GA–A23028
UC–420

A DECADE OF DIII–D RESEARCH

FINAL REPORT TO THE U.S. DEPARTMENT OF ENERGY

for the period of work
OCTOBER 1, 1989 THROUGH SEPTEMBER 30, 1998

by
PROJECT STAFF

DATE PUBLISHED: MARCH 1999

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1. INTRODUCTION

During the ten-year DIII–D tokamak operating period of 1989 through 1998, major scientific advances and discoveries were made and facility upgrades and improvements were implemented. Each year, annual reports as well as journal and international conference proceedings document the year-by-year advances (summarized in Section 7). This final contract report, provides a summary of these historical accomplishments. Section 2 encapsulates the 1998 status of DIII–D Fusion Science research. Section 3 summarizes the DIII–D facility operations. Section 4 describes the major upgrades to the DIII–D facility during this period.

During the ten-year period, DIII–D has grown from predominantly a General Atomics program to a national center for fusion science with participants from over 50 collaborating institutions and 300 users who spend more than one week annually at DIII–D to carry out experiments or data analysis. In varying degrees, these collaborators participate in formulating the research program directions. The major collaborating institution programs are described in Section 6.

1.1. HISTORY AND ACCOMPLISHMENTS OF THE DIII–D PROGRAM

The DIII–D tokamak program at GA has made many major scientific contributions to the worldwide fusion effort. The DIII–D Program pioneered plasma shaping and profile control as a means of improving performance, leading to second stable high beta core plasmas in DIII–D. Confinement has improved, particularly in discharges with optimized magnetic shear, with the energy confinement time reaching four times that of the standard ITER–89P scaling. Pioneering programs in electron cyclotron and fast wave heating and current drive have made progress in developing and demonstrating the understanding necessary to sustain the conditions of optimized shear. New and effective divertor geometries were devised, leading to the divertor configurations widely used today and projected for future fusion devices. The program continues to advance on a broad front, with major contributions in transport, stability, divertor physics, and in RF heating and current drive. The hallmark of the DIII–D Research Program is the integration of these science research topics into a program aimed at optimization of the AT. A ten-year chronology of DIII–D physics advances and facility improvements is shown in Fig. 1–1.

Plasma Physics Advances	Year	Facility Improvements
H–mode Plasma 6.8% beta 2.5 MA Divertor 4.6 s H–mode	1988	Glow clean between shots Install IBW antenna
8% beta 13 keV ion temperature 2.6 MA plasma 9.3% beta 3 MA double-null divertor	1989	Neutron shield complete n = 1 coil installed 19.6 MW NBI achieved
10.3 s H–mode NCS discovered 17 keV ion temperature electron heat pinch	1990	MSE operational
VH–mode discovered 11% beta	1991	1 MW FWCD operational Bias ring operational Boronization completed 1.6 MW ICH achieved
FWCD demonstrated τ_E = 0.42 s ECCD demonstrated	1992	110 GHz into plasma 8 Laser Thomson operating
Density control with divertor pump High triangularity studies τ_{He} / τ_E ~10 to 15 12.5% beta	1993	Divertor pumping operating DiMES 32-channel ECE operational
21 keV ion temperature Improved confinement NCS $\beta\tau$ = 1.7% s Understanding VH–mode Measured j(r) profile routinely	1994	Operating C–coil Pellet injector installed
BES data Record neutron rate 2.4×10^{16} s^{-1} $Q_{DD} = 1.5 \times 10^{-3}$	1995	0.5 MW, 110 GHz ECH Fast stroke divertor Langmuir probe 8-channel MSE 6 MW fast wave ICRF installed
1.5 × Greenwald 10 keV electron temp. with ECH Understanding divertor heat flow	1996	Divertor Thomson Data 1 MW, 110 GHz gyrotron
MHD mode wall stabilization Recombining divertor Divertor flow	1997	Lithium pellet injected Upper cryopump and baffle Remote operation from LLNL 32-channel MSE
Transport studies 5 s Transport barriers Off axis ECH current drive	1998	Two 110 GHz gyrotrons RWM saddle loops

Fig. 1–1. Ten-year chronology of DIII–D plasma physics advances and facility improvements.

1.1.1. ORIGIN OF THE PROGRAM

The GA 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.

The success of these experiments led to construction of the Doublet III device with its first plasma on February 25, 1978. In the first years of operation, it was the largest operating tokamak in the world and attained the highest current levels recorded at that time (2.2 MA). Experiments with a range of plasma configurations demonstrated the importance of elongation and shape control. Dee-shaped plasmas proved easiest to form and were projected to reach beta 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 3 MA and achieved superior levels of confinement and beta. Understanding of plasma stability, transport, divertor and current drive physics was developed.

The DIII–D Program has contributed outstanding results in most major areas of tokamak physics including: confinement, stability, boundary physics and technology, rf heating and current drive, and tokamak operations. This progress is typified by the evolution of the fusion triple product $n\tau T$. Over the last ten years, the performance has doubled every two years, reaching 7×10^{20} keV-s m^{-3}. In the following sections, we summarize the scientific progress relative to our previous five-year plan.

1.1.2. DIII–D SCIENTIFIC ACCOMPLISHMENTS

The First Five-Year Plan: 1989–1993. The 1989–1993 research plan encompassed four major tasks: (1) beta limit research, (2) confinement physics research, (3) rf heating research, and (4) current drive research. These tasks built on existing GA capabilities and available and future facility capabilities. Together, the four tasks contributed toward achieving the ultimate goal of the 1989–1993 research: the demonstration of high beta tokamak plasma with good confinement and completely noninductive current drive.

A central element of stability research was to establish the understanding of beta limits in noncircular plasmas. The stability boundaries were studied and mapped out and extended from previous 4.8% to 12.5% beta. These very favorable findings resulted in major changes in the European and Japanese research tokamaks as well in the design of future devices such as CIT, BPX, and ITER.

Confinement physics research concentrated on improvement by control of edge plasma conditions. Through vacuum wall improvements and conditioning as well as divertor modifications, the confinement quality was doubled through H–mode edge transport barrier and then doubled again through a VH–mode core transport barrier. These confinement improvements are shown to be related to plasma E×B flow shear which theorists were able to show would destroy and/or stabilize turbulent eddies causing energy transport.

During the first five-year period, rf heating research was constrained by the availability of higher power rf systems due to a vendor's technical development difficulties as well as programmatic budgetary constraints. These difficulties have subsequently been over come and this program element is now proceeding well. With the limited rf power levels available, the physics of heating and current drive were investigated and benchmarked with theoretical models.

Active particle control was initiated using a divertor pump and divertor diagnostic instruments are continually improved. A major collaborative advanced divertor program was initiated between GA, LLNL, ORNL, SNL, UCLA, and UCSD. Particle control was demonstrated, divertor plate heat flux was demonstrated, and extensive comparisons between measurements and modeling was initiated. This research formed the basis for several world tokamak divertor modifications.

The Second Five-Year Period: 1994–1998. The 1994 DIII–D Scientific Program was organized under two themes: AT and Divertor Development and Research Programs which were connected through an ultimate theme: Integrated AT Research.

AT Research. The 1994 AT Program Plan had three major research goals: *"(1) to develop physics understanding of the formation and sustainment of AT configurations; (2) to establish experimental validation of the physics of active rf current drive and efficiency optimization; and (3) to combine these two to provide a demonstration of optimized, long-pulse AT operation with simultaneous improved confinement, enhanced stability, and fully noninductive current drive at high beta."* These 1994 program goals were proposed to be accomplished through a number of studies to accomplish the objectives outlined in Table 1–1. Excellent progress was accomplished on all these objectives, except for the long pulse studies requiring the future high power microwaves upgrade system. As characteristic of research, results of these studies opened unexpected new opportunities and deeper scientific questions to be investigated in the future. Overall, the scientific progress was more or less as envisioned while the ITER support was more than envisioned in 1994.

TABLE 1–1
AN ASSESSMENT OF PROGRESS ON THE 1994 ADVANCED TOKAMAK RESEARCH GOALS
[A check (✓) indicates progress as anticipated, a minus (–) indicates less progress than anticipated, and a plus (+) indicates more progress than anticipated]

Goal	Progress
• Develop understanding of Advanced Tokamak regimes	✓
• Validate noninductive current drive (bootstrap utilized, fastwave to 0.3 MA, electron cyclotron by 1998)	✓
• Actively control, optimize, and demonstrate Advanced Tokamak regimes	
— Short pulse	✓
— Long pulse	–
• Stability and transport theory/experimental interaction	✓
• Disruption studies	✓
• Rotation effects with C–coil	✓
• Develop advanced (digital) plasma control	✓
• Provide ITER physics simulation	+

A measure of integrated performance is the fusion triple product; $n_i \tau_E T_i$. This product is composed of the central fusion fuel ion density, the global energy confinement time and the central fusion fuel ion temperature. Over the past five years, the DIII–D fusion triple product has increased four fold. This progress over the history of the DIII–D facility, shown in Fig. 1–2, is comparable to that of rapidly advancing activities such as the semiconductor computer industry. In the next five-year period, we do not intend extensive dedicated campaigns to push the triple product higher since these are very hot ion conditions. Instead, future research will concentrate on plasma regimes with near equal ion and electron temperature as characteristic of future fusion power plants.

A second measure of integrated fusion performance is by direct measurement of the fusion power produced normalized to the required plasma heating power. In the past five years, significant progress has been made through strongly shaped double-null plasmas, NCS plasma current profile, and broad plasma pressure profile. In this way, 28 kW of D–D fusion output power was produced in DIII–D with 18 MW of neutral beam input power. The D–D fusion gain was QD–D = 0.0015. For an optimum D–T fuel mixture, this is equivalent to a D–T fusion gain of Q_{DT}^{equiv} = 0.3. Compared to five years ago, this is an increase of a factor four.

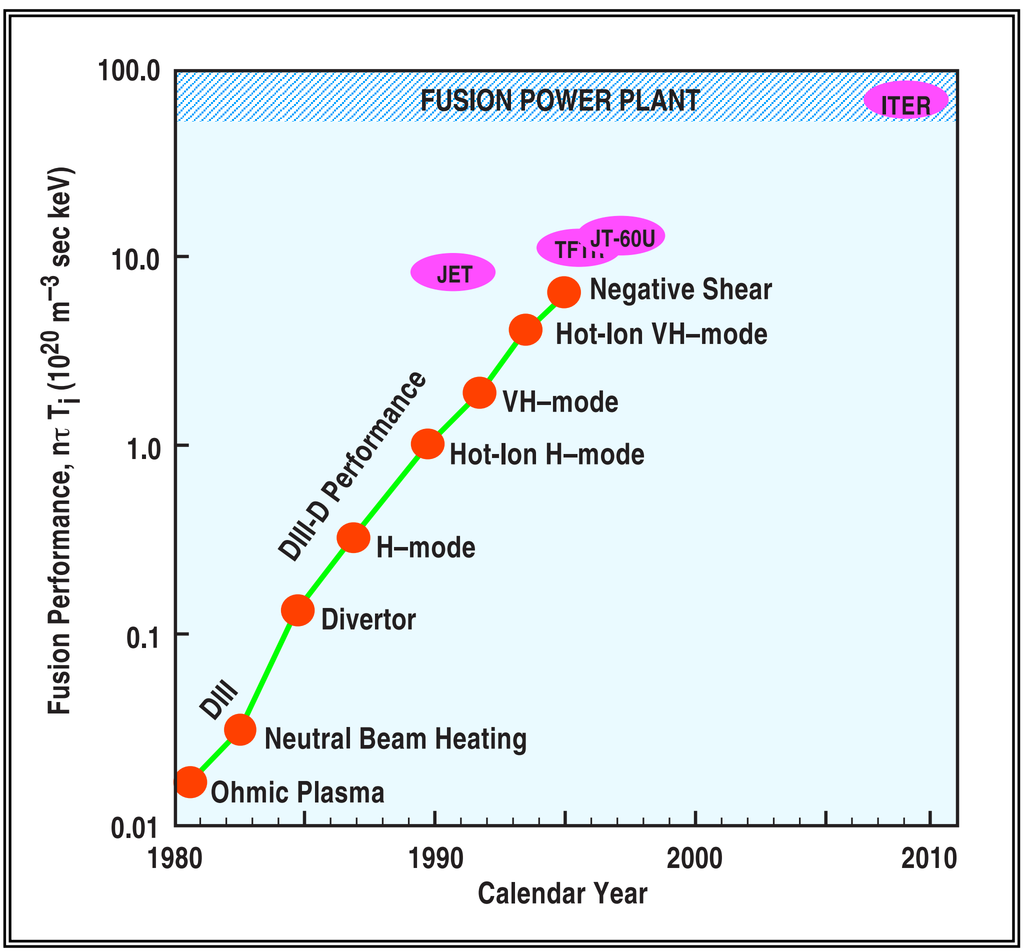

Fig. 1–2. Scientific progress: DIII–D fusion performance has doubled every two years.

1.1.3. DIII–D SCIENTIFIC PROGRESS AND ACCOMPLISHMENTS

1.1.3.1. Plasma Stability and High-Beta Physics. The DIII–D Program has made remarkable progress in understanding the nature of the β limit in tokamaks and in achieving higher values of the plasma β. Understanding of the stability of elongated discharges has led to operation with double-null discharges with high values of I/aB and the achievement of β = 13%. The associated development of detailed theoretical understanding, demonstrates that β values needed for power plant operation are credible and achievable. Accurate equilibrium calculations have demonstrated that regions of the plasma reach into the second-stable region. Understanding the role of the current profile in establishing β limits has led to the recognition that the limiting β value could be raised with properly optimized plasma profiles.

At high values of beta, self-driven neoclassical bootstrap currents become a significant contribution to the overall plasma current. This is beneficial for obtaining steady state discharges, but it can also lead to MHD instability. This coupling between the self-driven current and the plasma pressure, which establishes the stability limits, is referred to as neoclassical MHD. If finite size island structures form, the plasma pressure gradients flatten within the islands which causes the bootstrap current to weaken and the island to expand. This process has been shown to establish beta limits in plasmas in DIII–D and other tokamaks. Recent DIII–D experiments have compared this instability to theoretical models, and means of stabilizing the modes using localized currents driven by electron cyclotron current drive are being developed.

Important progress also has been made in understanding locked modes. These occur when rotating magnetic modes lock (to a stationary, local magnetic field asymmetry), often leading to a disruption. Locked modes lead to limitation of the operating space. Particularly, they restrict operation with the low-density target plasmas that are crucial for obtaining efficient rf current drive and VH–mode confinement. Experimentally, it has been shown that an external perturbation can be added to the tokamak field configuration to minimize intrinsic local asymmetries and, thus, substantially increase the operating space.

1.1.3.2. Plasma Confinement Physics. Early operation of the DIII–D device led to the routine attainment of the high confinement regime, the H–mode. DIII–D H–mode results show a strong increase in energy confinement τ_E with plasma current, consistent with worldwide tokamak results. DIII–D has made significant contributions toward understanding the physics of the transition from L–mode to H–mode. Improvements made in the DIII–D charge exchange recombination spectroscopy system provided important data on the change in the edge radial electric field across the transition. By measuring edge poloidal and toroidal rotation, temperature, and density of various plasma ions, our experimental data showed that the radial electric field changes just before the start of the transition, and that density fluctuations change right at the start of the transition in a localized layer where the radial electric field also changed. This is also the region where gradients of plasma density and temperature steepen after the transition indicating a decrease in local transport. Theory predicts that the increased shear in the E×B drift velocity leads to this transport reduction. A similar effect explains the VH–mode, with a broader region of E×B shear extending further into the plasma.

In the last several years, further confinement improvement has been obtained in the core of DIII–D plasmas by optimizing the magnetic and E×B shear. The initial signs of the improvement were the creation of obvious core transport barriers in discharges where manipulation of the current density profile had resulted in suppression of sawteeth. Core barriers have been formed in discharges with both positive and negative magnetic shear. The key factor in all these plasmas appears to be the same E×B decorrelation of turbulence that is operational in the plasma edge in H–mode and VH–mode. By optimizing the plasma pressure profile with suitably timed L-to-H transitions, we have created plasmas where the whole discharge has low transport. For example, ion thermal diffusivity at or below the standard neoclassical level has been attained across the whole plasma. Such plasmas have a DIII–D record triple product $n\tau T = 7 \times 10^{20}$ m^{-3} s keV. These discharges demonstrate that control of plasma current and pressure profiles can lead to significant confinement improvement over standard H–mode.

As part of the work on confinement improvement, a significant amount of work has been done on basic studies of local transport. These include assessment of local transport coefficients through power balance and perturbative approaches and comparison of coefficients for dimensionally similar discharges and off-axis heating experiments where the possible existence of a heat pinch term is indicated in experiments using either ECH or NBI. One of the key problems in thermal transport analysis is separation of the electron and ion thermal transport. The individual values are rendered uncertain by the uncertainty in the electron-ion power transfer term, which can depend on the difference of large numbers. The ability to separately heat the ions (with NBI) and the electron (with fast wave and ECH) has allowed us to reduce this uncertainty in discharges with combined heating because plasma can be made in which the uncertain power transfer term is actually a small component of the heat input to either species.

As part of the transport work, we have been actively involved in the ITER process, providing a significant amount of data to the ITER global confinement database. This has been combined with the data from tokamaks world wide to furnish means of predicting the confinement values in ITER. In addition, we have been one of the major players in the area of confinement investigations using nondimensional scaling. Indeed, it is probably only because of the careful nondimensional transport work done on DIII–D that this technique has been recognized as a reliable means of transport investigation. Because of DIII–D flexible shaping capabilities, we can match the plasma shape of other, less flexible machines, thus providing data for key tests of this technique.

In the past two years, so-called theory-based models of local transport have emerged, which have had some success in matching experimentally measured profiles from various machines. DIII–D experimentalist and theorists are actively involved in this work and DIII–D profile data makes up a significant part of the ITER profile database, which is being used to test these various theories. We are working with the whole transport community in testing and attempting to improve these models.

1.1.3.3. Boundary Physics and Technology. DIII–D work in boundary physics and technology has concentrated on understanding and developing the divertor configuration, including the demonstration of long pulse discharges. Divertor configurations similar to those developed on DIII–D have become prototypical for next-generation tokamaks and stellarators, including ITER. Research on DIII–D has led to the development of the lower advanced divertor configuration and the upper high-triangularity divertor presently operational in DIII–D. Important work also has been done on vessel wall conditioning and the transport of impurities from the plasma edge.

Early studies of the heat loads to divertor targets led to the recognition that these loads can be strongly peaked and that the heat distribution can depend on the confinement mode (ohmic, L–mode, H–mode). These studies also quantified the differences between single-null and double-null discharges, and showed that the heat loads in double-null discharges could be maintained as essentially up/down symmetric. It was then demonstrated that the heat loads could be managed by sweeping the location of the divertor strike point across the target plate.

Results from DIII–D also show that by injection of gas at the plasma boundary, both peak and overall heat load to the divertor target plates can be reduced to one-fifth of the original value. Puffing of deuterium gas reduces the peak heat flux to the divertor tiles and at the same time — Z_{eff} is constant or slightly reduced. These results show that in DIII–D, gas puffing is a promising method of reducing heat load to the divertor. These results are an encouraging proof of principle of the radiative recombining divertor concept.

The desire to better optimize and control the divertor configuration and, in particular, demonstrate density control in divertor H–mode plasmas led to the conception and implementation of the advanced divertor configuration in conjunction with a team of collaborators from GA, LLNL, ORNL, SNL, and UCSD. This involved the installation of a cryogenic pump and a biasable ring near the divertor X–point to pump neutrals and to allow an electric field to be applied to the plasma in the divertor region. The application of a bias voltage to the ring electrode was shown to result in a further increase in the divertor pumping. An extensive set of diagnostics was added to the lower divertor including bolometric tomography, spectroscopy, and a divertor Thomson Scattering System. These new diagnostics showed that the divertor was cold, 1 to 2 eV indicating that plasma radiation, recombination, and convective power flow are dominant processes in the radiative divertor region.

Successive improvements in the wall condition of the DIII–D device have led to remarkable improvements in both confinement and impurity level. The DIII–D vacuum vessel was constructed with the capability to bake to nearly 400°C for the purposes of decontaminating the graphite vessel wall in preparation for plasma discharges. Boronization, the *in situ* coating of the vessel walls with a thin layer of boron, has resulted in a substantial improvement in discharge operation, especially with high current, high energy plasmas. Discharges of 3 MA achieved a plasma energy of 3.6 MJ, $\langle\beta\rangle = 5.1\%$ at full toroidal field, thus fulfilling a long-standing DIII–D program goal of reaching high β at full plasma parameters. Boronization resulted in the discovery of the VH–mode, a confinement regime substantially improved from those previously achieved with confinement.

The poloidal location of the advanced divertor is optimized for pumping low-triangularity single- or double-null divertors. As noted above, many of the AT scenarios involve high triangularity plasma shapes. We recently installed the first phase of a high triangularity double-null divertor configuration consisting of an upper baffle and cryopump for density control in high-triangularity plasma shapes. This hardware is currently operational, and we have demonstrated density control in high-triangularity upper single-null plasmas.

1.1.3.4. RF Heating and Current Drive Physics. DIII–D RF heating and current drive research has investigated the use of electron cyclotron waves and fast waves in the ion cyclotron range of frequencies to heat electrons and to drive plasma currents. ECH has the advantage of easy coupling of the power to the plasma with simple antenna structures and localized deposition of the power in the plasma. Initial ECH experiments on DIII–D utilized a frequency of 60 GHz — the fundamental frequency at the maximum field of DIII–D. This system, while effective at localized heating of the plasma, was limited by the fact that coupling is cut off above relatively modest densities and that the unit size available for the power generation system (200 kW each) is small for high power experiments. Recently, two 1 MW sources at

110 GHz have been installed for heating at the second harmonic in order to address both of these issues. This system, has steerable mirrors to direct the power deposition at any minor radius along the resonance.

Experiments on DIII–D at 60 GHz have used all of the principal modes of ECH, including outside launch of the ordinary mode (O–mode) at the fundamental frequency, outside launch of the extraordinary mode (X–mode) at the fundamental and second harmonic, and inside launch of the fundamental X–mode. Propagation limits and absorption are well predicted by theory models. Central electron temperatures of 10 keV have been achieved with ECH. ECH provides the capability to increase the electron temperature closer to the ion temperature as will be the case eventually in burning plasmas. In addition to bulk heating, ECH has potential applications affecting confinement and stability. In DIII–D experiments, application of ECH has been shown to generate the H–mode of improved confinement, which is widely regarded as a test of the ability of a technique to heat without introducing significant levels of impurities; heating near the $q = 1$ surface has suppressed sawteeth; and applying ECH with the resonance near the edge in H–mode discharges with substantial neutral beam heating has stabilized the Edge Localized Modes, leading to improved energy confinement.

The ECH System is a key tool in performing critical experiments required for understanding and optimizing the tokamak concept. ECH is a unique heating technology in that the energy is coupled to the electrons in a localized spatial region of the plasma. This makes ECH a unique tool for studying transport through application of localized heat. In contrast to the predictions of standard techniques used to model energy transport in the tokamak, a series of careful experiments utilizing these capabilities revealed an anomalous inward flow of electron heat. Discovery of this heat pinch has stimulated new theoretical activity in the community which will hopefully be the key to unlock the puzzle of electron thermal transport.

