic fusion energy research. The task forces represent more urgent near term focus upon goal oriented research. The task forces and working groups are expected to change from year to year on the basis of recommendations from the research council.

Table 14-1
Task Forces and Working Groups for the Research Opportunities Forum

Physics Groups

High Priority Task Force / Working Group	Leader	Deputy	Steady-State Integration	Integrated Modeling	ITER Physics	Plasma Control and Operations	Fusion Science
ITER Demonstration Discharges	E. Doyle (doylej@fusion.gat.com)	J. DeBoo (deboo@fusion.gat.com)	✓	✓	✓	✓	
Rotation Physics (http://fusion.gat.com/diii-d/Rotphystf08)	W. Solomon (solomon@fusion.gat.com)	A. Garofalo (garofalo@fusion.gat.com)	✓	✓	✓		✓
ELM Control and Pedestal Physics	M. Fenstermacher (fenstermacher@fusion.gat.com)	R. Groebner (groebner@fusion.gat.com), P. Snyder (snyder@fusion.gat.com)	✓	✓	✓		✓
Steady-state high-beta operation	J. Ferron (ferron@fusion.gat.com)		✓	✓		✓	
Transport Model Validation	C. Holland (cholland@ferp.ucsd.edu), T. Rhodes (rhodes@fusion.gat.com)			✓			✓
Thermal Transport in the Plasma Boundary	J. Boedo (boedo@fusion.gat.com)	C. Lasnier (lasnier@fusion.gat.com)		✓			✓
Hydrogenic Retention	S. Allen (allens@fusion.gat.com)	D. Rudakov (rudakov@fusion.gat.com)			✓		✓
ITER Startup, Shutdown, and Vertical Stability	G. Jackson (jackson@fusion.gat.com)	T. Casper (casper1@llnl.gov)	✓		✓	✓	

More information on the high priority research areas is available in these memos:

- Announcement of high priority research topics
- Announcement of area leadership (including contact information)

14.7. FACILITY OPERATION AND DEVELOPMENT

The 2003–2008 Five-Year plan laid out an ambitions plan for increasing the scientific and operational capabilities of the DIII facility. For the first four completed years of this term, DIII-D successfully operated at 104% of the commitment for this period, with an average of 14.8 weeks per year. This included maintaining the successful operational output through the period of the Long Torus Opening in which the beamline reversal and lower divertor modifications were performed. Funding limitations led to a reduction in actual operational time as compared with the proposed 21 weeks per year during this period. Likewise, our achieved level of facility development fell short of the our plans given these limitations in program funding, nevertheless, a significant number of the hardware upgrades were implemented by innovative scheduling of the research operations and facility maintenance.

The 2003–2008 Five-Year DIII-D plan called for seven significant upgrades to the DIII-D facility:

1. Increase of the electron cyclotron power from 3 (10 s, 1 MW) and 3 (2 s, 0.8 MW) gyrotrons to a 9 MW 10 s system.
2. Reorient the one beam line from the co-plasma current direction to the counter-direction
3. Addition of internal nonaxisymmetric coil set and high bandwidth power supplies for RWM control
4. High triangularity double null pumping
5. Add new and upgrade existing diagnostics
6. Rebuild the Fast Wave RF transmitters
7. 10 s long pulse capability

Items 2, 3, 4, 5 and 6 were all completed during this five year program plan. A significant amount of work on item 7, the 10 s pulse length capability, was completed, including modification of the TF belt bus, and the acquisition of a prime power transformer for 10 s capability. Installation of the transformer remains as a future task. The electron cyclotron system was upgraded to a 6, 10 s, 1 MW each, gyrotron system. In addition to 10 new diagnostic instruments, a significant number of diagnostic system upgrades were implemented.

The high triangularity lower pump, the neutral beam re-orientation, the EC upgrade, and many of the diagnostic improvements were made possible by adjusting the operations schedule as shown in Fig. 14-1. Historically, DIII-D experimental operations occupied approximately two-thirds of the year, with approximately one-third of the year reserved for diagnostic calibrations, maintenance, and refurbishments. By continuing experimental operations from 2004 through the early part of 2005 without a major vessel entry vent, we were able to provide a long maintenance period, called the Long Torus Opening, and still complete our scheduled experimental run time in 2005 and 2006. Both operations and scientific research staff, normally carrying out research, were redirected during this period to focus on facility improvements,

Fig. 14-1. The elimination of a vent/maintenance period allows DIII-D to complete the scheduled run line and allows an intended maintenance period (long torus opening) for facility enhancements.