ECCD experiments have been carried out at 110 GHz. Extensive data analysis and modeling showed that about 170 kA of current was driven by 1 MW of power and that this is consistent with predictions.

Fast waves are also useful for electron heating and current drive. The fast waves are launched from the low field side of the plasma with a toroidal velocity which is close to the thermal speed of the electrons. This results in moderately strong electron Landau damping and transit time magnetic pumping which heats the electrons and drives current. Fast wave heating heats the high temperature center of the plasma preferentially. Good fast wave absorption requires high electron temperature and, thus, the Fast Wave Program is symbiotically linked to the ECH Program to achieve effective central heating and current drive. Strong central heating in discharges heated by neutral injection has also been observed. Pick-up loops on the vessel walls and reflectometer measurements are used to study wave propagation and absorption.

More recently, FWCD has been shown to be an effective method of electron heating and driving plasma current in the plasma core. The FWCD Program on DIII–D is a collaborative effort between GA and ORNL, which has provided a proof-of-principle demonstration of FWCD for application to DIII–D and other tokamaks. To avoid competing absorption mechanisms, such as absorption at ion cyclotron harmonics, we seek to maximize the single-pass absorption by first heating the electrons with ECH or neutral beam power.

FWCD experiments have led to record current drive by this means, about 290 kA. The magnitude of the driven current and its radial profile are in good agreement with theory. FWCD has been applied to discharges with NCS, where it has been effective at modifying the current profile and prolonging the duration of the negative shear phase. FWCD thus is a valuable tool, along with ECCD and NBCD, for controlling the shape of the current profile.

1.1.4. DIII–D OPERATIONS AND FACILITY IMPROVEMENTS

The DIII–D Research Program requires safe and reliable operation of the tokamak in new plasma configurations, refurbishing and improving the tokamak facility, and meeting the needs of expanding numbers of collaborators.

1.1.4.1. Operations. During the past ten-year period, scientific progress was paced largely by funding availability. As seen in Table 1–2, funding was generally below the plan. In order to maintain an appropriate operation level in the face of the facility fixed costs, it was necessary to defer upgrades, refurbishments, and normal procurement to the 1999 five-year period.

TABLE 1–2
COMPARISON OF TEN-YEAR PLAN AND ACTUAL
GA FUNDING LEVELS AND OPERATIONS WEEKS

	FY89	FY90	FY91	FY92	FY93	FY94	FY95	FY96	FY97	FY98
Planned GA Funding ($M)	31.5	39.5	48.0	52.3	38.1	40.7	38.3	38.6	42.3	49.1
Actual GA Funding ($M)	32.3	32.1	31.0	32.0	36.5	38.2	39.0	34.0	30.7	37.5
Weeks Operation	27	17	20	14	16	11	19	16	9	18

1.1.4.2. Tokamak Facility Improvements. In addition to the major upgrades described in Section 3.2.3.2, the 1994 DIII–D Five-Year Plan envisioned implementation of a number of facility improvements listed in Table 1–3. Although the computer capability was increased, the improvements were inadequate to handle the unanticipated large increase in data, so aggressive effort was undertaken in FY98.

1.1.4.3. New Diagnostics. New plasma diagnostics are needed to carry out the Research Program. Twenty-three new diagnostic instruments were installed in the past five years (see Table 1–4).

1.1.4.4. Data Acquisition and Analysis. In 1994, we anticipated that the data collected per shot would increase from 70 to 120 MBytes of data collected in 1998. Already in 1997, up to 220 MBytes is being collected on each shot (280 MBytes in early 1999). This larger amount of collected data is a result of increased sophistication of GA and collaborator diagnostic instruments, number of channels, and data collection rate. This increase in collected data has overloaded the computer and data retrieval systems. While the data acquisition and analysis computer systems have been somewhat improved, they are presently significantly under powered. Efforts are on-going to close this gap over the next few years with both new hardware and new data analysis software tools.

TABLE 1–3
FACILITY IMPROVEMENTS IMPLEMENTED IN FY94–98

Real time computer plasma control system (GA)
C–coil for magnetic error field correction (GA)
Pellet fueling (ORNL)
Operation of experiments from remote site (LLNL)
Increased computer capability (GA)

TABLE 1–4
NEW DIAGNOSTICS INSTALLED SINCE 1993

Current profile diagnostics
Motional Stark effect edge upgrade (LLNL)
X-ray spectrometer radial array (Russia) — one channel operational
Additional magnetic probes (GA)
Fluctuation diagnostics
Beam emission spectroscopy (edge and central) (U. Wis./GA)
Improved microwave reflectometry, scattering (UCLA)
Phase contrast imaging (MIT)
Core plasma diagnostics
Charge exchange recombination upgrade (GA)
Superheterodyne electron cyclotron emission (ORNL/U. Texas)
Direct E_r MSE (LLNL)
2D BES T_e Fluctuation (U. Wisc.)
Disruption Tile Current Arrays (GA)
Toroidal and poloidal asymmetries
Infrared and visible TV cameras (LLNL)
Upper divertor Langmuir probes and pressure gauges (GA, ORNL, SNL)
Divertor impurity transport and radiation
Multichannel divertor spectroscopy upgrade (GA)
Fast impurity gas injector (impurity pellet) (ORNL)
Normal incidence spectrometer (GA)
Divertor SPRED (LLNL)
EUV Spectroscopy (LLNL)
High Res. Bolometer (GA)
Flow Measurement (SNL)
Basic scrapeoff layer parameters
Reciprocating divertor probe (SNL/UCLA)
Divertor reflectometer (UCLA)
Divertor Thomson scattering (LLNL/GA)

1.1.5. TRANSITION TO A NATIONAL PROGRAM

During the past ten-year period, DIII–D has become a true National Program. Already in 1993, DIII–D operated with national and international participation from 30 institutions. Now there are 50 institutions collaborating and now two-thirds of the DIII–D research physicists are from institutions other than GA. Approximately, 300 scientific users spend more than a week at the DIII–D National Facility.

DIII–D has a Program Advisory Committee which meets regularly, and an Executive Committee composed of leaders of the major collaborating institutions. The program has unified technical and budget DOE quarterly reporting and reviews. Collaborators have chaired the 1996 and 1997 Research Planning Committees and collaborators now act in line management and project management roles. Needless to say, collaborators also lead experiment campaigns, are experiment session leaders, and represent the program at a wide range of meetings and other scientific forums. DIII–D has become even more of a national facility than was envisioned five years ago. This was due to a combination of a desire by the earlier collaborating institutions and due to a shrinkage in the number of other tokamak facilities as a result of the decline in fusion funding in FY96.

2. FUSION SCIENCE

2.1. 1998 STATUS OF DIII–D ADVANCED TOKAMAK RESEARCH

2.1.1. RESULTS FROM THE DIII–D SCIENTIFIC RESEARCH PROGRAM[1]

The DIII–D research program is aimed at developing the scientific basis for advanced modes of operation which can enhance the commercial attractiveness of the tokamak as an energy producing system. Features that improve the attractiveness of the tokamak as a fusion power plant include: high power density (which demands high $\beta = 2\,\mu_o\langle P\rangle/B^2$), high ignition margin (high energy confinement time τ_E), and steady state operation with low recirculating power (high bootstrap fraction), as well as adequate divertor heat removal, particle and impurity control. This set of requirements emphasizes that the approach to improved performance must be an integrated approach, optimizing the plasma from the core, through the plasma edge and into the divertor. Research results from DIII–D reported here include results from all these areas.

In the area of core physics research, we have demonstrated improved plasma performance and increased duration of the high performance phase in both H–mode and L–mode plasmas. Moving towards an eventual goal of fully non-inductive current drive, we have made the first tokamak demonstration of off-axis electron cyclotron current drive. It exhibits higher off-axis efficiency than previously expected theoretically. In edge physics research, we have established the role of the self-consistently generated edge bootstrap current in stabilizing ballooning modes and allowing edge second regime access. Edge pressure gradients more than a factor of two above the ballooning limit without bootstrap current have been experimentally measured. This has improved our understanding of the edge pedestal and ELMs, which affect both core and divertor performance. In addition, a physics model of the density limit has been tested on DIII–D which reproduces density limit results on present machines and scales favorably to larger devices. In the area of heat and particle control in the divertor, we have established a new understanding of convection and recombination in radiative divertor plasmas. Finally, we have enhanced the divertor radiation by plasma flows and impurity enrichment.

Progress Towards Integrated, Steady-State, Improved Performance Plasmas. In order to establish their future relevance for fusion, improved performance scenarios must demonstrate a path towards ultimate steady-state operation. This requires demonstrating that improved confinement plasmas can be sustained for long pulses at high beta values as well as developing the tools (e.g. current drive) which will be needed for steady state operation. DIII–D has carried out experiments in both these areas since the last IAEA.

Figure 2.1.1–1 demonstrates our recent progress in moving towards steady state improved performance discharges. In this figure, we measure our approach to steady state with $\tau_{duration}/\tau_E$, the duration

[1]Taylor, T.S., K.H. Burrell, D.R. Baker, et al., "Results From the DIII–D Scientific Research Program," presented at 17th IAEA Fusion Energy Conference, October 19–24, 1998, Yokohama, Japan, to be published in a special issue of Nucl. Fusion; General Atomics Report GA–A23007 (1998).

of the high confinement phase divided by the energy confinement time. We measure advanced tokamak performance through the product of normalized beta, $\beta_N = \beta\ (aB_T/I_p)$, and the confinement enhancement factor H relative to the ITER confinement scaling law. For H–mode, we will use H_{98y}, which is defined relative to the most recent scaling for thermal energy confinement time in ELMing H–mode [1]. As can be seen in Fig. 2.1.1–1, significant progress has been made in DIII–D in the past two years in advancing both quantities. The points describing recent shots are above and to the right of our earlier results, which is the direction of our advanced tokamak goal.

An example of such an improved performance discharge is shown in Fig. 2.1.1–2 [2]. Lines indicating the β_N and H_{98y} values required for ITER and the ARIES-RS reactor study are also shown, indicating that this discharge exceeds the ARIES-RS requirements on the β_N H_{98y} product. A β_N H_{98y} product exceeding 6 is sustained for 1 s (5 τ_E). The high performance phase of the shot in Fig. 2.1.1–2 was terminated at about 3 s by the initiation of a neoclassical tearing mode, probably triggered by an ELM.

Two approaches have been taken to improve plasma performance and duration as is illustrated in Fig. 2.1.1–3. Both utilize the technique of an early neutral beam injection during the current ramp that was developed over the past several years in producing core transport barriers in DIII–D [3,4], JET [5], JT60-U [6],and TFTR [7,8]. However, one approach [2] is more aggressive in pushing high power to reach high β_N while the second has emphasized more the long pulse aspects. Neither shot shows the rapid, localized change in temperature gradient characteristic of a strong, localized core transport barrier; however, transport analysis indicates improvement in ion thermal diffusivity over most of the discharge relative to standard ELMing H–mode [2].

Core ion transport barriers have been run for even longer durations in low current, L–mode edge discharges, as is shown in Fig. 2.1.1–4. These shots were specifically optimized for the full 5 second neutral

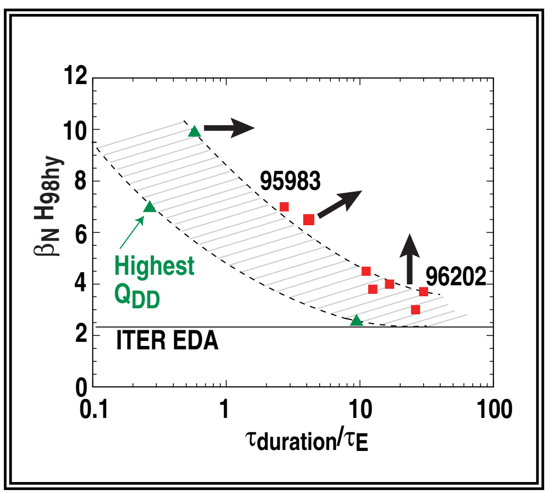

Fig. 2.1.1–1. Plot of the β_N H product versus normalized discharge duration for DIII–D shots from the time of the 1996 IAEA meeting (triangles) and more recent shots (squares). The shaded region shows our progress. All these discharges are H–modes; accordingly, the confinement enhancement factor H_{98y} relative to the H–mode scaling is used.

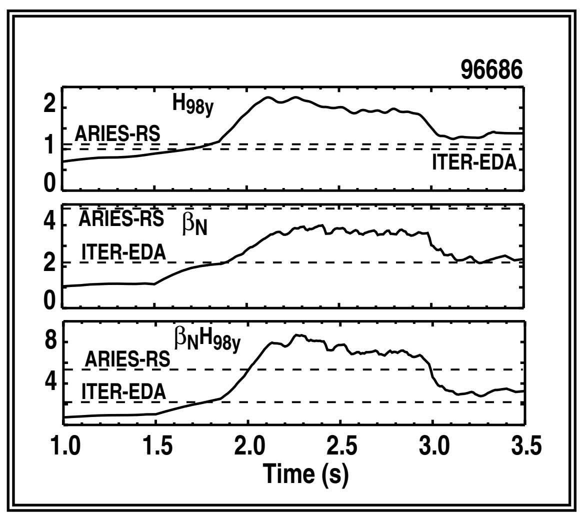

Fig. 2.1.1–2. Time evolution of a high performance DIII–D discharge. The 1 s duration is comparable to the current profile relaxation time scale. Some parameters of interest during the high performance phase are: β ~ 4.5%, n_e/n_{Gr} ~ 0.5, q(0) ~ 1, q_{95} = 4.4, τ_{th} ~ .21 s, and f_{bs} ~ 50%, where n_{Gr} is the Greenwald density and f_{bs} is the bootstrap fraction.

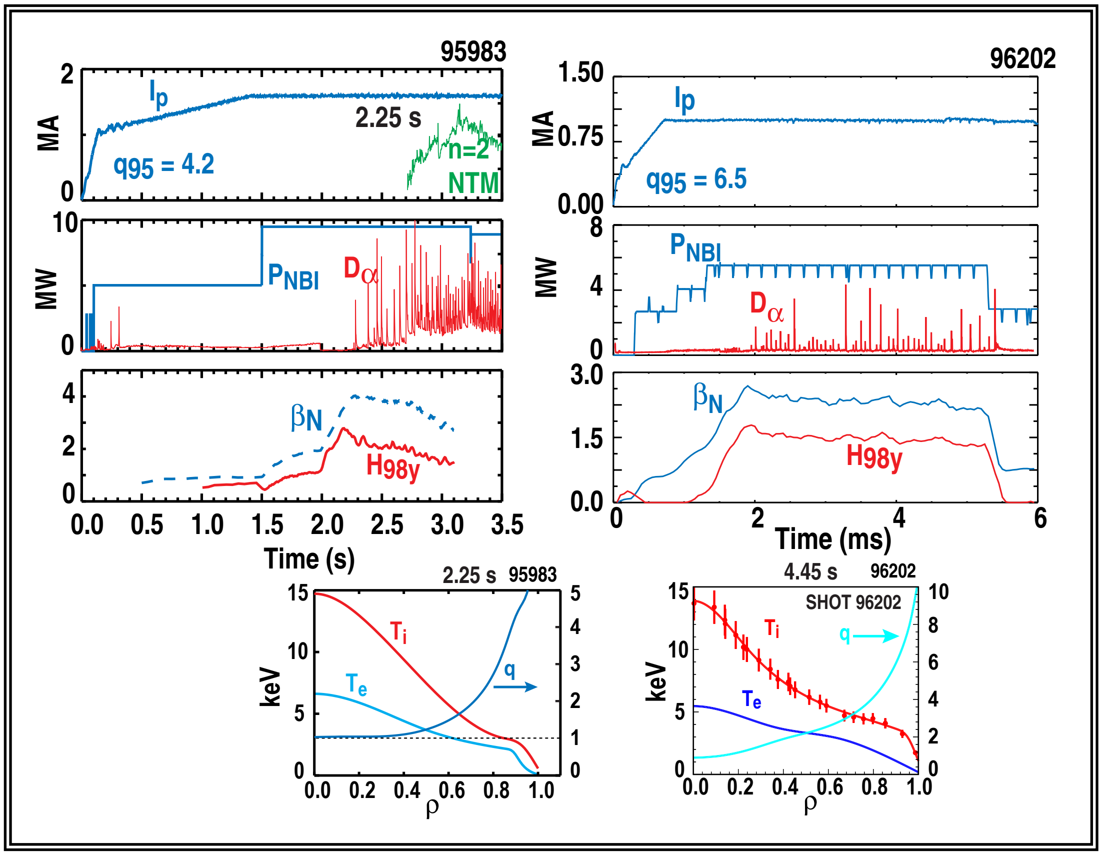

Fig. 2.1.1–3. Time histories and radial profiles of two recent DIII–D shots emphasizing long pulse high performance. Shot 95983 reached $\beta_N H_{98y}$ about 7 for 3 τ_E while 96202 achieved $\beta_N H_{98y}$ around 3 for 25 τ_E. (The corresponding β_N H products using the ITER89P L–mode scaling are 10 and 5, respectively.) The high performance phase of shot 96202 was terminated only when the neutral beam power was turned down while that of 95983 degraded due to the onset of neoclassical tearing modes. Shot 95983 was at 2.1 T toroidal field and had a line averaged density 5.4×10^{19} m^{-3} during the high performance phase while shot 96202 was at 1.9 T toroidal field and had a line averaged density of 2.7×10^{19} m^{-3}.

beam duration by running at low current and a relatively low density of 2×10^{19} m^{-3}. This discharge is an existence proof that it is possible to create an ion transport barrier which can last indefinitely.

A key feature in sustaining the good performance in the discharges in Figs. 2.1.1–3 and 2.1.1–4 is the absence of sawteeth. Detailed analysis of the current diffusion in these shots shows that this result is somewhat surprising, since neutral beam current drive alone should be enough to drive q(0) well below one. However, fast-particle driven MHD modes apparently broaden the neutral beam current drive profile, preventing this drop in q(0). In the discharges in Fig. 2.1.1–3, we observe fishbone oscillations while in the shot in Fig. 2.1.1–4, there are Alfvén eigenmodes present. Both of these modes are driven by fast particles and can redistribute these particles outward in radius.

Two major hurdles must be overcome in order to extend the discharges shown in Fig. 2.1.1–3 to higher performance and longer duration. First, as is shown in Fig. 2.1.1–3 (a), the performance in shot 95983 is degraded after 2.7 seconds by the onset of neoclassical tearing modes. This problem with neoclassical tearing modes is a common feature of many high performance discharges [2]. These modes are

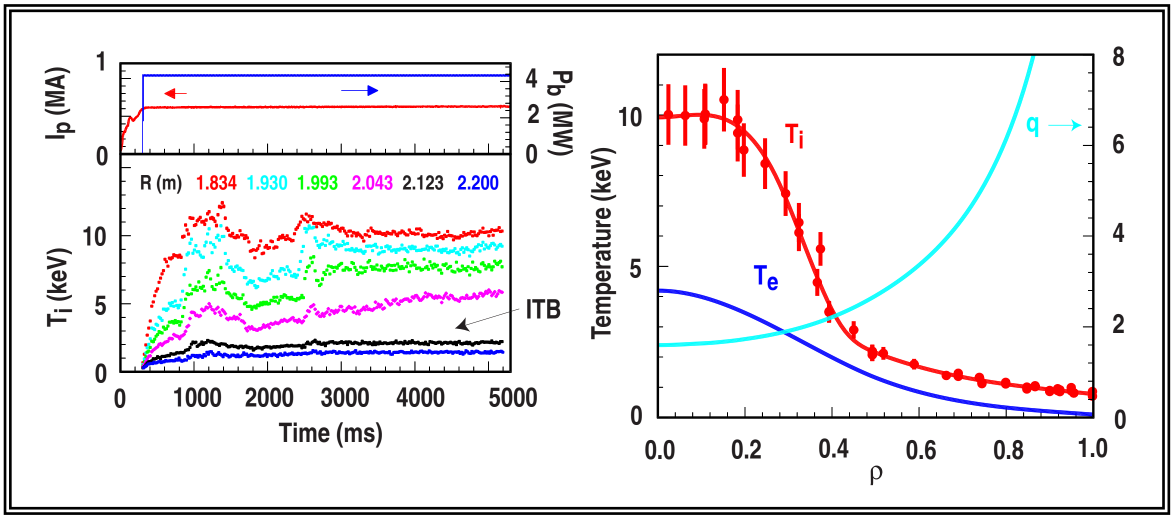

Fig. 2.1.1–4. Long pulse, L–mode edge discharge 94777 run at 1.9 T toroidal field. As is shown by the ion temperature time history, the core ion transport barrier lasts about 4.5 seconds in this shot, limited only by the duration of the neutral beam pulse. The major radius of the ion temperature measurements is listed at the top of the temperature figure. By the end of the beam pulse, even the q profile had essentially reached steady state, monotonic shape with q(0) = 1.6.

metastable, requiring a finite-size magnetic island to trigger instability. Finite-sized, seed islands can be triggered transiently, for example, by other MHD instabilities in the plasma, (e.g. sawteeth, ELMs or fishbones). As is shown in Fig. 2.1.1–5, the absence of sawteeth in shots like those in Fig. 2.1.1–3 removes one of the possible sources of seed islands for the neoclassical tearing mode and thus allows operation at a higher beta value.

The second hurdle is overcoming the effects of current diffusion so that q(0) remains above one, preventing destruction of the core transport barriers by sawteeth and removing this trigger of neoclassical tearing modes. Although sawteeth were not present in the shots in Figs. 2.1.1–3 and 2.1.1–4, the MHD oscillations which we believe broadened the beam driven current are undesirable from a performance standpoint. The measured fusion neutron rate in the shot in Fig. 2.1.1–4, for example, was about 1/3 of the value predicted assuming all the fast ions deposited near the axis slowed down where they were born. As is discussed presently, the electron cyclotron heating (ECH) systems now coming on line on DIII–D should allow us to confront both these hurdles through electron cyclotron current drive to both broaden the current profile and shrink the seed island.

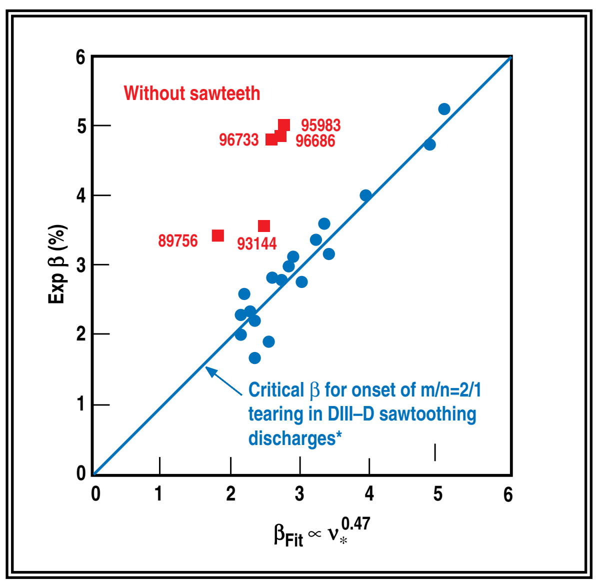

Fig. 2.1.1–5. The measured beta value for discharges similar to those in Fig. 2.1.1–2 (squares) significantly exceeds the neoclassical tearing mode limit established for sawtoothing discharges. The horizontal axis is the scaling value established in Ref. [9] while the line through the circular points is the best fit to the data for sawtoothing discharges. These neoclassical tearing modes are a mix of m/n = 3/2 and 2/1 cases.

Because of the need for current profile control for advanced tokamak operation, investigation of electron cyclotron current drive (ECCD) is a key portion of the DIII–D research. In the past year, we have demonstrated off-axis ECCD on DIII–D for the first time in any tokamak [10]. Electron cyclotron wave power at 110 GHz, which is resonant near the second harmonic of the electron cyclotron resonance, can be steered over a range of minor radii by tilting the launching mirror in the poloidal direction. The waves are given a toroidal velocity component so they interact with electrons traveling in a preferred toroidal direction, generating toroidal current. Analysis was carried out using motional Stark effect measurements of the internal magnetic field, allowing the local driven current density to be determined [11]. A 4-point vertical scan of the deposition location was made, covering the range of 0.1 to 0.5 in normalized minor radius ρ. Figure 2.1.1–6(a) shows the profile of ECCD which is driven at a ρ = 0.5 by 1 MW of electron cyclotron power. The integrated net current driven is 35 kA. The gross behavior of the plasma—the evolution of the internal inductance, the time duration before the entry of the q=1 surface into the plasma as signified by the start of sawteeth — is consistent with the effects expected from the measured current drive for the different locations of the power deposition. The magnitude of the driven current exceeds the value calculated by linear (TORAY) or quasilinear (CQL3D) codes. As is shown in Fig. 2.1.1–6(b), the theoretically predicted fall off in normalized efficiency with minor radius is not observed; the normalized efficiency at ρ = 0.1 and ρ = 0.5 are about the same. This result suggests that trapping of the heated electrons is much weaker than theoretically expected under the experimental conditions. These results strongly support the use of higher power ECCD as a means of sustaining current profiles with the optimized magnetic shear needed for advanced tokamak plasmas.

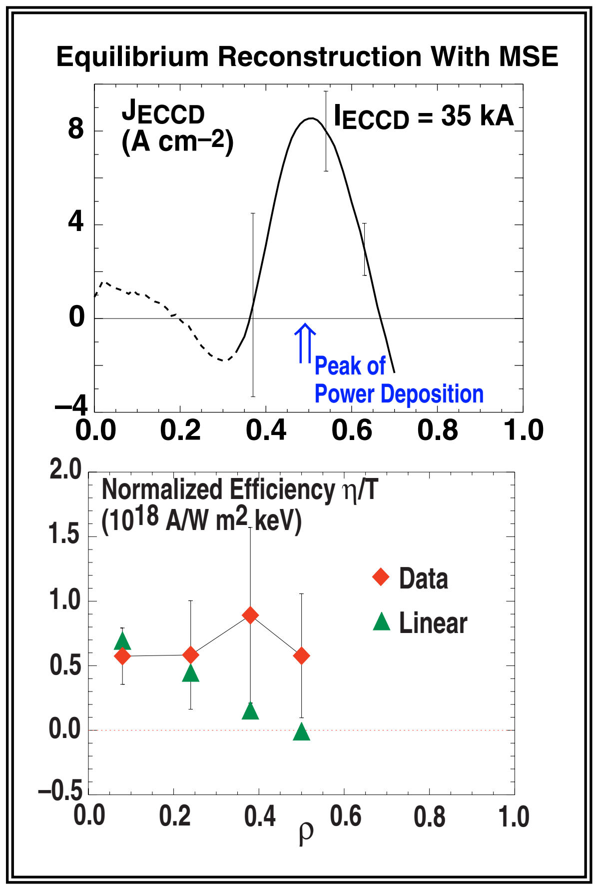

Fig. 2.1.1–6. (a) Profile of current density driven by ECCD for a case with power deposition at about half of the minor radius. (b) Normalized efficiency for ECCD as a function of the minor radius coordinate ρ. The current drive efficiency η has been normalized by the local electron temperature to remove the theoretically expected temperature dependence. The experimental results are compared to the linear TORAY calculations and show little decrease in normalized efficiency with ρ, contrary to theoretical expectations.

Progress in Understanding and Controlling Core Transport. In order to extend the improved performance results from present machines to future devices with confidence, we must finally develop a predictive understanding of tokamak transport. In addition, improved performance scenarios, especially in self-heated burning plasmas, will require development of new tools to control transport. Over the past two years, we have made progress in both understanding and control.

Over the past several years, fusion theorists have developed several new models of plasma transport [12-15]. Averaged over a large database of shots, each of these models do about equally well in predicting quasi-steady-state, equilibrium plasma profiles even though each model has a different mix of fundamental physics. Accordingly, to distinguish between models, some other test is needed.

Simulations have shown that perturbative transport experiments can provide a more critical test of transport models than equilibrium transport analysis. A perturbation source that deposits heat locally into the plasma particle species under study is preferred. Experiments have been performed on DIII–D using modulated ECH as the spatially localized perturbative heat source with the resonance absorption layer off axis. The electron and ion temperature responses are measured and the amplitude and phase of the perturbations (Fig. 2.1.1–7) and the equilibrium temperature profiles are compared to predictions from several transport models [16].

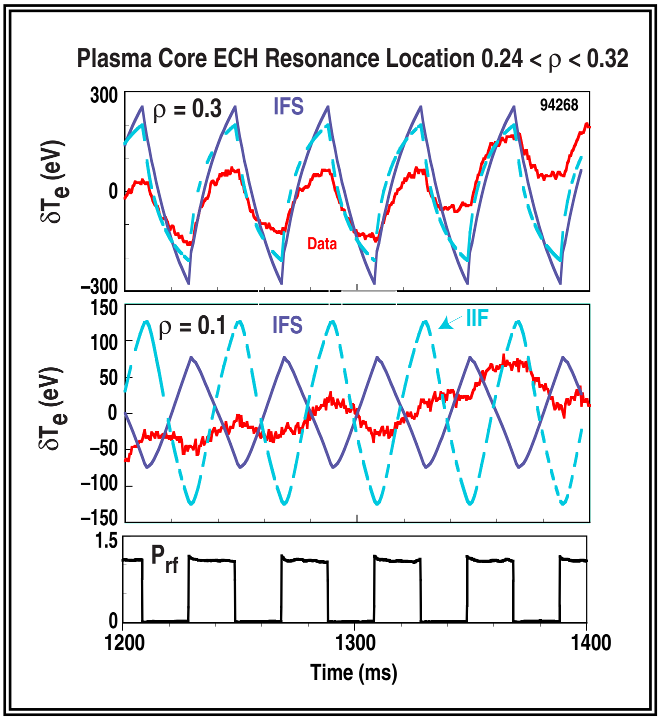

Fig. 2.1.1–7. Perturbed electron temperature, δT_e (eV), at ρ=0.3 and ρ=0.1 for measured data (solid, black lines), and simulated data from the IFS/PPPL model (solid gray lines) and IIF model (dashed gray lines). The ECH resonance location is at about ρ=0.3.

The results with off-axis heating indicate the electron and ion responses to the ECH perturbation are out of phase with each other at the plasma core and at the resonance layer. In general, the IFS-PPPL [12] and GLF23 [13] models predict reasonably well the ion response while the GLF23 and IIF [14] models do a reasonable job with the electron response. The GLF23 model includes the effects of both electron temperature gradient (ETG) and ion temperature gradient (ITG) driven turbulence as well as trapped particle modes, which may be why it fits the best overall. The GLF23 model fits the data best for the case with the ECH localized at ρ = 0.3; the comparisons for other heating locations were somewhat worse [16]. None of the models showed good agreement with both the ion and electron perturbative responses and the equilibrium profiles although the equilibrium profile fit of the GLF23 model was improved by including the effects of the measured, average E×B shear [16].

Although the creation of ion thermal and angular momentum transport barriers has been connected with E×B shear stabilization of turbulence both theoretically and experimentally [17–19], the physics governing the electron channel is much less well understood. Electron thermal transport barriers are much more difficult to form in DIII–D than ion barriers and seem to require much greater magnetic shear [20]. Electron heating with either ECH or fast waves has been used to probe the physics of core transport barriers [21,22]. For reasons that are not completely clear, central electron heating during the end of the core ion barrier formation phase tends to weaken the ion barrier, resulting in some reduction in core ion temperature and core ion rotation. This effect occurs only within the core barrier region with the ion profiles outside this region remaining unchanged by the additional electron heating. Both ion and electron thermal diffusivities increase after the application of the electron heating, with the electron diffusivity rising almost an order of magnitude [21]. The changes in the ion channel in these discharges are consistent with change in the E×B shearing rate relative to the low k turbulence growth rates [21]. The decreased ion rotation gives a decreased E×B shear while the growth rate changes little. However, the physics of the electron channel in these plasmas remains unexplained [21]. New FIR scattering measurements of short wavelength turbulence at k = 12 cm^{-1} have shown measurable turbulence whose onset is correlated with the start of the electron heating, which suggests high k turbulence may be affecting electron transport. Detailed stability calculations, however, have not yet identified an associated unstable mode [21].

A connection between confinement improvement and observed and calculated turbulence reduction has also been established in discharges run with neon or argon injection to reproduce the TEXTOR RI-mode plasmas [23,24]. As is shown in Fig. 2.1.1–8, injection of neon results in dramatic reduction in density fluctuations observed by beam emission spectroscopy around ρ = 0.8. As the fluctuations gradually decrease, confinement improves. Furthermore, calculations of gyrokinetic stability similar to those done in [18,25] demonstrate that adding neon to the plasma reduces the linear growth rate at all wavenumbers, consistent with the observed confinement improvement. As indicated in Fig. 2.1.1–8, the turbulence at smaller wavenumbers should already be stabilized by E×B shear effects. Similar effects have been seen with argon injection. In these shots, transport analysis demonstrates an improvement in both electron and ion thermal transport which correlates with the reduction in observed density fluctuations. An important feature of these discharges, relevant to the edge stability issues discussed in the next section, is the reduction in edge pressure gradient and edge bootstrap current in H–mode plasmas with neon or argon impurity

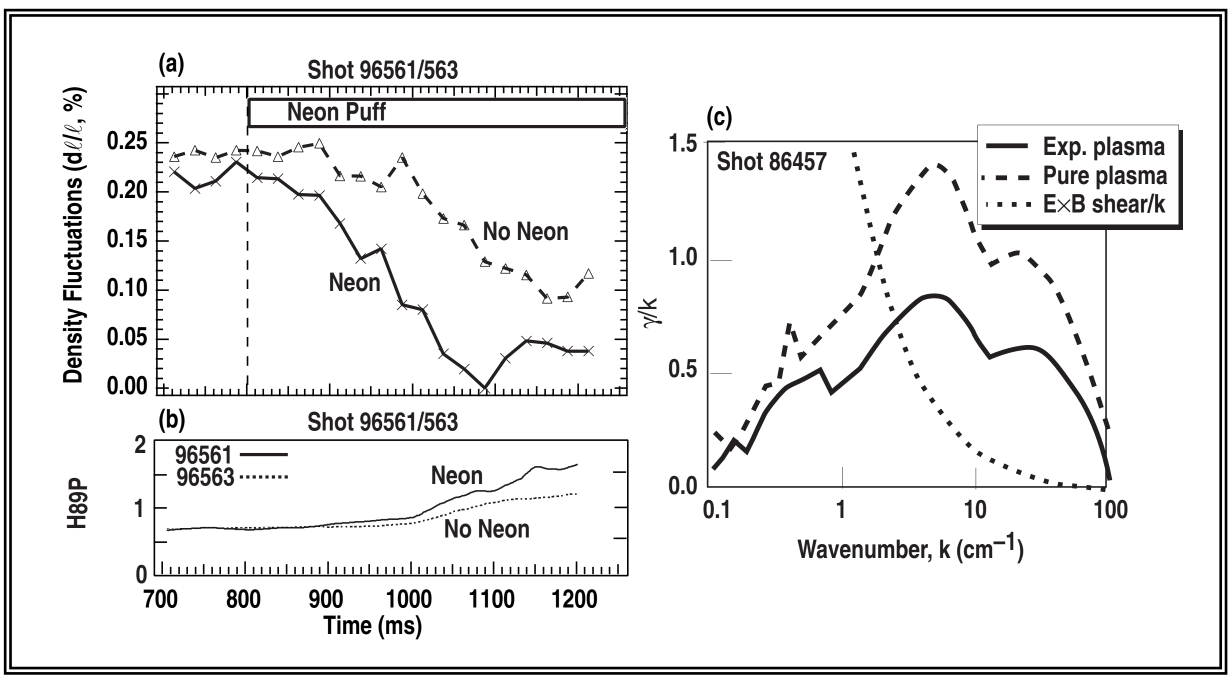

Fig. 2.1.1–8. Reductions in density fluctuations with neon impurity injection,(a), have been observed in L–mode discharges and may contribute to improved confinement,(b). For these L–mode discharges, the confinement enhancement factor is compared to ITER89P L–mode scaling. Gyro-kinetic stability code calculations for a similar discharge show that neon impurities in the mantle region, ρ~0.8, can reduce drift wave growth rates, (c), leading to stabilization of high k ETG modes. E×B shear is also large enough to affect the lower k turbulence associated with ITG modes.

injection. In spite of the edge pressure pedestal reduction, energy confinement remains the same or improves in these plasmas.

Edge Plasma Confinement and Stability. Another factor in obtaining steady-state improved performance discharges is control of edge transport and stability. The confinement physics and MHD stability of the H–mode edge pedestal affect both core plasma and divertor performance. The pedestal height influences overall plasma confinement, as is shown in Fig. 2.1.1–9. In addition, edge stability affects ELM frequency and amplitude, which have a major impact both on core transport barriers and on the divertor. Furthermore, impurity radiation in this edge region is a possible cause of the density limit. Finally, the interaction of the core plasma with the wall can have a major effect on the global MHD beta limits. Accordingly, understanding and controlling the physics of this edge region is a key issue for any future plasma which employs H–mode. In the past two years, DIII–D has demonstrated a strong connection between the pedestal height and core confinement and has demonstrated that the edge pressure gradients are not limited by the ballooning instability. In addition, we have shown that a model based on edge impurity radiation leads to a density limit very similar to the Greenwald prediction in present devices which scales much more favorably with machine size than previously anticipated. Finally, investigation of the physics of resistive wall modes has achieved β values up to 1.4 times the limit with no wall stabilization, has extended the duration of the wall stabilized period by a factor of three, and has produced a successful first attempt at active stabilization of the mode.

Both theoretical expectations [26,27] and the DIII–D results shown in Fig. 2.1.1–9 [28] indicate a connection between the edge pressure pedestal height and the overall energy confinement. This connection is much deeper than the trivial one provided by the edge setting the boundary condition for the plasma core, since a boundary condition effect with no other influence would simply produce a linear relationship between the pedestal pressure and the total stored energy which is not seen experimentally. In the absence of any other constraint, one would naturally want to optimize plasma performance by pushing the edge pedestal pressure to its maximum possible value to improve the energy confinement. Unfortunately, the pedestal pressure is limited by the onset of ELMs. In addition, optimizing plasma shape for the highest possible pressure pedestal usually results in large energy loss per ELM. As

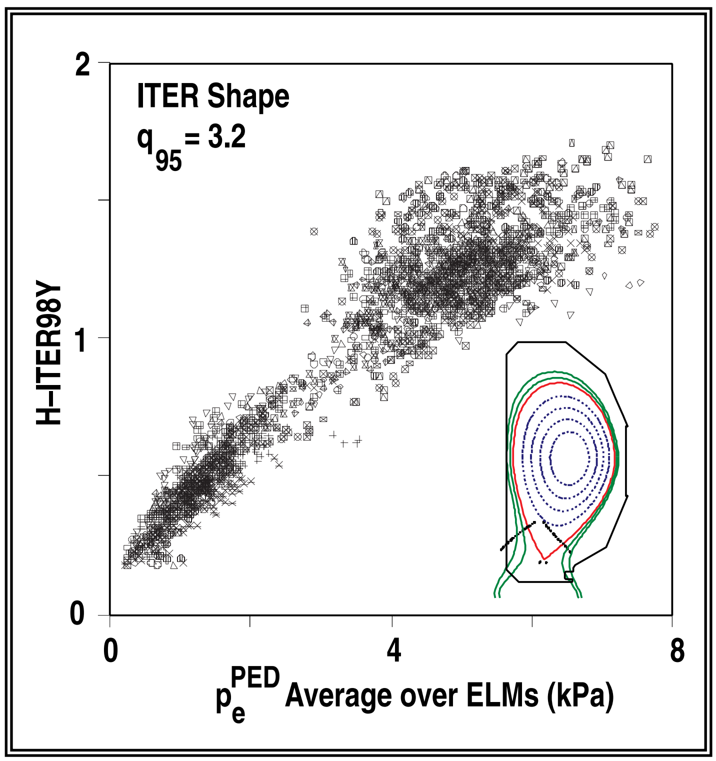

Fig. 2.1.1–9. H–mode energy confinement enhancement factor relative to ITER98 ELMy H–mode scaling increases with increasing H–mode pedestal pressure averaged over ELMs. The increase is approximately $P_{PED}^{2/3}$. This plot includes both Ohmic, L–mode and H–mode phases of a series of shots run in the ITER shape, which has a vertical elongation of 1.75, average triangularity of 0.24 and inverse aspect ratio of 0.34.

has been discussed in the ITER context [29], such large energy loss would be difficult to design for in a large device. Accordingly, control of the edge is needed to obtain the best possible core performance while not adversely affecting the divertor.

Because the ELM physics influences both core confinement and divertor performance, we have undertaken a systematic study of edge plasma stability. Although there has been considerable speculation that the edge pressure gradient just before an ELM is limited by high-n ballooning, detailed measurements on DIII–D have shown that the pressure gradient exceeds this limit by at least a factor of two [28]. As is shown in Fig. 2.1.1–10, we have determined that including the self-consistent edge bootstrap current in the ballooning stability calculation makes a major difference in the stability conclusions [30]. The bootstrap current, driven by the large edge pressure gradient, opens up a ballooning second stable region at the plasma edge. Accordingly, the edge pressure is not limited by high-n ballooning but rather by other, lower n MHD modes which are probably driven unstable by the large pressure and current gradients that ballooning stability allows [31].

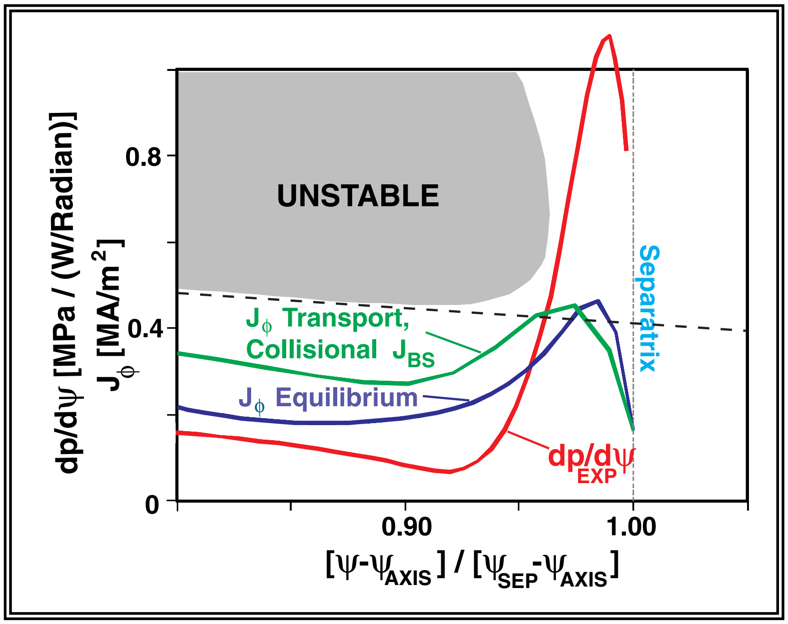

Fig. 2.1.1–10. Plot of the measured pressure gradient and inferred edge current density for a DIII–D discharge similar to those in Fig. 2.1.1–9: plasma current 1.5 MA, toroidal field 1.9 T, q_{95} = 3.4, inverse aspect ratio 0.36, vertical elongation 1.76, and average triangularity 0.28. The edge current density has been inferred from equilibrium reconstruction including MSE measurements and is compared with a transport calculation including a collisional bootstrap current model. The pressure gradients that would be unstable to high-n ballooning mode are shown. Note that the edge pressure gradient is not limited by ballooning modes and the experimental value greatly exceeds the ballooning limit that would be calculated by ignoring the bootstrap current, which is shown by the dashed line.

The highest performance DIII–D VH–mode and negative central shear H–mode discharges are limited by MHD stability at the edge of the plasma; the peak performance is usually terminated in these discharges by low to medium n ideal instabilities at the edge having the characteristics of a large ELM but which normally result in a loss of the transport barrier [32,33]. Recent analysis has demonstrated that the interaction of low n ideal kink and high n ballooning stability plays a crucial role in the attainment and sustainment of high performance. High n ideal ballooning second stability access permits the buildup of the edge pressure gradient. This allows high peak performance but ultimately results in destabilization of the more dangerous low n global edge instabilities which are manifested as the large ELMs that terminate the high performance. Conversely, closing the second stable access at the edge generally limits the pressure gradient and bootstrap current to values well below the low and intermediate n kink limits. This results in lower peak performance with smaller amplitude ELMs which also allow longer discharge duration.

A clear route to long-pulse high-performance operation is, therefore, to control the edge conditions to eliminate second stable access, and to raise the first regime ballooning limit just below the low and intermediate n kink limits. One method for achieving this is through the cross section shape, which can be systematically varied using the DIII–D control system. Calculations have shown that the equilibrium squareness is a useful tool for controlling the edge ballooning stability through its effect on the field line connection length [31,34]. (As its name suggests, high positive squareness discharges have almost square shapes.) Large positive squareness, or low, or negative squareness, can restrict second stability access. This has been exploited in recent experiments in DIII–D [35] in which the ELM frequency is increased and the amplitude reduced at large squareness [31]. Motivated by these results, recent calculations show that higher order local perturbations of the outboard shape, which greatly increase the field line connection length there, can also eliminate second stability access near the plasma edge, with little effect on the favorable low n kink stability properties of D-shaped plasmas [31]. This will be pursued in future experiments.

A second avenue for achieving control of the edge ballooning stability is to increase the edge collisionality to reduce the edge bootstrap current; lower edge current density hinders second stability access. Higher edge collisionality is achieved in DIII–D experiments by increasing the edge radiation by puffing deuterium and argon. In these experiments, the ELM frequency is typically reduced by roughly half and often the ELM magnitude is reduced as well. Figure 2.1.1–11 shows the time history of discharge 95011, in which argon was injected at 2 seconds. In this case, the ELM frequency was reduced by a factor greater than 2 with a small reduction in the ELM amplitude. The confinement is slightly improved by the change in ELM behavior. Figure 2.1.1–11(c) shows the calculated bootstrap current before and after the gas puff. The edge bootstrap current has been reduced and the peak is moved inward. This is reflected in the calculated ballooning stability in Fig. 2.1.1–11(b); the reduced edge bootstrap current has closed off the second stability access in this discharge.