14.7.1. EC Systems

The EC system plan in the 2003 five-year program plan called for increasing the EC to six 10 s 1 MW gyrotrons, and adding two 10 s, 1.5 MW gyrotrons, being developed by the VLT. The three new 1 MW gyrotrons were ordered and are operating. Two of the high voltage power supplies were modified to power two gyrotrons, and the controls of the third power supply were modified for the CPI gyrotrons. The 1.5 MW prototype gyrotron was built, but failed before delivering power to the DIII-D tokamak.

This Five-Year program plan will continue to increase the EC capability of the DIII-D tokamak, aiming for 12 MW capability. This will require completion of the development of the 1.5 MW gyrotrons, building a fourth power supply, and installing two additional transmission lines and an additional launcher.

14.7.2. Neutral Beam Systems

During the last five year contract period, the 210 deg beamline was disassembled and removed from the pit area and re-installed in the counter direction. This required disconnecting all water, electrical, cryogenic, and vacuum systems from the beam-line, and after re-installing, reconnecting them. The drift duct and bellows had to be redesigned and welded into the vessel. One of the 24 toroidal field bundles had to be machined to allow for installation in the counter direction. This work was completed on schedule. This reorientation of the beam line allowed independent control of total power (beta) and torque (rotation), allowing important controlled experiments in transport and stability to be completed. This capability has enabled significant and important new research to be carried out: a post deadline 2006 IAEA paper and a post-deadline 2006 APS paper resulted from this capability. New important research continues. In addition to the rotation of the beam line, GA developed the capability to refurbish and build ion sources: two ion sources were rebuilt.

This five-year program plan calls for re-establishing operation of 8 ion sources, 6 in the co-direction, reorienting two beam lines for off-axis NBI injection, and increasing the beams to 10 s, 20 MW capability.

14.7.3. Internal Nonaxisymmetric Coil Set

Twelve internal coils (I-coil), 6 above the midplane and 6 below the midplane, were installed with the goal of active control of the resistive wall mode, RWM. A very flexible patch panel was constructed that allows connection of the coils in a number of configurations, including toroidal modes n=1, n=2, n=3.

During the past five year program, 24 high frequency bandwidth amplifiers (audio amplifiers) were installed to provide fast active feedback. The resultant RWM physics program has been very productive. In addition, the coils were used to stabilize ELMs by ergodizing the boundary region.

In this Five-Year Program, we will install a set of coils on the inside wall to further pursue optimizing the stabilization of ELMs.

14.7.4. High Triangularity Lower Divertor

The 2003 plan was to install a high triangularity lower pumped divertor. In 2006, the lower divertor shelf was extended to provide pumping and density control for double null divertor discharges.

In this Five-Year Program Plan, we plan to replace the graphite tiles in the divertor region with carbon fiber composite tiles sufficient to handle 300 MJ 10 s pulses.

14.7.5. Diagnostics

A large number of diagnostic systems were refurbished and improved during the last five-year program. During the LTOA, improvements were made to 40 diagnostic systems: significant new capability was provided for a number of systems, ECE radiometer, FIR scattering, BES, Langmuir probes, CO2 interferometer, filterscopes. New diagnostics were also developed, counter viewing MSE, counter viewing CER, Mod B by D_{α}, MiMES (midplane), QMBs and new SXR poloidal arrays.

In this five-year plan, we are planning a significant number of new diagnostic systems to maintain leadership in fusion science experiments (see Chapter 8).

14.7.6. Fast Wave Systems

The fast wave system was brought back into operation during the last five years, and 3 MW of FW power injected into a DIII-D plasma. Rebuilding the power amplifiers in the two ABB systems was begun and should be complete by the end of 2008. The vacuum-wall feed-throughs on the antennas at 0 and 180 deg failed at 115 MHz: the failure was the result of an undetected arc at a location of minimum rf voltage placed in a vulnerable location at this frequency. The 0 and 180 deg antenna are now operating at 90 MHz pending the development of a more robust arc detection system.

This five-year program plan will continue increasing the power of the FW system, and we plan to replace the 285 deg antenna with a new (more modern design) long pulse antenna.

14.7.7. Infrastructure and the 10 s Pulse Upgrade

A number of refurbishments and upgrades were implemented over the last five years to maintain DIII-D as a reliable scientific facility for the coming decade. These include installation of the acquired prime power transformer, replacement of cooling towers that will provide adequate cooling for DIII-D's future plans, replacement of water hoses on the toroidal field coil, refurbishment of the Ohmic Heating Power Supply control system, begin the replacement of outdated CAMAC data acquisition with new PCI based data acquisition systems, and replacement of a number of computers systems.

14.8. SCIENTIFIC ACCOMPLISHMENTS IN THE PERIOD 2003–2007

Here we identify some of the past accomplishments that apply directly to the three main research themes for the period 2009–2013. These themes are highly interrelated when applied to ITER and burning plasma physics. For example, the issue of NTM stabilization is a concern for all ITER scenarios, but more so for AT scenarios that target higher normalized beta values. Further, the scientific understanding of NTM stabilization with ECCD is crucial for specifying a system that will function adequately, reliably, yet not be over-specified at unneeded cost.

All of these accomplishments have been described within the body of this proposal, in the specific applicable sections. Here we provide a focus on some that relate to present ITER issues and have established the ongoing DIII-D progress toward solutions.

14.8.1. Research in Support of ITER

- **ELM Mitigation.** ELMs remain a critical issue of active focus for ITER.
  - DIII-D discovered that edge-resonant magnetic perturbations can be used to quench ELMs. This has been demonstrated in the ITER shape, and at projected ITER collisionality. The design of an ELM suppression coil set is an active issue for ITER for which DIII-D will conduct the primary experiments.
  - DIII-D discovered the so-called Quiescent H-mode, an H-mode without ELMs, originally accessed by counter-I_p directed NBI. The operating space for QH mode has been expanded to lower, more ITER-relevant rotation, with a mixture of co-I_p and counter NBI, and to higher plasma density for greater projected fusion yield.
  - DIII-D will continue to push on the limits of the QH-mode operating space and establish utility for ITER, and the tokamak reactor.
- **Disruption Mitigation.** Required for long-term survival of the ITER first wall.
  - DIII-D has shown that a massive, rapid injection of a neutral gas jet can produce a soft landing in a discharge otherwise headed toward a disruption, the equivalent of a crash landing.
  - Continued research will test the best atomic composition for the injection, the use of solid pellet injection, and will seek to elucidate the effects upon diffusing the thermal energy and the electromagnetic energy.
- **Resistive Wall Mode Control.** ITER steady state scenarios with broad current profiles can be susceptible to the RWM.
  - Active feedback stabilization of the RWM with nonaxisymmetric perturbation coils both external and internal to the vacuum vessel has been pioneered in DIII-D. It works. The new DIII-D balanced beam capability has revealed that relatively small toroidal rotation appears to stabilize the RWM.
  - Ongoing DIII-D experiments will serve to provide detailed guidance for ITER as relates to the coil set, an active feedback system, and possible NBI torque to guarantee the advantageous rotation level.

- **NTM Stabilization.** Necessary to avoid reduction of energy confinement and fusion output in ITER, as well as reduce the risk for disruption.
  - DIII-D has applied leading expertise in EC wave physics to demonstrate active stabilization of 3/2 and 2/1 NTM modes with ECCD. Coupled with the DIII-D advanced control system expertise, multiple techniques have been developed to track the mode's spatial target for the current drive, and to actually preemptively inhibit the onset of a mode.
  - Future experiments will serve to identify the detailed EC system design for ITER, in terms of power, launching attributes, and response time.
- **Plasma Startup in ITER.** Must be robust and flexible for multiple multiple scenarios.
  - DIII-D addressed ITER concern that the nominal startup technique would be near the limit for vertical instability. DIII-D experiments verified this concern, and identified modifications to the technique to avoid the problem.
  - DIII-D will continue to utilize its flexibility in shaping and startup techniques to test the planned scenarios as well as identify more favorable ones for ITER.
- **Scenario Demonstration for ITER.** To achieve greater performance than the baseline.
  - Long pulse hybrid plasmas with performance consistent with Q ~ 10 in ITER have been demonstrated in DIII-D at low rotation, more consistent with ITER projections.
  - These experiments will continue with emphasis on integration with the divertor and ELM suppression capabilities.