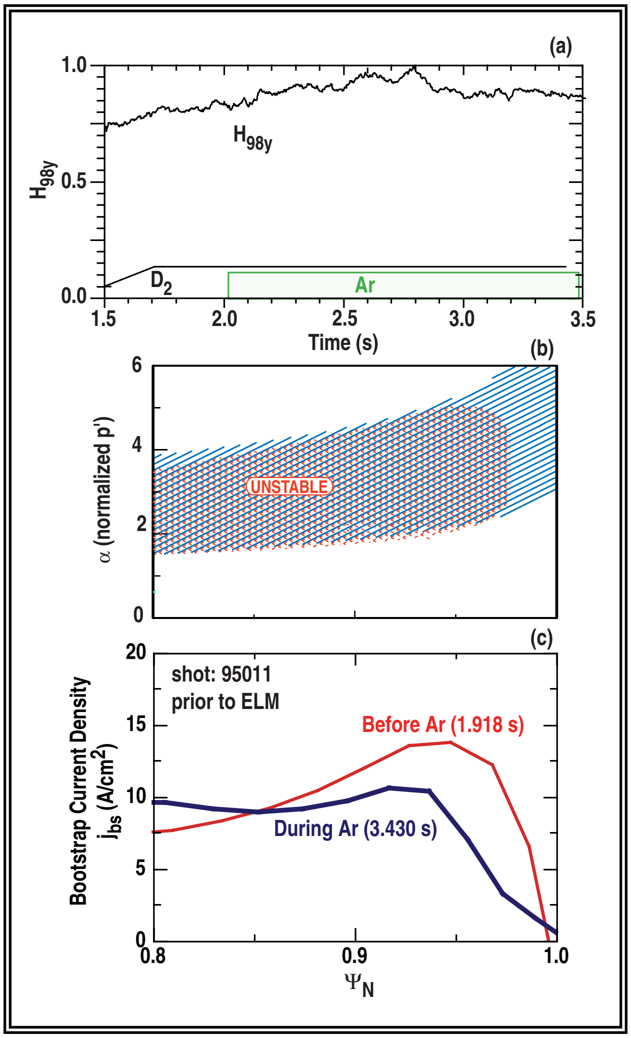

Fig. 2.1.1–11. Effect of Ar gas puffing on edge second stability access in discharge 95011. (a) Time history showing confinement enhancement H_{98Y} and D_2 and Ar injection (b) ballooning stability before and after injection of Ar and (c) computed bootstrap current density over the outer 20% of the plasma normalized flux.

A tokamak density limit scaling of the form $n_e \propto I_p/a^2$ has been reported by several authors [36,37] where I_p is the plasma current and a is the minor radius. However, extrapolation of this scaling to reactors can be misleading because the underlying physical processes have not been determined. We have conducted a series of experiments on DIII–D to determine the density-limiting processes in tokamaks [38,39]. Using the understanding gained through these experiments, we have succeeded in obtaining high confinement plasmas at densities well beyond the limit of the Hugill-Greenwald scaling [39,40]. A key result of these studies is that the n=0, m=1 MARFE condensation instability criterion [41] is in quantitative agreement with high resolution edge measurements on DIII–D [42]. Additionally, we have shown that the MARFE instability condition combined with ITER89P confinement scaling yields an edge density limit scaling of the form:

$$n_e^{crit} \propto \frac{I_p^{0.96}}{a^{1.9}} \xi^{-0.11} P_{heat}^{0.43} R^{0.17} B_T^{0.04} \left[\kappa^2 (1+\kappa^2)\right]^{-0.22} \quad ,$$

where ξ_i is the impurity concentration and κ is the plasma elongation. Except for a moderate power dependence this scaling is remarkably similar to the Hugill-Greenwald scaling. The insensitivity to all plasma parameters except I_p and minor radius a derives from the fact that the MARFE density threshold for low Z impurities (e.g. oxygen or carbon) for an electron temperature range of 10–100 eV increases with the fourth power of T_e. Accordingly, a MARFE nearly always occurs at the same boundary temperature (~20 eV). Therefore, the trade off between density and temperature in the stored energy determines the density scaling. Thus, we conclude that future devices with high edge temperatures can access densities well above the nominal Hugill-Greenwald limit.

Turning now to the physics of wall stabilization, we have developed a double current ramp technique to reliably and reproducibly make plasmas where the β_N values achieved indicate that wall stabilization of MHD modes is important [43]. In addition, improved diagnostics have allowed us to make a direct identification of the resistive wall mode (RWM) mode structure in the plasma interior using ECE spectroscopy. Using these shots, we have achieved a new physics understanding of wall stabilization. We have produced rotating, wall stabilized discharges with the ratio of β_N to the no wall β_N limit E_w up to E_w = 1.4±0.05. For example, in shot 92544, E_w exceeds unity for 200 ms,which is >30 τ_W. The time constant τ_W is the n =1 time constant of the wall (about 5.8 ms in this shot) and is a measure of the penetration time of the potentially unstable mode through the resistive vessel wall. Similar results with E_w well above unity have been obtained in a number of discharges run under similar conditions.

In all wall stabilized discharges, the plasma toroidal rotation is observed to slow down, which ultimately leads to destabilization of the resistive wall mode (RWM) when the plasma angular rotation speed Ω_{plasma} falls below some critical frequency Ω_c. The critical rotation speed Ω_c is robustly reproducible from shot to shot but is strongly dependent on plasma conditions, notably β_N. Investigation of the reasons for this decrease in Ω_{plasma} have determined a clear correlation between its onset and β_N exceeding $\beta_N^{no\,wall}$. However, there is no correlation of the slowing with fast particle driven MHD modes (TAE modes) or low n MHD activity during the slowing down period [44].

Active means of avoiding the RWM are being pursued by controlling either the plasma rotation or the RWM directly. As is shown in Fig. 2.1.1–12, preliminary results from open loop RWM control experiments have demonstrated that the RWM is suppressed by the application of an appropriate correction field using an external coil set located far outside the plasma. A series of discharges with reproducible RWM onset were run, but one discharge used an n = 1 (C-coil) perturbation which was proactively programmed to turn on at the time of the RWM onset with a phase opposing the mode (Fig. 2.1.1–12). As observed from plasma rotation and T_e profiles near q = 3, the RWM started to grow but was suppressed and the plasma recovered when the opposing field was applied. The n=1 radial field soaking through the vacuum vessel wall was measured by a saddle loop array. As is shown in Fig. 2.1.1–12, this field grows without bound in the reference shot without the external n=1 field but remains at a low level with the external field applied, indicating that control was achieved. New experiments in DIII–D with new active feedback power supplies are planned next year to pursue this further.

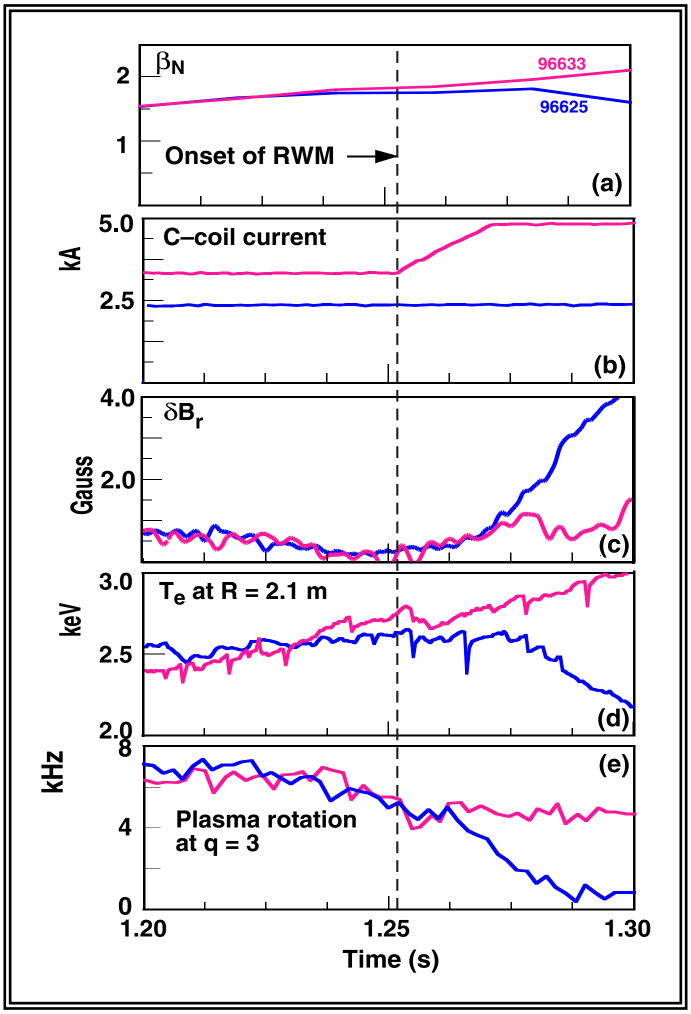

Fig. 2.1.1–12. Time history of discharges with (96633) and without (96625) pro-active control of the RWM (a) β_N (b) current in the C-coil (c) perturbed radial field measured on the saddle loops (d) ECE measurement of the electron temperature at R = 2.1 M. (e) plasma rotation at the q=3 surface. Note that the perturbed radial field grows without bound in the case with constant C-coil current but is stabilized by stepping up the C-coil current with the proper toroidal phase.

Divertor Physics. The key issues in the divertor area are adequate heat removal and simultaneous control of particles and impurities. The major research focus has been on the radiative divertor with additional impurities to enhance the radiation. The challenge here is to maintain sufficient impurity density in the divertor to promote the needed radiation while simultaneously keeping the impurities from overwhelming the core plasma.

Through experiments on DIII–D [45-48] we have demonstrated the efficacy of using induced scrape-off-layer (SOL) flows to preferentially enrich impurities in the divertor plasma. These SOL flows are produced through simultaneous deuterium gas injection at the midplane and divertor exhaust using cryopumping. Using this SOL flow, an improvement in enrichment (defined as the ratio of impurity fraction in the divertor to that in the plasma core) has been observed for all impurities in trace-level experiments (i.e., impurity level is non-perturbative), with the degree of improvement increasing with impurity atomic number. In the case of argon, exhaust gas enrichment using a modest SOL flow is as high as 17. Using this induced SOL flow technique and argon injection, radiative plasmas have been produced that combine high radiation losses ($P_{rad}/P_{input} > 70\%$), low core fuel dilution ($Z_{eff} < 1.9$), and good core confinement ($\tau_E \gtrsim \tau_{E,ITER98Hy}$).

Besides the improvement in impurity enrichment, application of this technique causes several advantageous changes in the plasma [49]. First, at a high flow level, the SOL broadens and its density increases to 1.5×10^{19} m^{-3} while the electron temperature remains approximately 10 eV. Such profiles provide excellent screening of impurities emanating from the vessel wall and an excellent environment for impurity radiation. Second, the ELM amplitude is reduced by approximately a factor of two relative to standard ELMing H-mode conditions. This reduction is accompanied by a proportional increase in the ELM frequency such that the time-integrated energy carried out by the ELMs is approximately the same, but the instantaneous perturbation on the edge and divertor plasma induced by each ELM is much smaller. Modeling has also shown that the ELM dynamics are important in the obtainable impurity enrichment with higher frequency ELMs leading to improved enrichment. These changes are accomplished without significant impact on the core energy confinement.

At the previous IAEA, we reported that parallel thermal conduction based on measured divertor density and temperature profiles in detached plasmas is too small to account for the divertor heat flux and postulated that in the cold divertor zone the dominant transport process is convection along the field lines [38]. A one dimensional interpretive model of the detached divertor plasma [49] has been developed for further understanding of the experimental observations. The model calculates the parallel heat flux in the divertor plasma by integrating plasma radiation, obtained from an inversion of the bolometer data, from the target to a point in the divertor plasma and using the target heat flux, measured by an IR camera, as the boundary condition. The difference between this heat flux and the conduction heat flux, obtained from the measured T_e profile, yields the convective component of the heat flux. It is found that in attached plasmas, as shown in Fig. 2.1.1–13(a), the conduction component accounts for nearly all the heat flux. In contrast, in the detached case, the conduction channel is insignificant compared to the total heat flux [Fig. 2.1.1–13(b)] and convection at approximately the sound speed is required to account for most of the heat flux [Fig. 2.1.1–13(c)]. Furthermore, it is

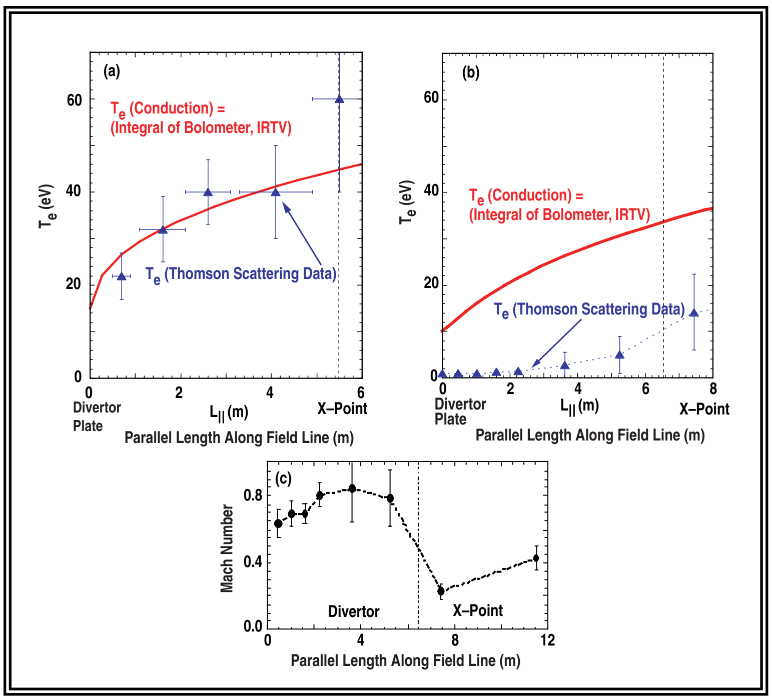

Fig. 2.1.1–13. (a) In attached plasmas, classical parallel conduction accounts for nearly all the divertor heat flux. (b) In the partially detached state the conduction channel is insignificant compared to the total heat flux. (c) Parallel flow Mach number near 1 can account for the balance of the heat flux.

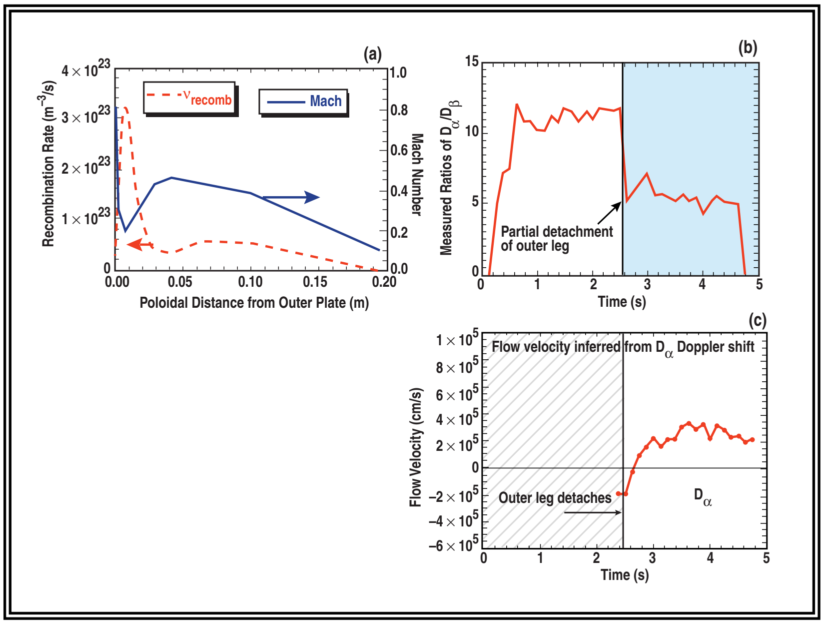

Fig. 2.1.1–14. (a) Modelling of a partially detached plasma shows Mach 0.4 flow in the bulk of the divertor plasma and copious volume recombination near the target plate. (b) D_α/D_β line ratio indicate recombination after detachment. (c) Flow speed of the order of 1/4 sound speed is measured by spectroscopy.

concluded that the observed intense radiation near the target plate must be due to volume recombination since the electron temperature measured by Thomson scattering is too low for excitation radiation.

These experimental results are supported by UEDGE modeling [50] which shows a broad regions of Mach ~ 0.4 and copious volume recombination near the target plate in detached plasmas [Fig. 2.1.1–14(a)]. Recent measurements confirm these experimental interpretations and UEDGE results. Visible and UV line ratio measurements [51,52] show direct evidence of volume recombination [Fig. 2.1.1–14(b)]. Plasma parallel flow speeds at or near the sound speed are also observed by spectroscopy [Fig. 2.1.1–14(c)] [51] as well as a Mach probe [53]. From Langmuir probe potential measurement [53], we also deduce poloidal $E_r \times B_T$ flows. The flow direction depends on the direction of the toroidal field and heat and particle flux associated with it is estimated to contribute significantly to particle exchange between the two divertor strike points and could explain the field-dependent divertor in-out asymmetry.

We have recently installed a divertor baffle and cryopump [54] at the upper divertor whose shape is matched for particle control in high triangularity plasmas ($\delta \sim 0.7$). This installation, combined with the more open pumped lower divertor allows a direct comparison of the effects of geometry on divertor and core plasma performance. A comparison of open/closed divertor operation was carried out with carefully matched plasmas. The cryopumps in each divertor were turned off for this comparison. We observed that the line-average density was very similar in the two cases, but the midplane D_α was reduced in the closed divertor. The density profile was less steep near the separatrix for the closed case, and the temperature responded to keep the electron pressure roughly constant. Transport modeling [54] indicates that the core ionization source was reduced by a factor of about 2.6 in the closed case. No changes in energy confinement during ELMing H–mode operation were observed, but the line average density at which partial detachment occurred was decreased by 20% for the closed case. With the upper cryopump turned on, we achieved active density control with $n_e/n_{Gr} = 0.27$, which is similar to the 0.22 achieved with the lower pump. This establishes an important particle control tool for high triangularity plasma operation in DIII–D. In 1999, we will install a third divertor cryopump for the purpose of pumping the inner strike point in the upper divertor [54] . In addition, a structure in the private flux region which protects the inner pump will serve also as a baffle to reduce the recycling by an additional factor of 2 and isolate the two strike points.

Conclusions. Research on DIII–D over the two years since the last IAEA meeting has made significant progress in the core, edge and divertor areas. We have demonstrated integrated, high performance ELMing H–mode plasmas with β_N H_{98y} ~ 6 for 5 τ_E (~1 s).

In the core physics area, we have

- Shown that core transport barriers can be sustained for the length of the neutral beam pulse (5 s) with no sign of degradation.
- Demonstrated off-axis electron cyclotron current drive with an efficiency well above theoretical expectations.
- Made critical tests of physics-based transport models.
- Produced evidence for passive and active wall stabilization of MHD modes.

In the edge physics area, we have

- Demonstrated the role of edge bootstrap current in edge second stability regime access.
- Developed and tested a physics model of the density limit which agrees with Hugill-Greenwald limit and which scales quite favorably to larger, hotter machines.

In the divertor physics area, we have

- Achieved a new understanding of convection and recombination in radiative divertor plasmas.
- Produced enhanced divertor radiation with scrape off layer plasma flows and impurity enrichment.

This scientific progress sets the stage for future DIII–D research. On a three year time scale, with 6 MW of ECH power, we are aiming at an integrated demonstration of advanced tokamak operation sustained for five seconds. In the nearer term, our experiments will emphasize expanding the spatial extent of internal transport barriers, regulating edge bootstrap currents, stabilizing neoclassical tearing modes, feedback stabilizing high-beta resistive wall modes, and developing the basis for radiative divertors in both single and double null configurations.

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2.1.2. PROGRESS TOWARDS SUSTAINMENT OF ADVANCED TOKAMAK MODES IN DIII–D[2]

Advanced Tokamak (AT) operating modes have been successful in improving the fusion performance of many existing tokamaks, as evidenced by the record D-D fusion reactivity achieved in DIII–D [1], JET [2], and JT-60U [3]. Through optimization of the plasma shape and radial profiles, AT modes lead to improved confinement and β, and higher bootstrap fraction relative to standard ELMy H–mode. Improvements are observed in many different AT regimes such as VH–mode, negative central shear (NCS) with an internal transport barrier (ITB), supershots, high β_p, and high ℓ_i. To date, however, the duration of peak performance in all of these modes is limited to a few energy confinement times (τ_E), generally as a consequence of evolving pressure or current profiles and eventual MHD instability. Before AT modes can be seriously considered as an operating mode for a future fusion reactor, present experiments must sustain AT performance in a controlled manner for longer pulse lengths. In this paper, we review results of recent experiments on DIII–D directed towards this goal.

The primary focus is on improving the performance and pulse length of discharges with an ELMy edge. The ELMy H–mode is inherently steady state, with the edge p′ and impurity concentration regulated by the repetitive ELM events. The ELMy H–mode regime has been studied extensively on most tokamaks and the confinement results have been compiled into a database by the ITER confinement database working group. The most recent scaling for the thermal energy confinement time is given by $\tau_{th}^{98y} = 0.0356 I_p^{0.97}\ R^{1.93}\ (a/R)^{0.23}\ n_{19}^{0.41}\ B^{0.08}\ M^{0.2}\ \kappa^{0.67}\ P^{-0.63}$ [4]. This scaling was generated from mostly sawtoothing ELMy H–modes with monotonic q profiles and q_0~1.

The most serious limitation on β in long-pulse ITER-like discharges appears to be the neoclassical tearing modes (NTM) [5,6]. NTM modes are classically stable tearing modes ($\Delta'<0$) that are driven unstable by a helically perturbed bootstrap current. The NTM mode requires a seed island to exceed a minimum threshold island width; this seed island can be provided by a sawtooth crash or other MHD perturbation. Depending on the density (or collisionality ν^*), NTMs limit β_N to the range β_N~1.7–2.5.

Our goal is to sustain higher β and confinement time relative to the ITER benchmark to achieve a more compact reactor concept with a high bootstrap fraction. The normalized quantity $\beta_N H_{98y}$ serves as a useful figure-of-merit for performance, where $H_{98y} \equiv \tau_{th}/\tau_{th}^{98y}$. At the 1996 IAEA conference, DIII–D reported on non-sawtoothing discharges (#89756, #89795) with H_{98y}~1.4 (H_{89p}~2.4) and β_N~2.9 sustained for up to 2 s in lower single null with triangularity of δ=0.3 [7]. These discharges utilized 1.2 MW of beam power during the I_p ramp to suppress sawteeth and cryopumping to maintain low density. Fishbone (m/n=1/1) bursts were observed throughout the high performance phase and appear to play a role in regulating the on-axis current to maintain q_0~1 thus avoiding sawteeth. Due to the absence of a sawtooth triggered seed island, the value of β_N achieved was almost a factor of 2 above the predicted NTM limit at the operational density (collisionality) However, fishbones can also provide a seed island for the NTM, limiting both the reproducibility of this regime and attempts to further increase β_N.