14.8.2. Steady-State Advanced Tokamak Research

- **Fully Noninductive Operation at high β_N.** Necessary for steady state with adequate fusion performance.
  - DIII-D has achieved for at least one resistive time 100% noninductive (NI) operation with $\beta_N < 3.5$ and more recently > 90% NI operation with $\beta_N > 3.5$, the latter limited only by the neutral beam pulse length capability. These were achieved with the so-called elevated q_{min} scenarios.
  - In the upcoming period DIII-D anticipates raising the latter case to 100% NI with additional capability in ECCD and NBCD.
- **Demonstrate Other Possible Steady State Scenarios.** High-ℓ_i scenario may not require active RWM stabilization.
  - High-ℓ_i discharges, formed with the ITER reference startup scenario, transiently achieved $\beta_N > 4.5$ with $H_{89} > 3$.
  - The high-ℓ_i scenario has had little experimental time relative to the elevated q_{min} scenarios. This promising result will be pursued in the future. The Fast Wave system is a key for active control of the sawtooth instability, which is present in these discharges.
- **Control Plasma Density in Steady State.** A basic necessity for steady state operation.
  - DIII-D has demonstrated good density control in double-null plasma configurations in both Hybrid and Advanced Tokamak discharges.

— Future experiments will continue to develop the capabilities to integrate density and heat load control with the steady state high β core.

14.8.3. Progress in Scientific Understanding

- **Core Transport Barriers at integer q_{min}.** The details of transport barrier formation are important for burning plasma understanding.
  — DIII-D discovered sustained core transport barrier formation at integer values of q_{min} when the background rotational shear is sufficiently large. Most importantly, this was theoretically predicted with GYRO simulations that indicate a role for zonal flow decorrelation of turbulent eddies.
  — This effect will be pursued both experimentally and theoretically in the future.
- **High-k Turbulence.** Electron energy transport remains a challenge to be understood.
  — DIII-D verified the existence of high-k turbulent activity, consistent with the Electron Temperature Gradient (ETG) mode activity, and further, that changes in this turbulence are correlated with changes in electron transport.
  — Diagnostic upgrades will be made to bring greater measurement capability to this promising result in pursuit of understanding electron energy transport.
- **"Intrinsic Rotation" physics.** Toroidal rotation in ITER must be known, predicted, as it is foreseen to affect system design as well as ITER performance.
  — Toroidal rotation exists in the tokamak without external torque applied to the interior. This is of practical value for ITER, but must be understood to be used to advantage in the planning of many auxiliary systems. In 2008 this topic is the subject of the DOE FES Joule Milestone.
  — DIII-D has unique capability in balanced NBI and diagnostic systems to discover the cause, and confinement of intrinsic toroidal momentum in the plasma.
- **Fast Ion Transport due to Instabilities Driven by Energetic Particles.** Applies to fusion product (alpha) transport in ITER.
  — DIII-D has obtained excellent agreement between the measured and theoretically predicted profiles of fast ions in the presence of incremental transport driven by such instabilities, indicating a theoretical understanding.
  — Upgraded fast ion profile diagnostics will allow DIII-D to further probe the details of this theoretical agreement.
- **The L-H Power Threshold depends upon Toroidal Rotation.** The physics of this threshold is yet to be understood in detail, yet is the operating condition for ITER.
  — With the emergence of the balanced NBI capability, DIII-D has discovered that the H-mode power threshold depends upon plasma rotation. It may be lower than presently believed for ITER, with the expected lower relative rotation level there.
  — Future experiments upon this exciting new phenomenon may at last reveal the critical mechanism for the bifurcation that leads to an edge transport barrier in the H-mode.

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APPENDIX A
LIST OF ACRONYMS

0-D	zero dimensional
2D	two dimensional
3D	three dimensional
AAs	audio-amplifiers
ABB	Asea Brown Boveri
ABB1	Asea Brown Boveri Transmitter #1
ABB2	Asea Brown Boveri Transmitter #2
AE	Alfvén eigenmode
AIMHD	almost ideal magneto-hydrodynamics
ARIES-AT	Advanced Reactor Innovation Evaluation Study - Advanced Tokamak
ALARA	as low as reasonably achievable
ALCATOR C-Mod	MIT Tokamak Modification
ALPS	advanced limiter-divertor plasma-facing systems
ANL	Argonne National Laboratory
AORSA	all orders spectral algorithm
AORSA RF	all orders spectral algorithm – radio frequency
APS	American Physical Society
APS/DPP	American Physical Society/Division of Plasma Physics
ARIES	Advanced Reactor Innovation Evaluation Study
ARIES-RS	Advanced Reactor Innovation Evaluation Study - Reversed Shear
ARRIBA	Alpha Radioisotope Remote Ion Beam Analysis
ASDEX-U	Axisymmetric Divertor Experiment Upgrade (Tokamak in Garching, Germany)
ASIPP	China Science Academy Institute of Plasma Physics, Hefei
AT	Advanced Tokamak
ATJ	Union Carbide graphite
BAAE	beta-induced acoustic Alfvén eigenmode
BAE Gap	beta-induced acoustic Alfvén eigenmode gap
BES	beam emissions spectroscopy
BPO	Burning Plasma Organization
C13	carbon 13; isotope of the element atomic number =12
CAL-OSHA	California Office of Safety and Hazard Abatement
CAMAC	computer automated measurement and control
CCFM	Centre Canadien de Fusion Magnetique (Varennes, Quebec, Canada)
CCR	central control room
CD	current drive
CEA-Cadarache	Commissariat a I'Energie Atomique, Cadarache, France
CECE	central electron cyclotron emission
CEMM	Center for Extended MHD Modeling
CER	charge-exchange recombination

CFC	carbon fiber composite
CFN-IST	Centro de Fusao Nuclear, Instituto Superior Tecnico, Portugal
CIEMAT	Centro de Investigaciones Energéticas, Medioambientales y Technológicas
C-Mod	MIT Tokamak Modification C
CODAC	computer operation, data acquisition and control
Consorzio RFX	Consortium Reversed Field Experiment, Italy
cPCI	compact phase contrast imaging
CPES	Center for Plasma Edge Simulation
CPI	Communications and Power Industries
CPU	central processing unit
CQ	current quench
CREATE-Italy	University group, Italy
CRPP-Lausanne	Centre de Recherches in Physiques des Plasmas, Lausanne, Switzerland
CSWPI	Center for Simulation of Wave-Plasma Interactions
CTF	Component Test Facility
CW	continuous wave
CX	charge exchange
DAM	data analysis monitoring
dc	direct current
DCLL	dual coolant lithium lead
DDD	detailed design description
DEC	DIII-D Executive Committee
DED	dynamic ergodic divertor
DEFC	dynamic error field correction
DEMO	demonstration power plant
DEMO-AT	demonstration power plant – Advanced Tokamak
DIII-D	Doublet Three Dee
DiMES	Divertor Materials Evaluation System
Div/Sol	divertor scrape-off layer
DN	double null
DND	double-null divertor
DOE	Department of Energy
DOE FES	Department of Energy Fusion Energy Sciences
DT	deuterium and tritium (fusion fuel)
DTI	Department of Trade and Industry, United Kingdom
EAE Gap	elliptical Alfvén eigenmode gap
EAST	Experimental Advanced Superconducting Tokamak
EC	electron cyclotron
ECC	edge coordinating committee
ECCD	electron cyclotron current drive
ECE	electron cyclotron emission
ECEI	ECE imaging system
ECH	electron cyclotron heating