Here we present recent results from two campaigns designed to improve these earlier results. First, techniques to produce and sustain q_{min}>1.5 in ELMy H–mode are explored. The motivation for this work is to eliminate fishbones and eliminate the 3/2 surface (and possibly the 2/1 surface) so that the NTM β_N limit can be increased. This work is also motivated by the possibility of obtaining a sustained internal

[2]Rice, B.W., K.H. Burrell, J.R. Ferron, et al., “Progress Towards Sustainment of Advanced Tokamak Modes in DIII–D,” presented at 17th IAEA Fusion Energy Conference, October 19–24, 1998, Yokohama, Japan, to be published in a special issue of Nucl. Fusion; General Atomics Report GA–A22999 (1998).

transport barrier (ITB) to further improve the confinement of ELMy H–modes. Second, we report on discharges obtained with a new shape and startup technique that yielded very high confinement (H_{98y} ~2) and beta (β_N~4) during infrequent ELMs with monotonic q profiles. These discharges reached performance levels comparable to ELM-free modes such as VH–mode and sustained this level up to 1 s.

ELMy Discharges with q_{min}>1.5. Current profiles with negative central shear and q_{min}>1.5 are routinely obtained by heating with P_{NBI}~5 MW during the current ramp with an L–mode edge. However, the development of ITBs in these plasmas results in pressure peaking that is difficult to control and often leads to disruption. Also, because of the cold L–mode edge, the current profile and q_{min} evolve more quickly making sustainment difficult. To avoid these problems, a new startup technique has been developed using H–mode during the current ramp as shown in Fig. 2.1.2–1(a). A brief flat spot in the current ramp at 400 ms, coupled with the biasing of the plasma shape toward lower null point in the ion ∇B drift direction, leads to a reproducible H–mode transition. By controlling the I_p ramp rate and the density, values of ℓ_i as low as 0.5 can be obtained, although for ℓ_i < 0, edge stability problems can lead to locked-modes during the I_p ramp. The best results are obtained with a slower I_p ramp and ℓ_i ~ 0.7 as shown in Fig. 2.1.2–1(a).

The high performance phase extends from 3.4–4 s where $\beta_N H_{98y}$~4.2 is sustained for 0.5 s. The high power phase is relatively short in this discharge due to the long formation phase and the termination of P_{NBI}. Similar discharges sustained somewhat reduced $\beta_N H_{98y}$~3.8 for 1.5 s, but these all suffered from continuous NTM activity. Discharge 93144 in Fig. 2.1.2–1 is free from any significant MHD mode. The

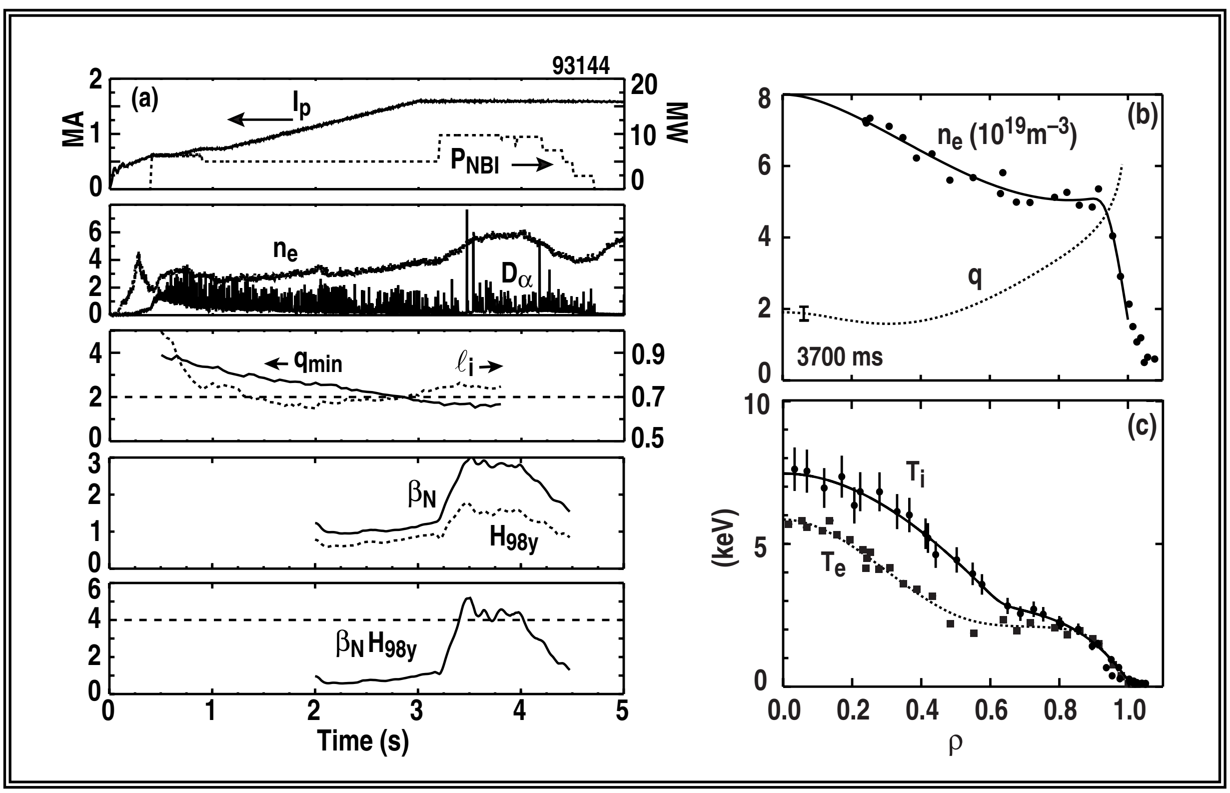

Fig. 2.1.2–1. (a) Time evolution of q_{min}>1.5 discharge (93144) with high performance from 3.5–4 s. (b) Profiles of n_e and q and (c) T_e and T_i at 3.7 s. (B_T=2.1 T, high triangularity shape with δ=0.77, κ=1.85; H_{89p}=2.5 at 3.7 s)

profiles of q, n_e, T_e, and T_i at 3.7 s are shown in Fig. 2.1.2–1(b) and 2.1.2–1(c). The density profile is more peaked than standard ELMy H–mode, but comparable to the other improved performance discharges such as those discussed in the introduction with a monotonic q profile. The T_i and T_e profiles show a somewhat steeper gradient at ρ~0.5 than is observed in monotonic q profile discharges, indicating a weak ITB. However, the ITB is much weaker here than in low density L–mode edge NCS discharges at comparable power. We note that the line average density in H–mode is ~2 times that in L–mode edge NCS discharges, and with this density level we would not expect to observe strong ITBs with an L–mode edge either. The shear reversal [Fig. 2.1.2–1(b)] is weaker in these ELMy H–mode discharges compared with L–mode edge NCS discharges with an ITB. It could be argued that the ITB only forms with strong NCS, but there is evidence on DIII–D that ITBs can form at low density even with monotonic q profiles [8]. Rather, it appears that strong ITBs reinforce the hollow current profile through the off-axis bootstrap current, thus leading to stronger shear reversal.

Although sawteeth and fishbones are not present in the NCS discharges, NTMs continue to be the limiting instability as illustrated by the growth of a resistive m/n=5/2 mode in discharge 93149 shown in Fig. 2.1.2–2. This discharge is similar to 93144 in Fig. 2.1.2–1 except that P_{NBI} is increased to ~ 12 MW. For NTM modes, we expect the mode amplitude to scale with β_N^2, which roughly holds as shown in Fig. 2.1.2–2(b). The classical tearing stability parameter Δ′ was calculated to be negative for this discharge indicating that it should be stable to the classical mode. Although the NTM is not catastrophic in this case, it does result in a saturation of β_N. Despite the increased P_{NBI}, β_N is actually slightly less in 93149 compared with 93144 which has no MHD mode. Note that the mode already exists at a low level in Fig. 2.1.2–2 prior to the step up in P_{NBI} at 2.8 s. For this series of discharges the fast magnetics data on DIII–D was set to cover the high power phase only (t >2.35 s), so it is not clear what initiates the low level mode present at 2.4s. One speculation is that the mode begins as q_{min} passes through q=5/2 at ~ 2 s, then grows later when P_{NBI} is increased.

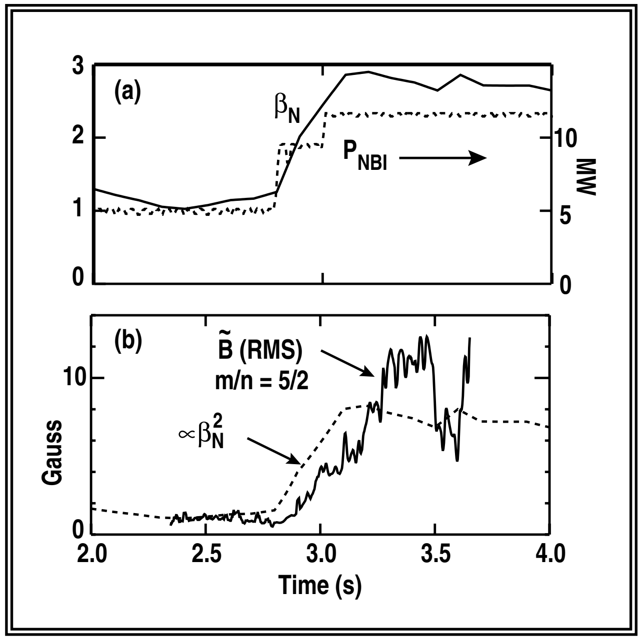

Fig. 2.1.2–2. (a) Time evolution of β_N and P_{NBI} for a q_{min}~2 discharge with NTM activity. (b) Fluctuation amplitude scales with β_N^2 as expected for NTMs.

High-Performance Regime with Infrequent ELMs. A new regime has been developed in DIII–D where performance equivalent to ELM-free modes such as VH–mode is sustained through many low frequency ELMs. As shown in Fig. 2.1.2–3, β_N~3.8, H_{98y}~2, and $\beta_N H_{98y}$>6 are sustained for 1 s. Lines indicating the β_N and H_{98y} values required for ITER-EDA and the ARIES-RS reactor study are also shown, indicating that this discharge exceeds the ARIES-RS requirements for the $\beta_N H_{98y}$ product. The q profile is monotonic with q_0~1 and 1/1 fishbones (but no sawteeth) are present throughout the high performance phase. Some parameters of interest during the high performance phase are: β_t~4.5%, n_e/n_{Gr}~0.5, q_{95}=4.4,

τ_{th}~0.21 s and f_{bs}~50%, where n_{Gr} is the Greenwald density and f_{bs} is the bootstrap fraction. The high performance phase is terminated at ~3 s by the initiation of a m/n=2/1 NTM. Although difficult to judge from Fig. 2.1.2–3, when the NTM begins, the ELM frequency increases to a level more typical of ELMing H–modes at these operational parameters.

Determining the key elements for accessing this regime is still under investigation, but several operational characteristics can be identified. The initiation of a discharge similar to that shown in Fig. 2.1.2–3 is shown in Fig. 2.1.2–4. This discharge reaches slightly higher values of β_N and H_{98y} compared with Fig. 2.1.2–3, but the duration is shorter ~0.7 s. The discharge begins with a fast I_p ramp of ~10 MA/s to 1 MA, followed by a slow ramp rate to the final current of 1.6 MA. Beam power of 5 MW is injected at 0.1 s, leading to an extremely hollow J profile (ℓ_i ~0.3) with ρ_{qmin}~1 at 0.2 s. MHD modes due to the unstable skin current profile cause the current to penetrate rapidly leading to a weak NCS profile at 0.5 s. By the time the beams step up to full power (all co-current beam injection), the q profile has evolved to be monotonic with q_0~1, l_i~1.1, and fishbones are present.

An important feature of these discharges is the plasma shape (see bottom of Fig. 2.1.2–4). A high triangularity (δ=0.77, κ=1.85) single null shape with the X-point at the top of the vessel is utilized. The ion ∇B drift direction is down which raises the H–mode transition power threshold significantly. With this shape it takes 0.5 s at P_{NBI}=9.5 MW before the H–mode transition occurs. This long L–mode phase allows an ITB to form prior to the H–mode transition (H_{98y} ~ 1 in L–mode at 2 s) which enhances the performance after the H–mode transition is made. This technique has been used previously to maximize the fusion power in DIII–D [1], but in those cases the double null shape was symmetrized after the

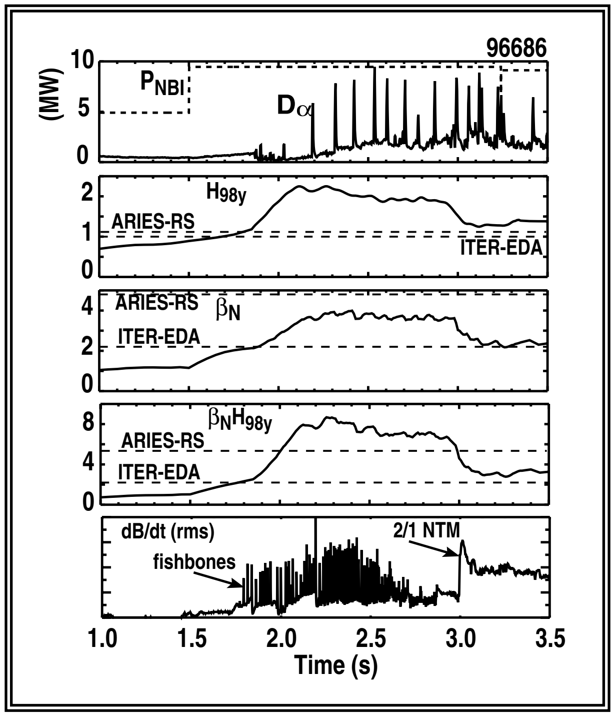

Fig. 2.1.2–3. Time evolution of discharge 96686 with β_N ~ 3.8 and H_{98y}~2 (H_{89p}~3.2) sustained for 1 s during infrequent ELMs. Dashed lines indicate the ITER-EDA design and ARIES-RS reactor study requirements for β_N and H_{98y}.

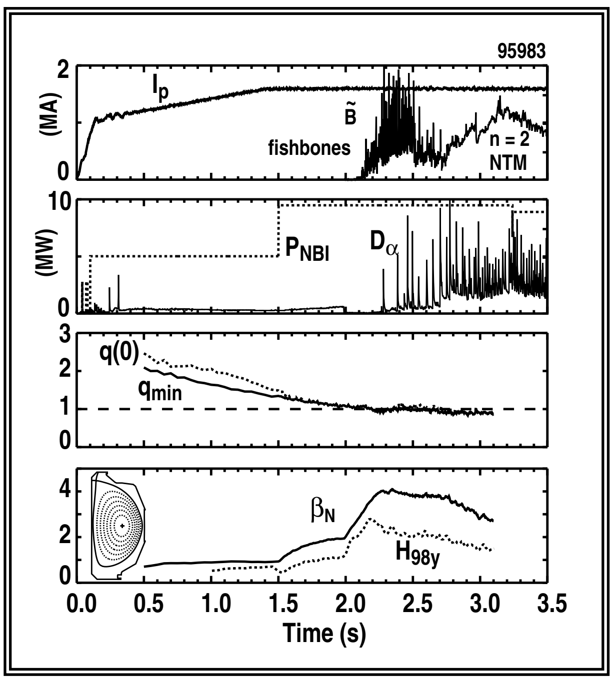

Fig. 2.1.2–4. Time evolution of 95983, illustrating the formation phase and q profile evolution. β_N~3.9 is sustained during infrequent ELMs from 2.2–2.7 s.

H–mode transition and high performance extended through the ELM-free phase only. In these discharges, the shape remains single null with the X–point at the top and close to the wall throughout the discharge.

The exact importance of the startup and plasma shape is not clear. Since the target q profile is simply monotonic with q_0~1, we believe that the early fast I_p ramp and early beam power should not be critical. However, we have been unable to reproduce this regime without a startup similar to that used in Fig. 2.1.2–4. The shape probably plays a more important role, since edge stability is sensitively dependent on edge parameters such as p', J_{95} and collisionality. Also, the close proximity of the X–point to the wall results in higher than usual recycling.

The infrequent ELMs also appear to play a significant role in achieving sustained high confinement. In Fig. 2.1.2–5, the time between ELMs (τ_{ELM}) is plotted versus the H_{98y} at the time of the ELM for several discharges. For τ_{ELM}<20 ms, confinement is close to standard ELMy H–modes with H_{98y}~1 (this represents the later part of these discharges after beta collapses). As τ_{ELM} increases to ~100 ms, confinement similar to VH–mode [9] levels is obtained. The energy loss per ELM can be quite large in this regime, ranging from 2%–5% of the plasma stored energy. With the long period between ELMs, however, the stored energy quickly recovers after the ELM. Infrequent ELMs are necessary but not sufficient to achieve the higher performance. For example, low power ELMy discharges often have infrequent ELMs but the confinement is not significantly improved.

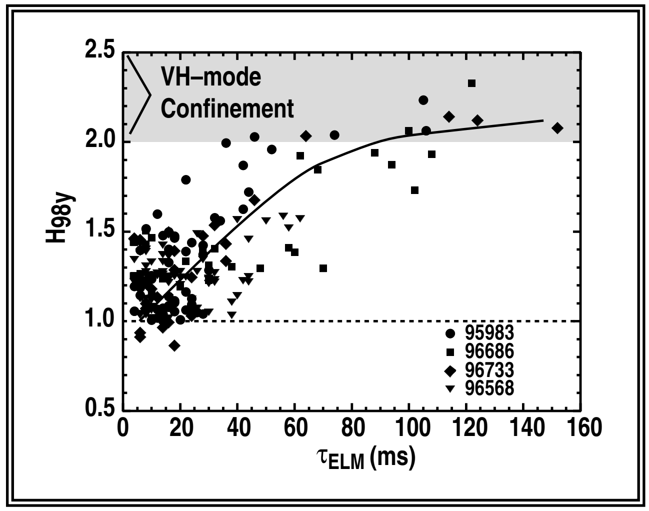

Fig. 2.1.2–5. Confinement enhancement, H_{98y}, improves with period between ELMs. Shaded region indicates typical confinement range for VH–modes on DIII–D.

Another factor in these discharges is that the toroidal rotation and the resulting radial electric field and E×B shear are sustained through the infrequent ELMs. For discharge 95983 in Fig. 2.1.2–4, $v_\phi(0)$~320 km/s and $E_r(\rho$~0.5) ~120 kV/m are sustained from 2.2 to 2.7 s.

Profiles for discharge 95983 (Fig. 2.1.2–4) at 2250 ms, just before the first ELM, are shown in Figs. 2.1.2–6(a–e). For comparison, the profiles of a VH–mode with the same beam power and plasma current are also plotted for a time slice just before the first ELM. In the case of the VH–mode, plasma performance returns to standard ELMing H–mode after the 1st ELM. One clear difference between the two regimes is the peaking of density in 95983 compared with the VH–mode which shows a hollow density profile typical of VH–modes. The high edge density in VH–mode is correlated with the high edge Z_{eff} seen in Fig. 2.1.2–6(b). Compared with the VH–mode, discharge 95983 has somewhat higher Z_{eff} in the core (due to startup conditions) but a lower Z_{eff} at the edge. After the first few ELMs the edge Z_{eff} in 95983 is reduced to < 3.

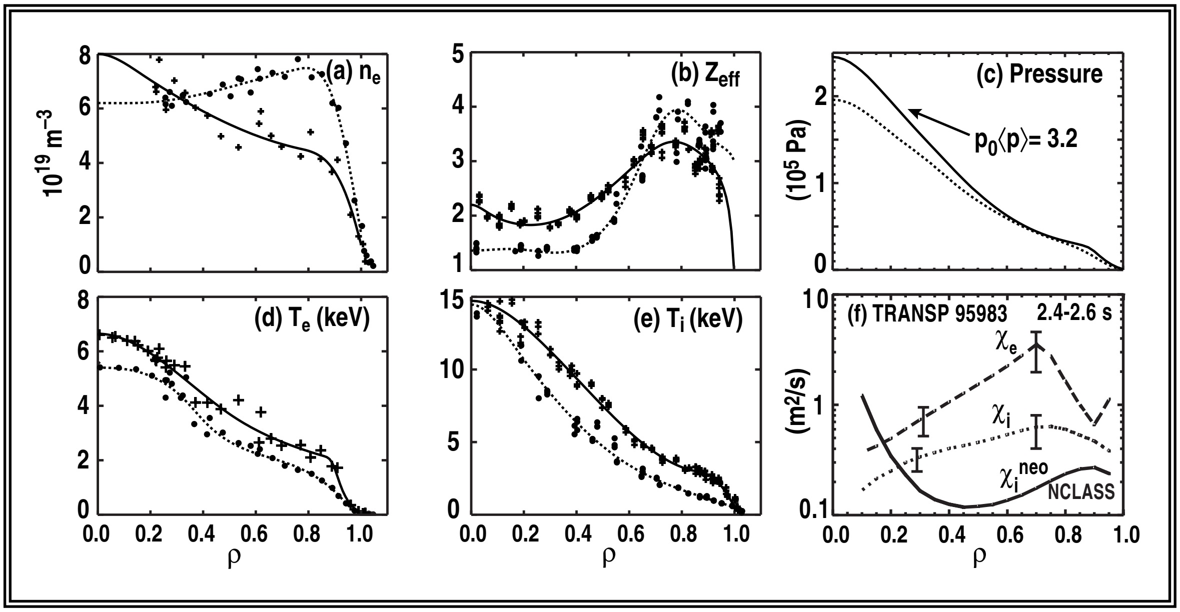

Fig. 2.1.2–6. (a–e) Profiles for 95983 at 2.25 s, just before the first ELM (solid curves and plus symbols) and VH–mode profiles (83710 at 2.75 s) just before the first ELM (dashed curves and sold circles). (f) χ profiles for 95983 during ELMs (2.4–2.6 s) calculated from TRANSP.

Transport analysis has been performed using the TRANSP code for discharge 95983. In Fig. 2.1.2–6(f), the experimental diffusivities χ_i and χ_e are plotted along with the neoclassical calculation for χ_i during the ELMing phase. For this figure, the χ's are averaged over 200 ms during the ELMing phase to smooth over the ELM fluctuations. The reduction in χ_i relative to a standard ELMing H–mode is about 2–3, while χ_e is reduced a more modest amount. Note that there is no indication of an abrupt internal transport barrier, χ_i is simply reduced uniformly across the plasma. Although not plotted here, during the ELM-free phase of 95983, χ_i is equal to the neoclassical χ_i (within error bars) over most of the plasma radius as seen in other high performance discharges [1].