ECHPS #1	gyrotron power supply #1
ECHPS #2	gyrotron power supply #2
ECHPS #3	gyrotron power supply #3
ECHPS #4	gyrotron power supply #4
EF	error field
EFCCs	error field correction coils
EFDA	European Fusion Development Agreement
EFDA-JET	European Fusion Development Agreement – Joint European Torus
EFIT01	EFIT using equilibrium model 01
EFIT02	EFIT using equilibrium model 02
EFIT3D	EFIT in three dimensional mode
EHO	edge harmonic oscillation
ELM	edge localized modes
EP	energetic particle
EPS	European Physical Society
ESL	Edge Simulation Laboratory
ESnet	Energy Sciences Network
ETB	edge transport barrier
ETG	electron temperature gradient
EU	European Union
Euratom/IST	European Atomic Energy Community/Instituto Superior Tecnico, Portugal
EUV	extreme ultraviolet
F8	DIII-D field shaping coil #8
F8A	DIII-D field shaping coil #8A
F8B	DIII-D field shaping coil #8B
FACETS	framework for core-edge transport simulations
FDF	Fusion Development Facility
FDF/CTF	Fusion Development Facility/Component Test Facility
FES	Fusion Energy Sciences
FFCC	Fusion Facilities Coordinating Committee
FIDA	fast ion D-alpha
FIR	far infrared
FMIT	fusion materials irradiation test
FP	Fokker-Planck
FPA	final power amplifier
FPGA	field programmable gate array
FSP	fusion simulation project
FTE	full-time equivalent
FW	fast wave
FWCD	fast wave current drive
FZ (Julich)	Forschungszentrum, Jülich, Germany
GA	General Atomics
GCNMP	globaly convergent Newton method

GPSC	SciDAC Center for Gyrokinetic Particle Simulations of Turbulent Transport in Burning Plasmas
GUI	graphical user interface
H&CD	heating and current drive
H/D	hydrogen/deuterium
HBT-EP	high beta tokamak – extended pulse
HF-CHERS	High Field Charge Exchange Recombination Spectroscopy
HFS	high field side
HP	Hewlett Packard
HV	high voltage
HVDC	high voltage direct current
IAEA	International Atomic Energy Agency, Vienna
ICC	Innovative Confinement Concepts
ICF	Inertial Confinement Fusion
ICH	ion cyclotron heating
ICRF	ion cyclotron radio frequency
ICRH	ion cyclotron resonance heating
ICRH/CD	ion cyclotron resonance heating/current drive
IEA	International Energy Agency, Paris
IEEE	Institute of Electrical and Electronics Engineers, New York
IFMIF	International Fusion Materials Irradiation Facility
IFP-CNdR	Instituto di Fisica del Plasma, Milan, Italy
IFRC	International Fusion Research Council
IFS	Institute for Fusion Studies
IM	integrated modeling
INEL	Idaho National Engineering Laboratory
IO	ITER Organization
IPA	intermediate power amplifier
IPC	integrated plasma control
IPO	ITER Project Office
IPP	Institute for Plasma Physics
IPR	Institute for Plasma Research
IR	infrared
IRTV	infrared television
IT	Fundameteel Onderzoek de Materie (Fusion Research Institute, the Netherlands)
ITB	internal transport barrier
ITG	ion temperature gradient
ITPA	International Tokamak Physics Activity
IVCC	in-vessel control coils
IW	inner wall
JAEA	Japan Atomic Energy Agency
JAERI	Japan Atomic Energy Research Institute