Edge Stability Analysis. The discharges shown in figs. 3 and 4 do not collapse at the first elm as do vh–modes [10], for example. Early speculation on why these elms are more benign focused on the possibility that the edge p′ was reduced. However, as seen in Fig. 2.1.2–6, although the density gradient is substantially reduced compared with VH–mode, the temperatures are higher, so that the final edge pressure gradient and pedestal height are actually higher in 95983 than the VH–mode. Profiles of q, p, and $\langle J_{||}\rangle$ from a kinetic efit prior to the first ELM at 2.25 s are shown in Fig. 2.1.2–7. The EFIT equilibrium reconstruction utilizes external magnetic measurements, MSE data–including E_r corrections–and pressure profiles including the calculated fast ion contribution. The q profile shows a large flat region with q~1 while the current density shows a large edge current peak due to the edge p′ and the resulting bootstrap current. Calculations of the bootstrap current density is shown in Fig. 2.1.2–7(c). The location and amplitude of the measured bootstrap peak is in good agreement with this calculation. Note that the overall current at the edge is somewhat reduced because the edge surface voltage is actually negative during the ELM-free phase leading up to this time. Ballooning and Mercier stability have been calculated with the results plotted in Fig. 2.1.2–7(d). The reduction in edge shear due to the edge bootstrap current opens up 2nd stability access at ρ~0.95 which allows the high edge pressure gradient. This result is typical of most ELM-free,

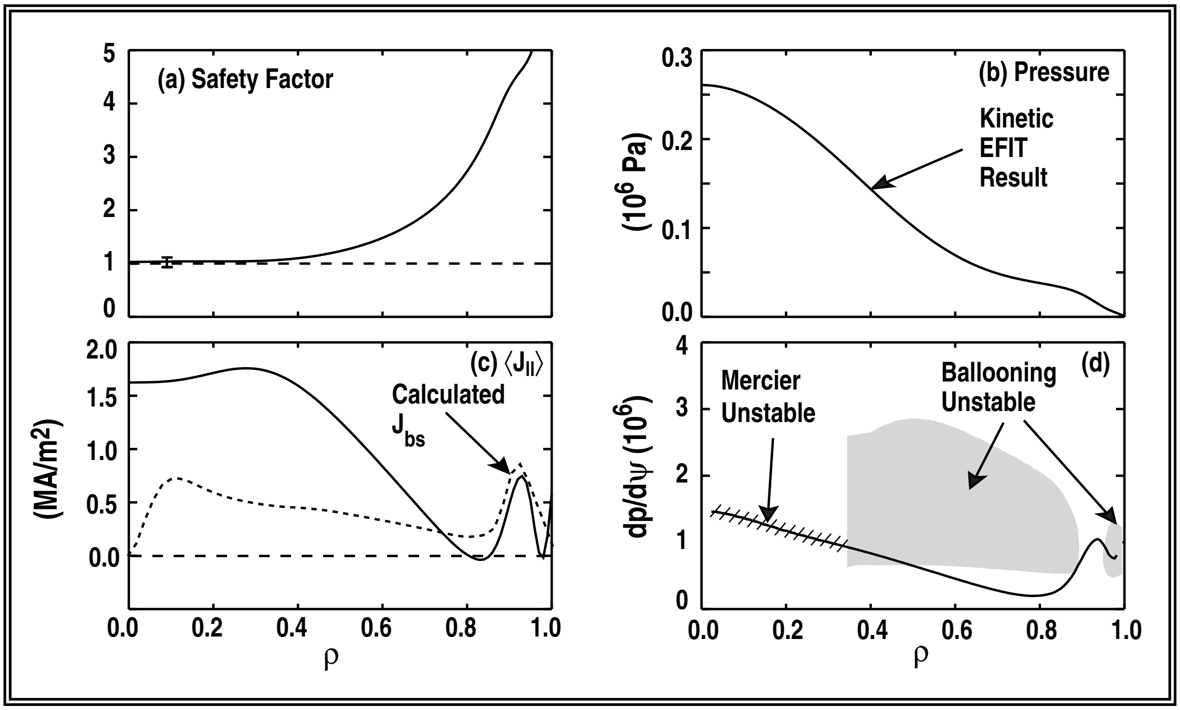

Fig. 2.1.2–7. (a–-c) Kinetic EFIT equilibrium profiles for 95983 at 2.25 s. The dashed line in (c) represents the calculated bootstrap current density for the profiles shown in Fig. 6. (d) Ballooning/Mercier stability diagram at 2.25 s. The solid curve is the experimental value of dp/dψ.

high triangularity plasmas on DIII–D. The core is Mercier unstable due to low value of q inside ρ~0.35 and the large pressure gradient in this region. We note that recent theoretical work on the stabilizing effects of fast ions may modify the Mercier criterion in this region [11].

GATO calculations of ideal n=1–4 stability has also been performed for this timeslice. In general, GATO predicts that this regime is marginally stable to n≥1 edge "peeling" modes, although depending on the exact details of how the input equilibria is prepared, GATO can also find these modes to be unstable. The important point, however, is that experimentally there seems to be no coupling between the edge mode and a more destructive global mode. The reasons for this are not known, but we suspect that unique features of the shape and edge current profile may be playing a role. Despite the large current spike observed at ρ~0.95 in Fig. 2.1.2–7(c), the average current from 0.75 < ρ <1 is not so high at this time because the surface loop voltage is slightly negative.

Neoclassical Tearing Modes. All of the discharges produced to date that achieve high performance (β_N>3.5) during ELMs revert to a standard H–mode with a soft beta collapse accompanied by MHD activity that has the characteristic signature of the NTM. These characteristics are that (1) $\Delta' < 0$ indicating modes are classically stable; (2) a seed island threshold width must be exceeded before the mode can grow; (3) the mode amplitude saturates at a level that can be predicted by NTM theory and is $\propto \beta_N^2$. Since none of these discharges have sawteeth, the seed island must be generated by another mechanism. In many cases fishbone bursts provide the seed island, although there are also cases where the NTM trigger is not so clear. For discharge 95983 in Fig. 2.1.2–4, the NTM clearly starts to grow after a fishbone burst as shown in Fig. 2.1.2–8. Here contours of constant mode amplitude are plotted for n=1 (dark

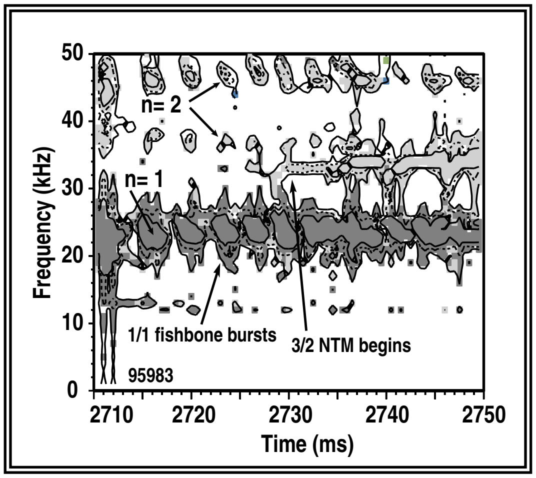

Fig. 2.1.2–8. Contours of constant B-dot mode amplitude for n=1 (dark shade) and n=2 (light shade) modes. A continuous 3/2 mode is triggered by a fishbone burst at 2730 ms.

shade) and n=2 (light shade) modes. In addition to the 1/1 fishbone bursts and the associated second harmonic, one can observe the coupled 3/2 mode at ~38 kHz. The first few 3/2 bursts die away, but at 2730 ms the 3/2 mode becomes continuous and begins to grow. The 3/2 mode number has been confirmed by analyzing the poloidal magnetic probe array and by comparing the 3/2 mode frequency with the fluid rotation velocity at the q=3/2 surface. Both the seed island width and the saturation amplitude have been estimated from Mirnov signals for this discharge and agree with predictions from NTM theory.

While NTM modes appear to limit β_N in many of the high performance ELMy H–mode discharges discussed in this paper, the β_N limit is significantly higher than the limit in sawtoothing ITER-like discharges. In Fig. 2.1.2–9, the critical β_N for onset of NTMs is plotted versus a function of density (or collisionality ν*) as determined by La Haye, *et al.* for sawtoothing ITER-like discharges [12]. The β_N limit for the high performance ELMing H–mode discharges discussed in this paper are also plotted and range from 50% to 100% higher than the sawtoothing limit. Despite this success in raising the NTM β limit, the NTM modes limit our ability to further increase β and, since the time at which the seed island triggers the mode varies considerably from shot to shot, the reproducibility of these discharges is poor. Techniques to calculate and produce profiles that are more robustly stable to these modes is an important next step.

Discussion. Over the past two years, improved performance has been achieved in a variety of ELMy H–mode discharges in DIII–D. With normal-frequency ELMs (50–100 Hz), H_{98y}~1.4 (H_{89p}~2.5) and β_N~2.9 has been sustained for ~ 1.5–2 s (10–15 τ_E) in discharges with both monotonic (q_0~1) and NCS q profiles. The absence of sawteeth in these discharges plays an important role in accessing higher confinement and β relative to the ITER benchmark. In the NCS discharges, sawteeth are simply eliminated by raising q_{min} well above unity. In the monotonic q profile discharges, fishbones appear to play a role in maintaining q_0 at, or slightly above, unity, thus preventing sawteeth. Since there is no obvious reconnection of flux during fishbones, the mechanism for sustaining q_0~1 is not clear.

By removing the core sawtooth instability, more peaked density and temperature profiles and larger core

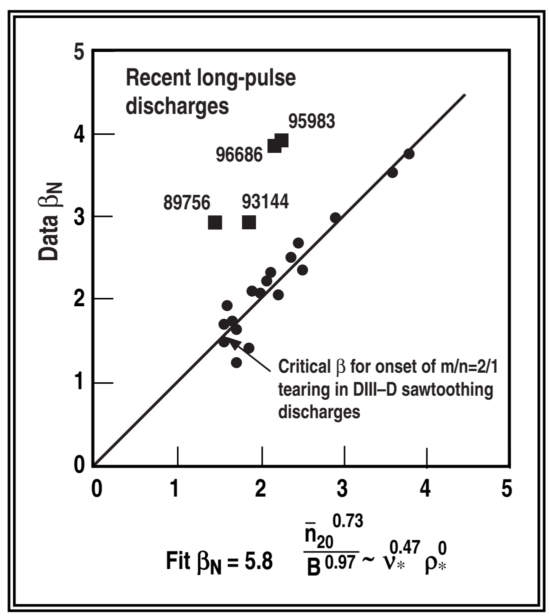

Fig. 2.1.2–9. The beta limit for discharges discussed in Section 2.1.2 (squares) exceed the NTM limit established for sawtoothing discharges (circles and line) by a factor of ~2.

rotation (E×B shear) are observed, enabling a ~40% increase in global confinement compared with the ITER98y scaling. Although core confinement is improved in these discharges, the sharp localized ITBs seen in L–mode edge NCS discharges are not observed here.

Regarding stability, the absence of sawtooth induced seed islands allows β_N values up to 2 times the previously established neoclassical tearing mode limit. In spite of this, NTMs triggered by other MHD events (fishbones, ELMs) remain a limitation to both the reproducibility of long-pulse discharges and further attempts to increase β_N.

A new regime characterized by infrequent ELMs (≤ 10 Hz) has produced performance comparable to VH–mode, but sustained for longer pulse lengths. The best of these discharges produced H_{98y}~2 (H_{89p}~3.2) and β_N~3.8 for 1 s (~5τ_E). Again, here the q profile is monotonic with q_0~1 and there are fishbones but no sawteeth. Ion thermal diffusivity is reduced over most of the discharge to ~2–3 times neoclassical. Although the ELMs are not small in this regime (2%–5% energy loss per ELM), they are benign in the sense that no global MHD mode that could cause a core β collapse is triggered. The long period between ELMs allows the stored energy and toroidal rotation time to recover, giving confinement properties that are closer to ELM-free regimes than ELMy regimes. The pulse length of most of these discharges is ultimately limited by the triggering of resistive NTMs rather than ideal MHD modes that terminate typical VH–modes.

The reasons for the improved stability are still being investigated, but we can identify several features that may be important. First, the upper single null high-triangularity shape has a high H–mode power threshold that results in a long enhanced L–mode phase. Upon the H–mode transition, high performance is obtained rapidly, before the current density in the edge region can fully develop. Although a large bootstrap peak at ρ~0.95 is observed, the average current density in the region 0.8 < ρ < 1.0 is not so large and this is beneficial for stability to edge peeling modes. Also, the X–point is located close to the wall in these discharges which results in larger recycling near the X–point. This may have a subtle effect on details of the edge profiles and gradients.

Future plans include making use of the upcoming high-power electron cyclotron current drive (ECCD) system on DIII–D. By driving current off-axis, we will be able to achieve improved control over the q profile and can sustain the q profile for longer pulse lengths. The ECCD system will also be used for experiments on stabilizing NTMs, which are a significant limitation in long-pulse discharges.

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2.2. CONFINEMENT AND TRANSPORT

2.2.1. BEHAVIOR OF ELECTRON AND ION TRANSPORT IN DISCHARGES WITH AN INTERNAL TRANSPORT BARRIER IN THE DIII–D TOKAMAK[3]

Recent experiments on the DIII–D tokamak have been performed to further elucidate the conditions of and underlying physics behind the formation of internal transport barriers (ITB), or regions of reduced transport. Discharges with ITBs, produced by application of neutral beam preheating to low density discharges during the initial current ramp, were previously employed as the target for the highest fusion performance achieved in DIII–D [1,2]. In these plasmas, increased heating power was applied later in the discharge to expand the ITB and combine it with an H–mode edge to produce a state where neoclassical ion thermal transport was achieved throughout the entire volume.

The ITB often forms in the early phase of neutral beam heated discharges in DIII–D with negative central magnetic shear (NCS), in a region localized near the magnetic axis. The ITB region continues to develop and expand during the low-power "preheating" phase. In this paper, we will discuss the early evolution of this discharge, during the phase where the transport barrier forms and expands. The ITB expansion phase is characterized by ITB growth events superimposed on the steady evolution of the discharge. Although this behavior is consistent with theory, a puzzling feature is that in some discharges, these events correlate with low-order rational values of the safety factor *q*, but in others do not. This implies that there might be two different processes involved; one where the events are triggered by magnetohydrodynamic (MHD) instabilities, and another where the events are related to the behavior of microturbulence.

Another feature of the ITB discharges previously noted is that although ion thermal transport is reduced, in many cases to neoclassical levels, the behavior of the electron thermal transport often remains anomalous. Recent experiments have utilized rf electron heating to probe the underlying physics behind the electron thermal transport. The hypothesis on which these experiments were based is that electron thermal transport may be controlled by electron temperature gradient (ETG) turbulence. Although there are indications that this might be the case, other, not yet fully identified processes appear to be at work as well. Several possibilities are discussed in this paper.

Transport Barrier Formation and Development

Initiation of the Internal Transport Barrier. ITBs are often formed in discharges in DIII–D during the early, "preheating" phase of a discharge (Fig. 2.2.1–1). During this phase, low to moderate (2.5–5 MW) neutral beam power is applied to a plasma with negative central magnetic shear $\hat{s} = r/q\ \partial q/\partial r < 0$. The transport barrier forms in the core of the discharge even with this low level of power, but does not expand outward until and unless the power is increased. The requirement of sufficient power to form the ITB implies the existence of a power threshold. In DIII–D, this threshold is approximately 2.5 MW in full-field discharges ($B_T = 2.1$ T).

[3]Greenfield, C.M., C.L. Rettig, G.M. Staebler, et al., "Behavior of Electron and Ion Transport in Discharges with an Internal Transport Barrier in the DIII–D Tokamak," presented at 17th IAEA Fusion Energy Conference, October 19–24, 1998, Yokohama, Japan, to be published in a special issue of Nucl. Fusion; General Atomics Report GA–A22990 (1998).

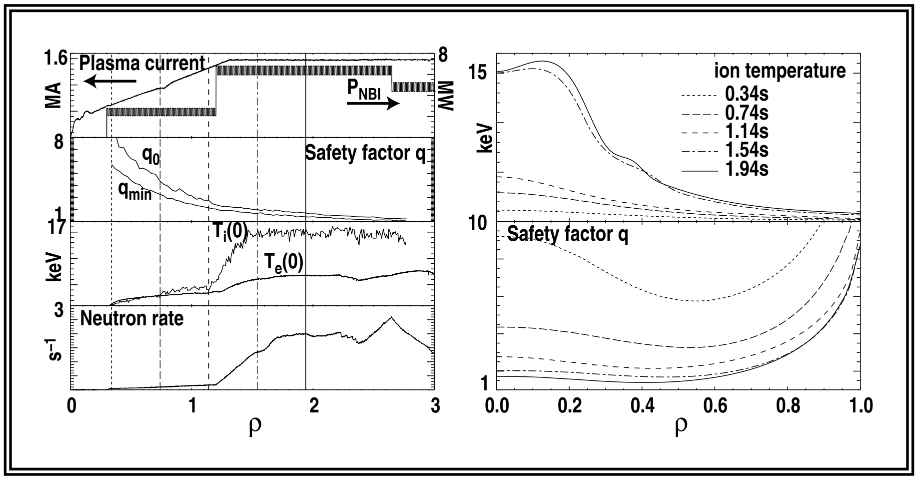

Fig. 2.2.1–1. Waveforms and profiles of a typical discharge with an internal transport barrier. Shot 92389, I_P = 1.5 MA, B_T = –2.1 T, P_{NBI} = 2.5–6.5 MW.

These observations are in agreement with theory [3], which predicts that such transport barriers can be formed when the $\vec{E} \times \vec{B}$ shearing rate exceeds the calculated growth rates of drift wave turbulence [4]. For this stabilization mechanism to be effective, however, the drift wave turbulence must first become the leading driver of transport. Negative central magnetic shear is important here, as it, along with finite Shafranov shift of the magnetic axis, stabilize MHD instabilities which might otherwise dominate transport in this region. Also, the elevated central q values prevent the onset of sawtooth instabilities that would otherwise limit core performance.

The first consequence of the preheating is that it increases the core electron temperature and conductivity, and therefore the current diffusion time. This results in the aforementioned favorable current density profile, which is peaked off-axis. Without the early heating, the current profile would rapidly evolve to become peaked on-axis. The heating does not prevent the current profile from becoming monotonic, rather it only delays this condition. These discharges, therefore, are inherently transient. Future experiments in DIII–D will address this by applying noninductive, local current drive to maintain the current profile, either on-axis in the direction opposed to the plasma current, or off-axis in the parallel direction. This may be done using counter-injected neutral beams, electron cyclotron current drive or fast wave current drive.

The second effect, once the discharge has evolved to a state where MHD is not a leading driver of transport, is to increase the pressure and rotation gradients to generate the large $\vec{E} \times \vec{B}$ shear necessary to suppress microturbulence and therefore locally reduce transport.

Expansion of the Internal Transport Barrier. Once the ITB is formed, the reduced transport allows the pressure and rotation gradients to further increase, thereby generating more $\vec{E} \times \vec{B}$ shear and further reduced transport. If the applied power is at or slightly above the threshold value, the barrier is formed, but remains stationary at $\rho \approx 0.3$. At higher power levels, typically 5 MW or above, the plasma enters a feedback loop where transport, $\vec{E} \times \vec{B}$ shear and fluctuations evolve toward a state of very low transport. In this condition, the ITB expands outwards to encompass a larger portion of the plasma volume (Fig. 2.2.1–1). The previously reported [1,2] discharges in which the ion thermal transport was reduced to neoclassical levels throughout the entire plasma were the most extreme example of this phenomenon and requires an H–mode edge. In typical discharges in DIII–D with an L–mode edge, however, the transport barrier does not expand past $\rho \approx 0.5$.

The barrier expansion phase of the discharge is characterized by an evolution that is anything but quiescent. During this development, stepwise growth events are observed in the otherwise steadily evolving ion temperature profile, as well as transient local decreases (Fig. 2.2.1–2). The steps often correlate with similar events observed in the plasma rotation and electron temperature. Also, consistent with predictions of numerical modeling of the discharge dynamics [5], transient reductions in fluctuation amplitudes are observed at the same time as the steps (Fig. 2.2.1–2). Measurements of the local change in electron temperature across one such event (Fig. 2.2.1–3) reveal the existence of a strong transport barrier, in this case at $\rho \approx 0.4$. The temperature increase is confined within the transport barrier, while the profiles outside this region are relatively unaffected.

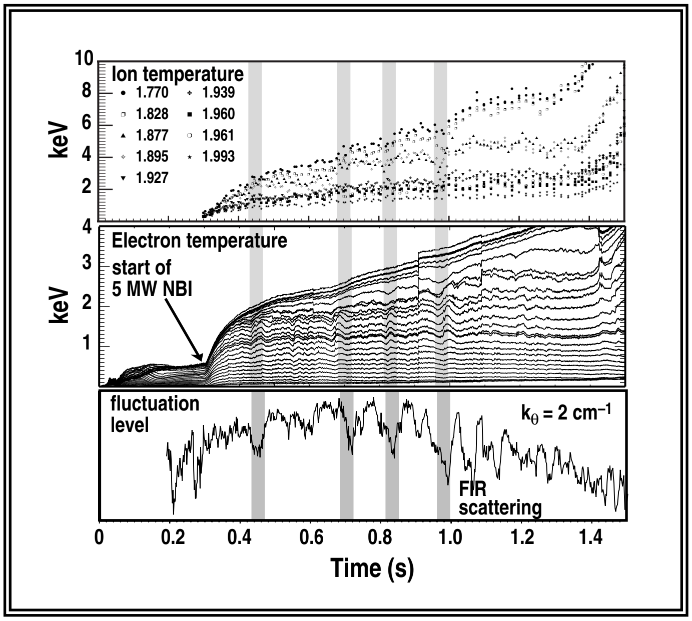

Fig. 2.2.1–2. Evolution of a discharge with a developing ITB during the preheating phase (0.3–1.37s). Shots 89939–89949 (composite), I_P = 1.6 MA, B_T = –2.1 T, P_{NBI} = 5 – 8 MW. The gray bars denote transport barrier growth events.

One consideration in evaluating the data is the possibility that the "bursting" behavior is due to local or global MHD instabilities which are momentarily triggered as the safety factor q passes through low-order rational values. Although it is certainly true that the current profile is evolving during this phase, and that the minimum safety factor q_{min} does periodically pass through integer values, the transport events noted from the kinetic profiles and fluctuation measurements do not usually appear to correlate with integer q values (Fig. 2.2.1–4). We have reasonable confidence in this assertion for these discharges. Even a systematic error in the q profile would not bring the transport events into line with integer q crossings. Also, since the current profile is rather flat in the vicinity of the transport barrier, it is difficult to determine the exact time when integer q

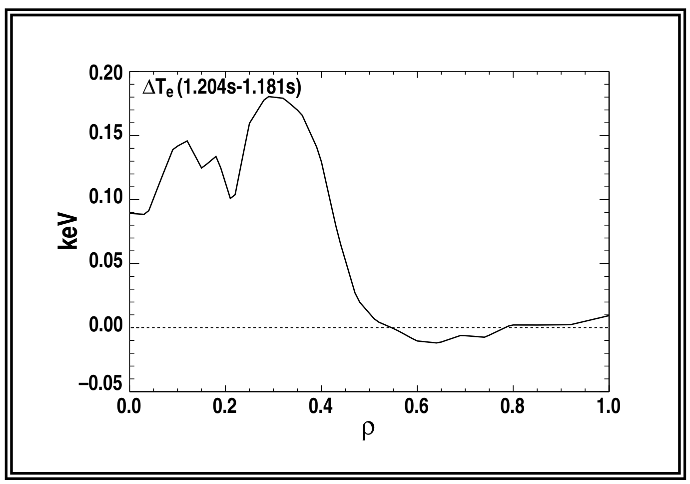

Fig. 2.2.1–3. The electron temperature rise during an ITB growth event is confined to inside the ITB, indicating that the ITB actually behaves as a local barrier. Shot 94100, I_P = 1.0 MA, B_T = –2.0 T, P_{NBI} = 4 MW.

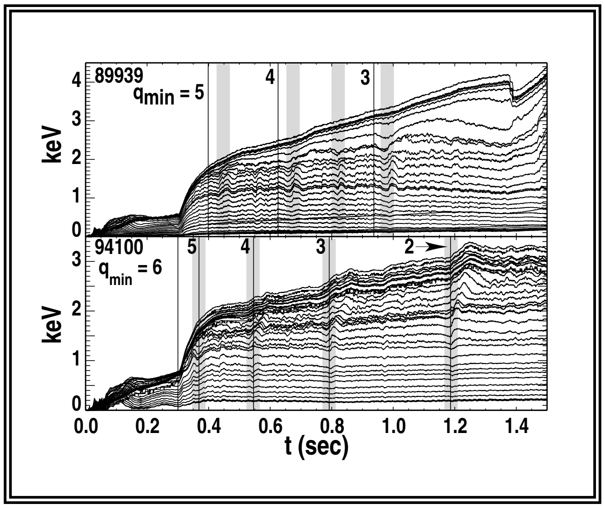

Fig. 2.2.1–4. Electron temperature (ECE) for a discharge which does not exhibit a correlation between ITB growth events and integer q_{min} (89939, I_P = 1.6 MA, B_T = –2.1 T, P_{NBI} = 5 – 8 MW) and one which does (94100, I_P = 1.0 MA, B_T = –2.0 T, P_{NBI} = 4 – 8 MW).

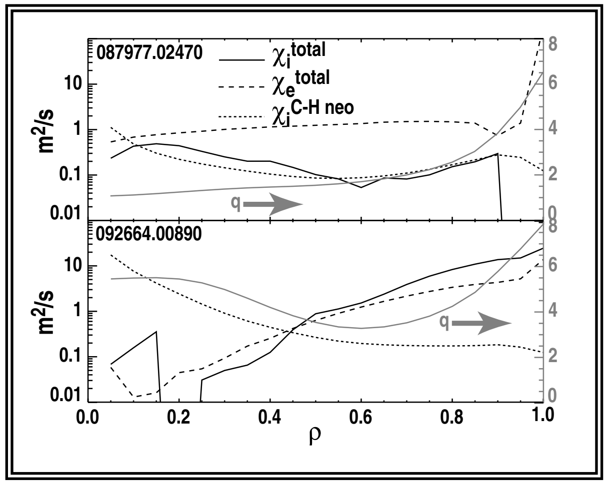

Fig. 2.2.1–5. A low current, low power discharge forms and maintains an ITB until the neutral beams are turned off. The temperature and density profiles are ready after2.5 s, with the q-profile reaching a near-stationary state for the last 1.0 s. Shot 94777, I_P = 0.6 MA, B_T = –1.9 T, P_{NBI} = 4 MW.

crossings occur. However, at least at some of the transport events, the local and minimum q values are both far enough from an integer value to make the MHD arguments implausible for this case.

There are, however, counter-examples exhibiting transport events that appear to contradict the above reasoning. In these discharges, integer q values are well correlated with the ITB growth events (Fig. 2.2.1–4). The reason why we observe such similar behaviors differing in their temporal correlation with integer q values in similar discharges is not well understood, and is currently under investigation. The two sets of discharge evolutions also both lead to the same state, where the ion thermal transport is reduced to neoclassical levels at and inside the ITB.

Anomalous Electron Thermal Transport. In most plasmas, the electron diffusivity χ_e remains anomalously high even when a transport barrier is established for ions. In the discharge shown in Fig. 2.2.1–5, for example, the ion thermal transport has decreased to neoclassical levels throughout the plasma, but transport in the electron channel remains anomalously high. In some discharges with strongly reversed central magnetic shear, the electron diffusivity is reduced as well (Fig. 2.2.1–5). Whether strongly reversed magnetic shear is a necessary and/or sufficient condition for electron ITB formation is not currently known.

We have identified a reproducible tool to *increase* transport in the electron channel. Applying central electron heating to a discharge with an ion transport barrier can have a large deleterious impact on transport in the electron channel. Experiments have been done in DIII–D using both electron cyclotron (ECH) and fast wave (FW) power to heat electrons in target discharges established as detailed in Section 2. The central electron temperature increases upon application of additional electron heating (Fig. 2.2.1–6), but far less than would be expected under conditions of constant electron diffusivities. In fact, transport analysis performed using the TRANSP [7] code indicates that the core electron diffusivity increases by a full order of magnitude on the application of this electron heating. Perhaps equally intriguing is the fact that

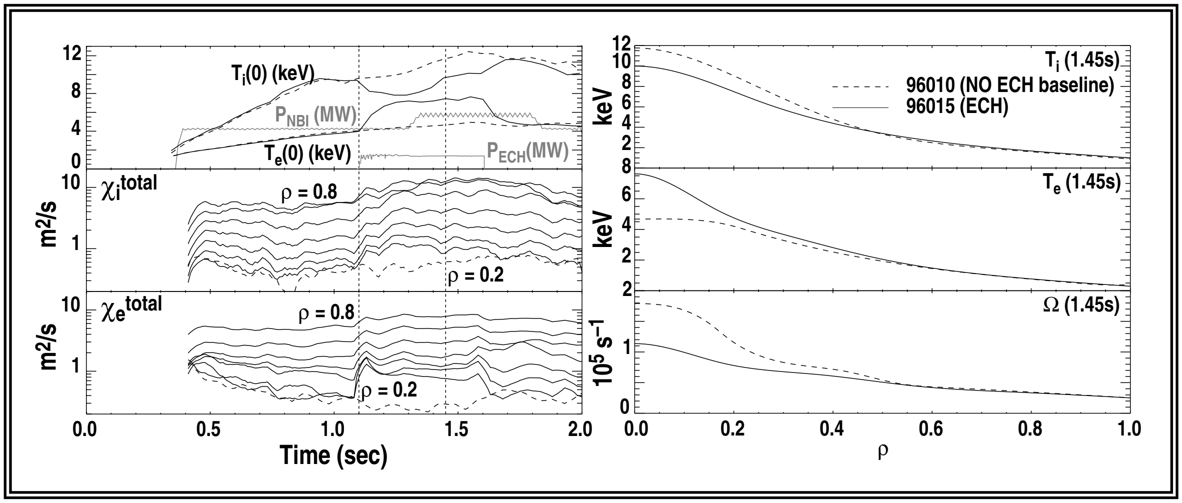

Fig. 2.2.1–6. Application of ECH, heating only electrons near the resonance at ρ = 0, interrupts the formation of an internal transport barrier. Shots 96010 (no EH) and 96015 (1.1 MW ECH 1.1–1.6 s). I_P = 1.6 MA, B_T = –2.0 T, P_{NBI} = 4.3–5.7 MW.

application of 1.1 MW of (ECH) electron heating to the discharge of Fig. 2.2.1–6 actually results in 20%–40% reductions to the central ion temperature and impurity rotation, and associated large increases to the ion thermal and angular momentum diffusivities.

This appears not to be a direct consequence of the heating method. The results are very similar for both the FW and ECH cases, each of which exhibits increased diffusivities in the electron, ion and angular momentum channels (Fig. 2.2.1–7). We concentrate here on the ECH discharge, since we have more confidence in the power deposition calculations used in the TRANSP analysis. The statements made here, however, could just as easily be made with regard to the FW heated discharges [8].

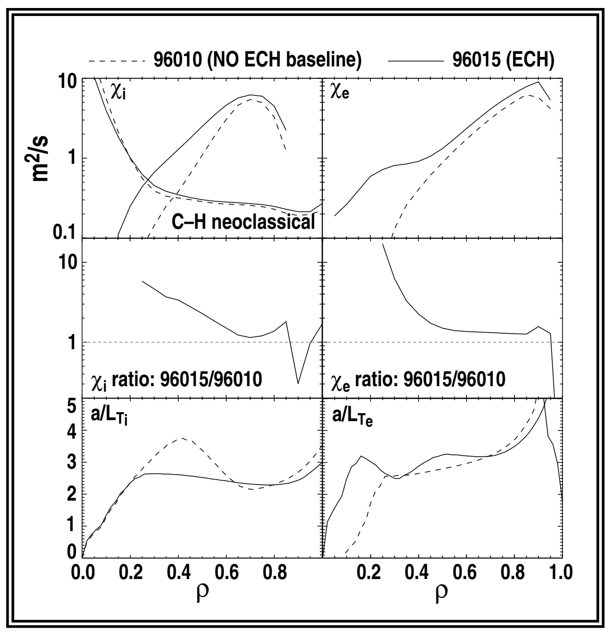

Fig. 2.2.1–7. Comparison of diffusivities with and without EH. The normalized temperature gradients a/L_{Ti} and a/L_{Te} also indicate a reduced ITB with EH. Shots 96010 (no EH) and 96015 (1.1 MW ECH 1.1–1.6 s). I_P = 1.6 MA, B_T = –2.0 T, P_{NBI} = 4.3–5.7 MW.

The transport behavior appears consistent with the hypothesis of $\vec{E} \times \vec{B}$ shear suppression of turbulence leading to ITB formation. The discharge heated only with neutral beams exhibits a ion ITB in this region (Fig. 2.2.1–7). Examination of the $\vec{E} \times \vec{B}$ shearing rate profile in this discharge (Fig. 2.2.1–8) indicates two maxima, with turbulence suppression most likely in the vicinities of $\rho \approx 0.2$ and $\rho \approx 0.6$. In the discharge with additional electron heating (EH), the shearing rate is sharply decreased and has collapsed to a single maxima (Fig. 2.2.1–8). This is reflected in the ion temperature profile as a reduction in the normalized ion temperature gradient a/L_{Ti} at $0.2 \le \rho \le 0.6$ (Fig. 2.2.1–7).

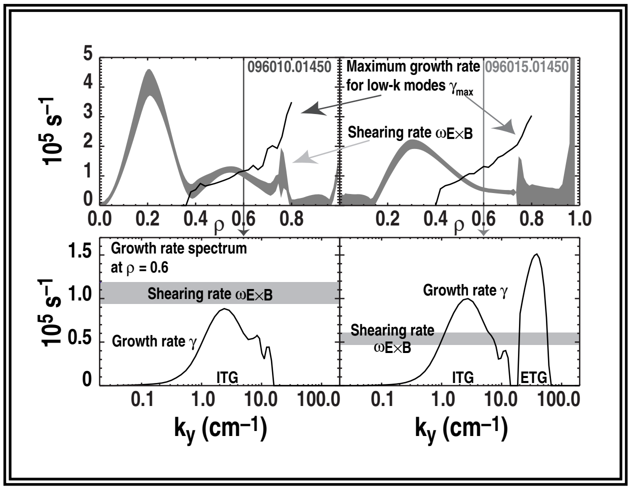

Fig. 2.2.1–8. Microturbulence growth rates from GKS code compared to E×B shearing rates. Shots 96010 (no ECH) and 96015 (1.1 MW ECH 1.1–1.6s). I_P = 1.6 MA, B_T = –2.0 T, P_{NBI} = 4.3–5.7 MW.

Calculations of microturbulence stability have been made using a linear gyrokinetic stability (GKS) code [9] which has been extended to non-circular, finite aspect ratio equilibria [10] with fully electromagnetic dynamics [11]. The calculated maximum growth rate of long-wavelength microturbulence is shown in Fig. 2.2.1–8. The $\vec{E} \times \vec{B}$ shearing rate is either equaled or exceeded at ρ ≈ 0.4 and ρ ≈ 0.7. Although the growth rates do not substantially increase with the application of EH, the shearing rate is reduced in the same region, indicating the loss of the ITB.

Beam emission spectroscopy (BES) measurements [12] of low-*k* fluctuation have been made in these discharges (Fig. 2.2.1–9). Consistent with the calculated shearing and growth rates, these observations indicate reduced fluctuations encompassing the transport barrier at ρ ≈ 0.5–0.6, and no reduction in the region where the shearing rate is matched or exceeded by the growth rate. Unfortunately, BES data in the region where the barrier was eliminated with EH was not obtained, but we would expect to have seen higher levels of turbulence indicated in this region.

A slight increase in the normalized electron temperature gradient a/L_{Te} is seen in the same region with application of EH (Fig. 2.2.1–7). As previously stated, calculation of the growth rate spectrum at ρ ≈

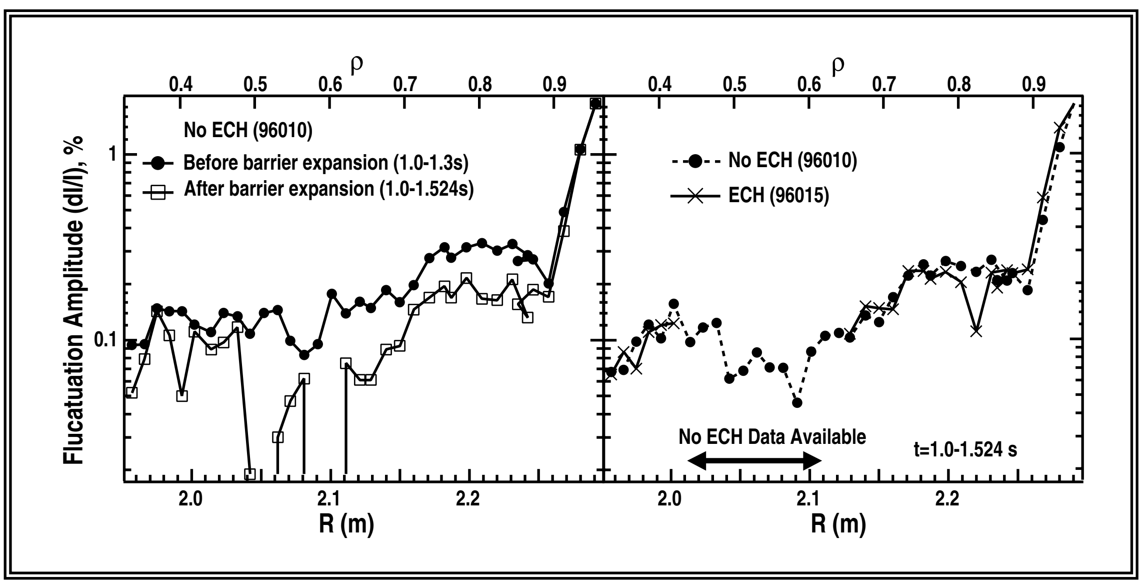

Fig. 2.2.1–9. BES measurements of low-*k* (≤2 0 cm^{-1}) fluctuations in discharges with and without EH. Data from the ECH discharge appears similar, but was not obtained in the region where we expect the biggest difference. Shots 96010 (no EH) and 96015 (1.1 MW ECH 1.1–1.6 s). I_P = 1.6 MA, B_T = –2.0 T, P_{NBI} = 4.3–5.7 MW.

0.6 indicates that the low-*k* (1–5 cm^{-1}) turbulence growth rates should be suppressed by $\vec{E} \times \vec{B}$ shear without ECH, and should become visible with ECH. An additional feature (Fig. 2.2.1–8) appearing in the ECH discharge at high-*k* (> 20 cm^{-1}) may indicate destabilization of electron temperature gradient (ETG) modes. It is believed that the ETG turbulence may be responsible for limiting a/L_{Te} in this case, but DIII–D has no diagnostics capable of observing such short-wavelength activity.

There appears to be a fundamental difference between the physics controlling transport around $\rho \approx$ 0.5–0.6 and that at smaller radii. Dramatic reductions are seen in the both the electron and ion temperature gradients of both discharges for $\rho \leq 0.2$ (Fig. 2.2.1–7). Application of ECH extends the strong electron temperature gradient region inward to $\rho \approx 0.1$, and appears to have no impact on the ion temperature gradient. Throughout the region near the magnetic axis, however, no drift wave turbulence is predicted unstable for either discharge, yet the temperature gradients are clearly limited by some other physical process. One clue to the controlling process may be the appearance of high-*k* (12 cm^{-1}) fluctuations in far-infrared (FIR) coherent scattering measurements [13] from the discharge with ECH (Fig. 2.2.1–10) at $\rho \approx 0.1$. These fluctuations, at small but measurable amplitude, rotate in the electron diamagnetic direction. At the same time in similar discharges, no signal was detected at either 6 or 9 cm^{-1}.

We have identified two candidates for at least some of the physics involved in this process. Both prospects are believed to preferentially impact transport in the electron channel. First, the BALLOO code [14] indicates instability to the resistive interchange mode in a small region centered at $\rho \approx 0.2$ in the ECH discharge (Fig. 2.2.1–8). How this mode should be manifested in the plasma is not known, but it might be consistent with the high-*k* fluctuation measurement. A second prospect is the appearance of a collisionless microtearing mode as proposed in Ref. [15]. The potential for these modes to be present in and have an impact on these discharges is currently being evaluated. In general, they are believed capable of appearance at short wavelengths, and may be highly localized in *k*-space. This could be consistent with FIR scattering measurements at short wavelengths.

In these experiments, electron heating was applied during the preheating phase, prior to the increase in neutral beam power that triggers an ITB growth phase. ECH or FW electron heating appears to limit development of the ITB during the high-beam-power phase of the discharge. The resulting state is one where the temperature gradients in the vicinity of the ITB are reduced compared to the no-EH case. The reduced gradients result in reduced temperatures at smaller radii and increased diffusivities, despite the fact that locally, the normalized temperature gradients may be the same. The important physics, then, is the destabilization of low-, and perhaps high-*k* turbulence at the ITB location. How this occurs as a direct consequence of the central electron

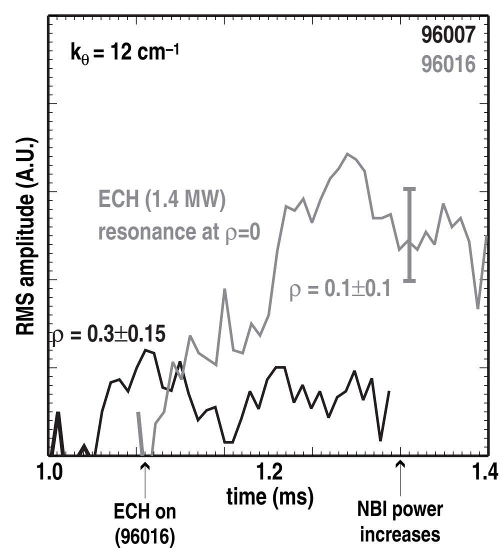

Fig. 2.2.1–10. FIR scattering detects a small signal at cm^{-1} during the ECH pulse (1.1 MW ECH 1.1–1.6 s). I_P = 1.6 MA, B_T = –2.0 T, P_{NBI} = 4.3–5.7 MW.

heating is uncertain. In addition, we see that another not yet positively identified process is at work nearer to the magnetic axis. This process may affect both the discharge with and that without additional electron heating. Investigation of this region will continue.

Summary. Internal transport barriers are routinely produced in DIII–D by applying moderate levels of neutral beam power to low density plasmas during the current ramp. The resulting elevated central electron temperature and the associated high conductivity of the core produce a current profile that is peaked off-axis, and exhibits negative central magnetic shear. This magnetic configuration is favorable for the elimination of MHD instabilities from the core, thereby allowing the formation of an ITB.

When the ITB forms, a feedback loop involving the steep pressure and rotation gradients leads to increased $\vec{E} \times \vec{B}$ shear, which leads in turn to reduced turbulence and transport and back to the gradients again. The final state is a region of very low transport that can encompass a large portion of the plasma. The evolution leading to this state, however, is highly dynamic, exhibiting transport events that have in some, but not all, cases been associated with the safety factor profile crossing through integer values. The transport events are associated with transient reductions in turbulence and highly localized transport barrier behavior that moves outward with the "steps."

Although our understanding of thermal transport in the ion channel has improved considerably, we have not yet come to the same level of understanding of the electron channel. In experiments where we probed the transport response to electron heating, both the ion and electron channels were impacted. This is partially due to a failure of the ITB to completely develop during the high power phase of the discharge. Both long- and short-wavelength turbulence are predicted to have been destabilized in the region where the ITB would have continued to develop in the ECH discharge. Without more complete measurements of fluctuations in the plasma, we cannot be certain that such turbulence actually appears in the experiment. The impact on the temperature profiles, however, appears consistent with the modeling. Other processes impacting transport closer to the magnetic axis have not yet been positively identified. Both experimental and modeling efforts to understand these effects will continue in the future.

Acknowledgements. M. Kotschenreuther of the University of Texas and W. Dorland of the University of Maryland contributed valuable insight in their discussions with one of the authors (G.M.S.).

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2.2.2. COMPREHENSIVE ENERGY TRANSPORT SCALINGS DERIVED FROM DIII–D SIMILARITY EXPERIMENTS[4]

Significant progress has been made recently towards predicting and understanding heat transport in L–mode and H–mode plasmas on DIII–D using the related methods of similarity and scale invariance. In these experiments, the dependences of transport on the relative gyroradius ($\rho_* \sim T^{1/2}/aB$),), plasma beta ($\beta \sim nT/B^2$), normalized collision frequency ($\nu \sim na/T^2$), and safety factor ($q \sim aB_T/RB_p$) are measured one at a time while keeping the other dimensionless parameters fixed (including those related to plasma shape and T_e/T_i). Experimentally determining the transport scalings in this way helps to distinguish between various proposed instability mechanisms of turbulent transport and permits a comprehensive energy confinement scaling relation to be developed that is founded in the principles of plasma physics. In addition, the T_e/T_i dependence of transport is being studied to test an important predicted scaling of theory-based transport models.

H–Mode Plasmas. The scalings of heat transport with ρ_*, β, ν, and q have been measured on DIII–D for H–mode plasmas. The results provide a strong experimental constraint on theoretical models of turbulent transport. Gyroradius scaling experiments in low q discharges have shown gyro-Bohm-like scaling for both the heat [1] and particle [2] transport, $B_{\tau E} \propto \rho_*^{-3.15\pm0.2}$. This scaling is consistent with the majority of anomalous transport theories that assume that the radial wavelength (or radial correlation length) of the turbulence scales with the Larmor radius. Other H–mode experiments have found energy confinement to have only a very weak beta dependence, $B_{\tau E} \propto \beta^{0.03\pm0.11}$, which favors theories of anomalous transport for which E×B transport is dominant over magnetic flutter transport [3]. The measured collisionality scaling falls between those of the collisionless ion temperature gradient (ITG) and collisionless trapped electron modes and that of the resistive ballooning mode [4], $B_{\tau E} \propto \nu\angle^{0.42\pm0.03}$. The ν scaling of the dissipative trapped electron and dissipative trapped ion modes was not observed.