JET	Joint European Torus, Abingdon, Oxfordshire, United Kingdom
JET DTE1	JET DT Experiment #1
JET-EP	Joint European Torus – Extended Pulse
JFT-2M	tokamak at JAEA Tokai
JT-60SA	Japan Tokamak-60 Super Advanced (planned)
JT-60U	largest Japanese tokamak
KBM	kinetic ballooning modes
KBSI	Korean Basic Science Institute
KFA Jülich	Kernforschungsanlage, Julich (German Nuclear Research Institute)
Khrkov IPT	Kharkov Institute for Plasma Theory, Ukraine
KSTAR	Korean Superconducting Tokamak Advanced Research
LANL	Los Alamos National Laboratory
LBNL	Lawrence Berkeley National Laboratory
LCFS	last closed flux surface
LCSs	local control stations
LFS	low field side
LHCD	lower hybrid current drive
LHD	Large Helical Device, Japan
LIB	lithium beam
LIM	limiter
LINUX	computer operating system
LLNL	Lawrence Livermore National Laboratory
LSF	load sharing facility
LSN	lower single-null (plasma shape)
LTOA	long torus opening activity (DIII-D modifications 2005)
MAST	Mega-Ampere Spherical Tokamak (UKAEA-Culham)
MCI	Monte-Carlo integration
MDS	multichord divertor spectrometer
MDSplus	data handling software system
MFTF	Mirror Fusion Test Facility (LLNL)
MG	motor generator
MG2	motor generator #2
MGI	massive gas injection
MHD	magnetohydrodynamic
MiMES	Midplane Material Evaluation Station
MIRNOV	Russian physicist whose name is synonomous with magnetic fluctuations in tokamak
MIT	Massachusetts Institute of Technology, Cambridge, Massachusetts
Mod B	Modulo-B, surfaces of magnetic field strength
MOU	memoranda of understanding
MPI	message passing interface
MSE	motional Stark effect
MST	Madison Symmetric Torus, U. Wisconsin

NB	neutral beam
NBCD	neutral beam current drive
NBI	neutral beam injection
NFC Project	National Fusion Collaboratory Project
NFRC	National Fusion Research Center, Korea
NIFS	National Institute for Fusion Science, Nagoya, Japan
NPA	neutral particle analyzer
NRC	Nuclear Regulatory Commission OR National Research Council
NREL	National Renewable Energy Laboratory
NSTX	National Spherical Torus Experiment
NTM	neoclassical tearing modes
NYU/Courant Inst	New York University/Courant Institute
OASCR	Office of Advanced Scientific Computing Research
OCB	open/closed magnetic boundary
OFES	Office of Fusion Energy Sciences
OFES/OASCR	Office of Fusion Energy Sciences/Office of Advanced Scientific Computing Research
ORISE	Oak Ridge Institute for Science Education
ORNL	Oak Ridge National Laboratory
OSF	open software foundation
OSM	onion-skin method
PAC	Program Advisory Committee
PCI	phase contrast imaging
PCS	plasma control system
PEGASUS	Spherical tokamak at University of Wisconsin
PF	poloidal field
PFC	plasma facing components
PFM	plasma facing materials
PIs	principal investigators
PLC	programmable logic controller
PLOT12	commercial scientific plotting software
PMI	plasma-material interaction
PPPL	Princeton Plasma Physics Laboratory
PRC	People's Republic of China
PS	power supply
PS1	power supply #1, (for DIII-D gyrotron system)
PS2	power supply #2, (for DIII-D gyrotron system)
PS3	power supply #3, (for DIII-D gyrotron system)
PSI	plasma surface interaction
QMBs	quartz microbalance
R&D	research and development
RC	Research Council
RE	runaway electron

rf	radio frequency
RFA	resonant field amplification
RFP	reversed field pinch OR Request for Proposal
RMP	resonant magnetic perturbations
RSAE	reversed shear Alfvén eigenmode
RT	real time
RWM	resistive wall mode
SAIC	Science Applications International Corp.
SAN	storage area networks
SAV	Slovak Academy of Sciences
SC	superconducting
SciDAC	Scientific Discovery through Advanced Computing
SCRs	silicon controlled rectifiers (power supply component)
SDSU	San Diego State University
SN	single null
SNLA	Sandia National Laboratory-Albuquerque
SNLL	Sandia National Laboratory-Livermore
SOL	scrape-off layer
SPAs	switching power amplifiers
SPRED	ultraviolet spectroscopy system
SQL	standard query language
SS	steady state
SSO	steady state operation
SST-1	steady state tokamak (India)
ST	spherical torus
STAC	Science and Technical Advisory Committee
STAR	name of local computer cluster
SWIM	Center for Simulation of Wave Particle Interaction with Magnetohydrodynamics
SWIP	Southwest Institute of Physics, China
SXR	soft x-ray
TAE	toroidicity-induced Alfvén eigenmode
TAG	Technical Advisory Group
TBM	test blanket module
TBWG	test blanket working group
TCV	Tokamak a Configuration Variable, Switzerland
TEM	trapped electron mode
TEXTOR	Torus Experiment for Technology Oriented Research, Jülich, FRG
TF	toroidal field
TFTR	Tokamak Fusion Test Reactor Experiment (former PPPL tokamak)
TGLF	trapped gyro-Landau fluid
TokSys	tokamak system toolbox
TORE-SUPRA	tokamak at Cadarache, France

TQ	thermal quench
TRANSMAC	commercial software to interface PC and MAC computers
TRINITI	Troitsk Institute for Innovation and Thermonuclear Research, Moscow
TS	Thomson scattering
TS1792	Union Carbide graphite
TST	Tokyo Spherical Tokamak
TTF	Transport Task Force
TVs	televisions
TWAs	traveling wave antennas
TWG	Technical Working Group
UC Berkeley	University of California, Berkeley
UC Davis	University of California, Davis
UC Irvine	University of California, Irvine
UCLA	University of California, Los Angeles
UCSD	University of California, San Diego
UKAEA	United Kingdom Atomic Energy Authority
UNIX	computer operating system
U.S. or US	United States
USBPO	U.S. Burning Plasma Organization
UT Austin	University of Texas at Austin
UVC	Universal Voltronics Corporation
UW	University of Wisconsin
V&V	verification and validation
Vdc	voltage – direct current
VDEs	vertical displacement event
VFI	vertical field inductor
VLT	Virtual Laboratory for Technology
VoIP	voice over internet protocol
VRVS	Virtual Room Videoconference System
VSWR	voltage standing wave ratio
VUV	vacuum ultra violet

APPENDIX B
LIST OF CODES

Code	Purpose
ATLAS	stochastic magnetic field topology
BALOO	MHD ballooning stability
BOUT	edge turbulence transport
C2	transport simulation
CAMINO	MHD ballooning stability
CERAUTO	automated CER analysis code
Condor	an open source queuing system for high throughput computing
CORSICA	transport simulation
CQL3D	heating and current drive
CRONOS	transport simulation
CURRAY	rf ray tracing
DCON	MHD ideal stability
DEGAS	Neutral transport
DIVIMP	edge impurity transport
E3D	3D Monte-Carlo heat transport code
EFIT	equilibrium reconstruction
EGK	turbulence simulation
EIRENE	neutrals transport
ELITE	edge MHD stability
EMC3-ERIENE	3D edge transport with neutrals
EMC3	3D edge transport
GATO	MHD ideal stability
GEM	turbulent transport
GENRAY	rf ray tracing
GKS	turbulent transport
GLF23	turbulent transport
GS2	turbulent transport
GTC	turbulent transport
GTNEUT	neutral transport
GYRO	turbulent transport
IDL	data display and analysis
IMFIT	integrated modeling
IPEC	error field response
KPRAD	radiation dynamics
M3D	ideal MHD dynamics
MARS	resistive MHD
MARS-F	extended version of MARS
MBC	MHD ballooning stability

MIST	impurity transport
NCLASS	neoclassical transport
NFREYA	neutral beam deposition
NIMROD	3D MHD simulation
NMA	resistive wall modes
NOVA	energetic particle instabilities
NOVA-K	energetic particle instabilities
NUBEAM	neutral beam deposition
ONETWO	transport simulation/analysis
ORBIT	particle orbits
ORBIT-RF	particle orbits in presence of RF
PELLET	pellet ablation
PEST	ideal MHD stability
PEST3	ideal MHD stability
PTRANSP	plasma simulation
RTEFIT	real-time equilibrium reconstruction
SOLPS	edge transport simulation
SOLPS5	edge transport simulation
SOLPS5-EIRENE	edge transport simulation
TEMPEST	edge turbulence
TEQ	equilibrium solver
TOQ	equilibrium solver
TORAY	rf ray tracing
TORAY-GA	rf ray tracing
TORIC	rf wave modeling
TRANSP	transport simulation
TRIP3D	3D field line topology
TSC	tokamak simulation
UEDGE	edge simulation/analysis
V3FIT	3D equilibrium reconstruction
VALEN	resistive wall mode feedback
VMEC	3D equilibrium reconstruction
XGC	edge transport
XGC-2	edge transport
XPTOR	transport simulation
ZIPFIT	between-shot profile analysis