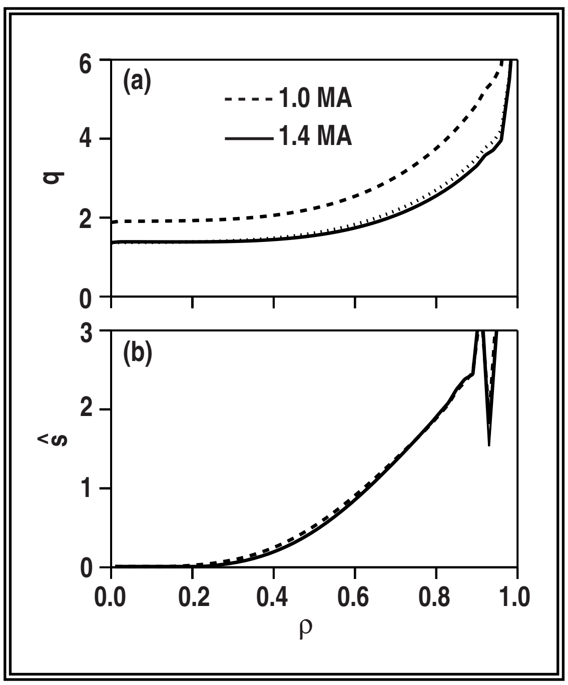

Fig. 2.2.2–1. Radial profiles of (a) safety factor, and (b) magnetic shear for H–mode discharges. The dotted line in (a) represents the 1.0 MA profile scaled to 1.4 MA.

Recent experiments on DIII–D have found a strong safety factor scaling of heat transport at all radii for H–mode plasmas [5]. In the first experiment, the safety factor was varied by a factor of 1.4 at fixed magnetic shear (see Fig. 2.2.2–1) while the other dimensionless parameters such as ρ_*, β, ν, and T_e/T_i were kept constant. The confinement time was found to scale like $\tau_E \propto q^{-2.42\pm0.32}$ for this case. A local transport analysis also found a strong safety factor dependence of the effective thermal diffusivity, as shown in Fig. 2.2.2–2, the magnitude of which agreed

[4]Petty, C.C., T.C. Luce, F.W. Baity, et al., "Comprehensive Energy Transport Scalings Derived From DIII–D Similarity Experiments," presented at 17th IAEA Fusion Energy Conference, October 19–24, 1998, Yokohama, Japan, to be published in a special issue of Nucl. Fusion; General Atomics Report GA–A22992 (1998).

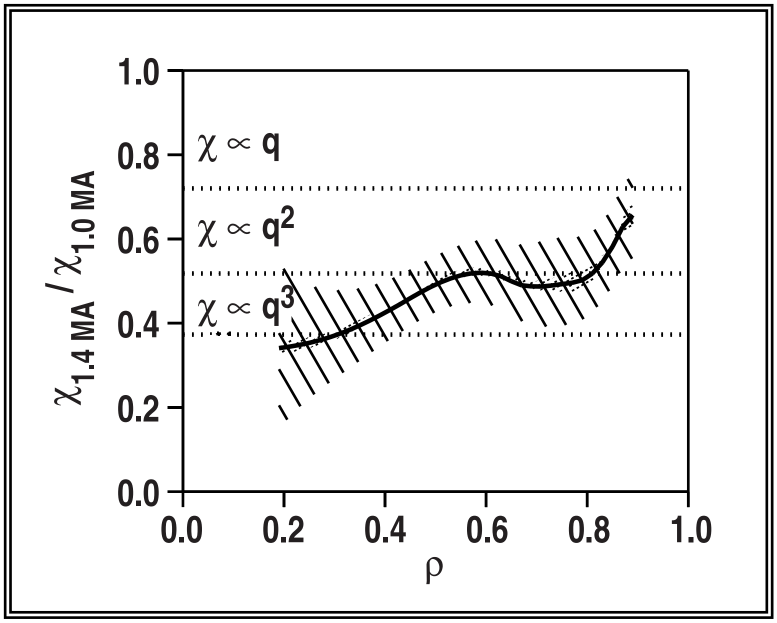

Fig. 2.2.2–2. Ratio of effective thermal diffusivities for H–mode discharges with fixed magnetic shear. The lined shading indicates the standard deviation of the random error.

with the scaling of the global confinement time. This transport scaling is close to the expected scaling of the resistive ballooning mode and is near to the upper limit of the scalings for the toroidal ITG mode and the collisionless trapped electron mode. In the second experiment, the safety factor and magnetic shear were both varied such that q_{95} was scanned at fixed q_0. A weaker confinement scaling was measured for this case, $\tau_E \propto q_{95}^{-1.43\pm0.23}$; this weaker scaling was attributed to the smaller variation in the volume-averaged q profiles rather than the change in the magnetic shear [5].

The combined ρ_*, β, ν, and q scalings of heat transport for H–mode plasmas on DIII–D reproduce the physical parameter dependences of empirical scalings derived from global confinement databases, with the possible exception of weaker power degradation. Converting a confinement scaling relation from dimensionless variables to physical (dimensional) variables is a straightforward algebraic manipulation. Assuming a power law form for the scaling relation, the dimensionless parameter scalings for low q H–mode plasmas on DIII–D can be summarized as

$$\begin{aligned} \tau_E &\propto B^{-1}\ \rho_*^{-3.15\pm0.2}\ \beta^{0.03\pm0.11}\ \nu^{-0.42\pm0.03}\ q_{95}^{-1.43\pm0.23} \\ &\propto I^{1.43\pm0.23}\ B^{0.66\pm0.38}\ n^{-0.39\pm0.11}\ T^{-0.70\pm0.16}\ L^{1.30\pm0.31} \\ &\propto I^{0.84\pm0.16}\ B^{0.39\pm0.20}\ n^{0.18\pm0.07}\ P^{-0.41\pm0.06}\ L^{2.00\pm0.24} \quad , \end{aligned} \tag{1}$$

where L represents the physical size scaling (*i.e.*, *a*, R, etc.) needed to make the scaling relation dimensionally correct. Thus, it can be seen that the dimensionless parameter scaling approach yields a definitive prediction for the size scaling of confinement from single machine experiments. For comparison, the confinement time derived from a dataset of H–mode plasmas on DIII–D and JET is [6,7]

$$\tau \propto I^{0.9}\ B^{0.3}\ n^{0.2}\ P^{-0.5}\ L^{1.5} \quad . \tag{2}$$

Comparing Eqs. (1) and (2) finds that the physical parameter scalings derived from DIII–D similarity experiments agree with those derived from a regression analysis of multi-machine confinement databases to the 2σ level. Another interesting comparison can be made using a confinement scaling for ELM-free H–mode plasmas that is nearly dimensionally correct [8],

$$\tau_{\mathrm{ITER-93H}} = 0.036\ \mathrm{I}^{1.06}\ \mathrm{B}^{0.32}\ n_{19}^{0.17}\ \mathrm{P}^{-0.67}\ \mathrm{R}^{1.9}\ a^{-0.11}\ \mathrm{A}^{0.41}\ \kappa^{0.66} \quad . \tag{3}$$

A comparison of Eqs. (1) and (3) finds that the B, n, and size scalings agree to within 1σ, while the difference in the I scalings is only a little larger. The main discrepancy is in the power scaling, where the DIII–D experiments find a weaker power degradation than ITER-93H (owing partially to the weaker beta scaling), leading to a more optimistic projection for H–mode confinement on larger machines [9].

L–Mode Plasmas. The dependences of heat transport with ρ_*, β, and ν also have been measured for L–mode plasmas on DIII–D. The ρ_* scalings of the electron and ion heat transport were measured separately [10], with the electron diffusivity scaling gyro-Bohm-like, $\chi_e \propto \chi_B\, \rho_*^{1.1\pm0.3}$, and the ion diffusivity scaling worse than Bohm-like, $\chi_i \propto \chi_B\ \rho_*^{-0.5\pm0.3}$. Here, χ_B = T/eB is the Bohm diffusion coefficient. The scaling of the global confinement time could vary from gyro-Bohm-like to Bohm-like depending upon whether the electrons or ions dominated the heat transport. The beta scaling of energy confinement was close to zero, $\beta\tau_E \propto \beta^{-0.05\pm0.10}$, with the electron and ion thermal diffusivities having the same scaling to within the experimental errors [3]. The scaling of energy confinement with collisionality in the banana regime was also close to zero, $\beta\tau_E \propto \nu^{-0.02\pm0.03}$, with the electron and ion heat transport again having the same scaling to within the experimental uncertainties [4].

By combining the ρ_*, β, and ν scalings, the power degradation and density scaling of energy confinement can be uniquely determined for L–mode plasmas. However, this calculation is complicated by the fact that the ρ_* scalings of the electron and ion thermal diffusivities are not the same. If we limit ourselves to the typical case of approximately equal electron and ion heat conduction, then the global confinement exhibits Bohm-like scaling, and the scaling of the energy confinement time in physical parameters is

$$\begin{aligned}\tau_E &\propto \mathrm{B}^{-1}\ \rho_*^{-2}\ \beta^{-0.05\pm0.10}\ \nu^{0.02\pm0.03} \\ &\propto n^0\ \mathrm{T}^{-1.1} \\ &\propto n^{0.5}\ \mathrm{P}^{-0.5} \quad .\end{aligned} \tag{4}$$

The B and I dependences of τ_E cannot be determined until the safety factor scaling of transport is measured for L–mode plasmas. Comparing Eq. (4) with the commonly used ITER-89P L–mode scaling relation [11],

$$\tau_{\mathrm{ITER-89P}} = 0.048\ \mathrm{I}^{0.85}\ \mathrm{R}^{1.2}\ a^{0.3}\ \mathrm{n}_{20}^{0.1}\ \mathrm{B}^{0.2}\ \mathrm{A}^{0.5}\ \kappa^{0.5}\ \mathrm{P}^{-0.5} \quad , \tag{5}$$

one sees that the power degradation factors are the same but Eq. (5) has a weaker density scaling than what was measured on DIII–D.

T_e/T_i Dependence of Transport. In order to further differentiate between various theory-based transport models, the scaling of transport with T_e/T_i is also being studied. Experiments in L–mode plasmas with internal transport barriers (ITB) on DIII–D have shown that intense electron heating, using either fast waves or electron cyclotron heating, in a beam heated plasma with $T_i \gg T_e$ increases the electron and ion thermal diffusivities and slows the plasma rotation [12]. Further experiments on DIII–D have studied the T_e/T_i dependence of heat transport in ELMing H–mode plasmas without ITBs and with $T_e \sim T_i$. In these experiments, increasing the ratio of T_e/T_i at fixed beta resulted in an increase in both the electron and ion heat transport as well as the particle transport. This result, combined with related H–mode experiments that varied T_e at fixed T_i, and *vice versa,* can be summarized as tE $\langle T_i \rangle^2/\langle T_e \rangle^2$. This strong scaling may be limited to the conditions near those measured. In addition, since the toroidal rotation also decreased with increasing T_e/T_i, some of the transport change may be only indirectly related to T_e/T_i owing to the small decrease in $\omega_{E\times B}$ with electron heating.

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2.2.3. COMPARISON OF L–H TRANSITION MEASUREMENTS WITH PHYSICS MODELS[5]

Global scaling of the H–mode power threshold (P_{TH}) and local conditions at the edge of the plasma just before an L-H transition have been studied in the DIII–D tokamak. Besides the usual dependence on density and toroidal field, at least three other effects have been found to have a significant influence on P_{TH}. These include: the effect of a sawtooth crash, which can trigger an L-H transition; the direction and magnitude of the ion ∇B drift relative to the X–point location, which can change P_{TH} by factors of 2 to 3; and the effect of neutrals, which may have more subtle and counter-intuitive effects on P_{TH}. In our analysis, P_{TH} is defined as the power flowing across the separatrix, P_{SEP}. Each of these effects has been studied experimentally and compared with physics models or numerical calculations. In addition, parameters measured at the plasma edge just before an L-H transition have been analyzed and compared to theories of the L-H transition. Operational space of L– and H–mode is given in terms of dimensionless edge parameters. It is found that the edge pressure gradient may be more important than the magnitude of the edge temperature.

Sawteetch Effects. Over half of the L-H transitions in the DIII–D transition database are triggered by sawteeth. The sawtooth crash provides an additional transient power flow to the edge of the plasma where the L-H transition takes place. This power flow depends on the inversion radius of the sawtooth, the stored energy, and the dissipation of the power as it flows to the plasma edge. In an experiment in which the sawteeth were suppressed by neutral beam heating during the early current ramp phase of the discharge, P_{TH} increased from 3 MW in the sawtooth triggered case to 5 MW when the sawteeth were suppressed, (reverse B case in Fig. 2.2.3–1). Including the additional power flow to the plasma edge due to sawteeth [1] in the calculation of the power flowing across the separatrix, P_{SEP}, we find the toroidal field dependence of P_{TH} is weakened. Thus edge power flow due to sawteeth may significantly influence the observed P_{TH} scaling.

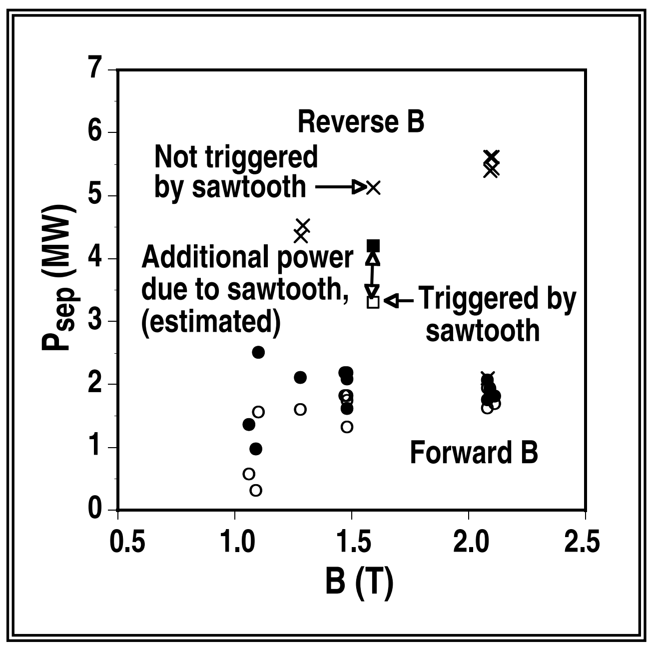

Fig. 2.2.3–1. Toroidal field scaling of the H–mode power threshold when accounting for sawteeth power. Open symbols indicate L–H transitions triggered by a sawtooth crash, closed symbols indicate threshold power when additional sawteeth power is taken into account, crosses indicate transitions not triggered by sawteeth. Forward B data < 3 MW, reverse B data > 3 MW.

∇B Drift Effects. The direction of the ion ∇ B drift relative to the X–point location has a dramatic influence on the magnitude of P_{TH}. Hinton [2] and later Hinton and Staebler [3] have attributed this effect to neoclassical cross-field fluxes of both heat and particles driven by poloidal temperature gradients on the open field lines in the scrape-off-layer (SOL). The magnitude of these fluxes scale like $\sim(n/r)(T/B)(\partial T/\partial\vartheta)$, where r is the minor radius, T the temperature, and ϑ the poloidal angle. The flux surface average of these cross-field fluxes

[5]Carlstrom, T.N., K.H. Burrell, R.J. Groebner, et al., "Comparison of L–H Transition Measurements with Physics Models," presented at 17th IAEA Fusion Energy Conference, October 19–24, 1998, Yokohama, Japan, to be published in a special issue of Nucl. Fusion; General Atomics Report GA–A22989 (1998).

is zero unless asymmetries such as the gradient of B and/or the poloidal temperature gradient lead to a net flux. In its simplest form, these fluxes influence P_{TH} by either adding to or subtracting from the power flow to the edge of the plasma. A 1D analysis of heat conduction in the SOL suggests that these cross-field fluxes can be a significant fraction of the input power [4]. It was proposed that some of the observed scaling of P_{TH} is due to the variation of the magnitude of these fluxes and may not be intrinsic to the scaling of the physics of the L–H transition itself. For instance, the increase of P_{TH} at low density may be due to the reduction of the ∇B effect as the sheath limit for parallel heat conduction is reached and the poloidal temperature gradient is reduced. Many qualitative features of this model are in agreement with observations of P_{TH} scaling, such as the existence of a density threshold, the importance of the X–point position, and the increase of P_{TH} in double-null configurations.

In order to further test these ideas, a series of experiments was carried out in which plasmas with identical operational parameters except for the direction of the toroidal field were compared. In these discharges, the neutral beam power was modulated at a low level (12.5%, 0.3 MW) in order to keep the plasma in L–mode in the forward B case (∇B drift toward the X–point). This resulted in power levels far below P_{TH} in the reverse B case where P_{TH} ~5 MW. Motivated by the idea that edge parameters control the L-H transition, we compare the edge n_e, T_e, T_i, and ∇P_e profiles evaluated at the pedestal of the density profile determined by a hyperbolic tangent fit [5] in Fig. 2.2.3–2. There is almost no difference in the value of these parameters between the two directions of the toroidal field, even though one discharge is very near the L-H transition and the other is very far away in terms of power. Also shown in Fig. 2.2.3–2, are the edge parameters for the reverse B case, when the power level is just below the threshold, (5 MW). Although the edge density remains the same, (the line average density was held constant), the edge temperatures and pressure gradients are much greater than the forward B case.

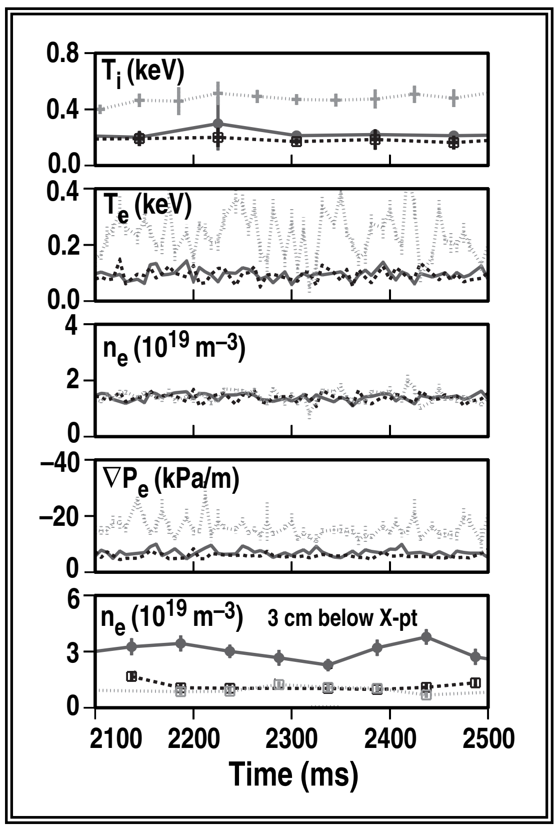

Fig. 2.2.3–2. Edge and divertor parameters for forward (solid) and reverse B (dashed) at 1 MW and reverse B at 5 MW (light dashed).

Preliminary analysis of the divertor conditions show that significant differences between these discharges appear near the X–point region. The electron density just below the X–point measured by Thomson scattering in the forward B case is 4-5 times greater than the reverse B case, as shown in Fig. 2.2.3–2. The cause of this high-density region and its influence on the L-H transition is under investigation. It may be evidence of the ion ∇B drift carrying heat and particles across the X–point into the private flux region, or it may be the result of E×B flows in the divertor. If neutral penetration into the core plasma raises P_{TH} as discussed in the next section, then this high-density region may reduce P_{TH} in the forward B case by preventing neutrals from reaching the X–point region of the core plasma.

Several theories of the L-H transition consider the edge pressure gradient as a key parameter for the transition (Section 5). As shown in Fig. 2.2.3–2, the forward

B edge electron pressure gradient is slightly higher than the reverse B case at 1 MW. This may be evidence for cross-field fluxes in the SOL playing a role in determining the edge pressure gradient. However, calculations of the cross-field fluxes described above, based on measured SOL and divertor temperatures and densities, result in powers of only a few tens of kilowatts. These fluxes are considered to be too small to contribute significantly to the overall power balance. However, it is still possible that these fluxes affect the edge plasma, especially near the X–point, and influence the L-H transition threshold.

Neutrals. The effect of neutrals on the L-H transition has been studied in a series of experiments where a heavy gas puff was used to ramp up the density and a divertor cryopump was used to ramp down the density during an L-H transition. Extensive transport and neutral modeling of the plasma edge region using B2.5 and DEGAS indicates that during heavy gas puffing, the SOL density increases and shields the region just inside the separatrix from neutrals [6]. This reduces the neutrals in this region and lowers P_{TH}. When the cryopump is used, the neutral penetration is greater and P_{TH} increases. There is a good correlation between $P_{SEP}/\bar{n}$ and the ratio of the maximum charge exchange damping rate $(\nu_{cx})_M$ to the neoclassical damping rate (μ_{neo}) of the poloidal flow when evaluated for average radii in the range $0.9 < r/a < 0.95$ as shown in Fig. 2.2.3–3 [6]. Good correlation is also found with the poloidally averaged neutral decay length. Further experiments and inter-machine comparisons are needed to identify the proper dimensionless parameter for the effect of neutrals on the power threshold.

Local Edge Parameters. A technique of fitting a hyperbolic tangent to the edge profiles themselves has eliminated the scatter caused by the flux surface reconstruction [7] and has improved the localization of the plasma edge [5]. With this technique, we have determined that the position of the maximum edge density gradient remains relatively constant across the L-H transition, and is therefore, a good location to evaluate the local edge conditions relevant to the formation of the edge transport barrier in H–mode. However, in order to facilitate comparisons with other devices, we have evaluated edge parameters 2 cm inside the separatrix. We find this location roughly corresponds to the edge density pedestal determined from the hyperbolic tangent fit. An operational space diagram of T_e and n_e evaluated 2 cm inside the separatrix is shown in Fig. 2.2.3–4. Although there is a trend for pre-transition data (LH) to be at higher temperatures, these data are not well separated from the normal L–mode data. Therefore, these parameters do not clearly resolve the L–H transition operating space. For comparison, a fit to the L-H data on ASDEX-Upgrade is also shown. The DIII–D data generally fall a factor of 2 below the ASDEX-Upgrade data, indicating that the edge temperature alone is not a critical parameter for the L-H transition. Collisionality of the edge plasma varies in the range of 5–50, and often increases slightly after the L-H transition as the edge density rises. Collisionality alone is therefore, not likely to be a key parameter.

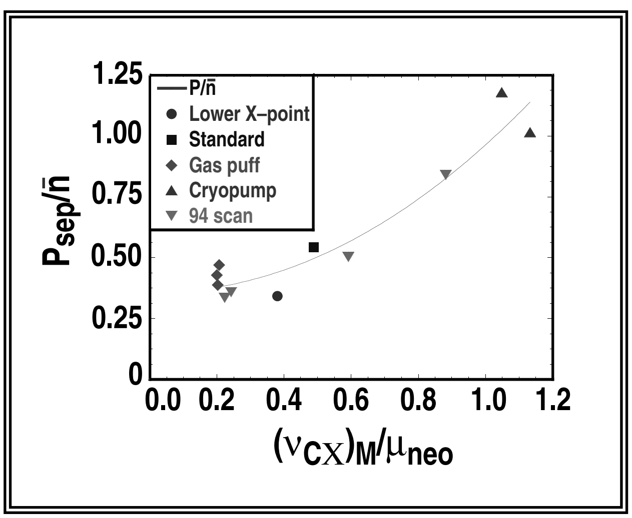

Fig. 2.2.3–3. The dependence of $P_{SEP}/\bar{n}$ versus $(\nu_{CX})_M/\mu_{neo}$ for the r/a =0.95 surface (taken from Fig. 18, Ref. [6]).

The improved localization of the edge parameters now permits more detailed comparisons with L-H transition theories. In a model based

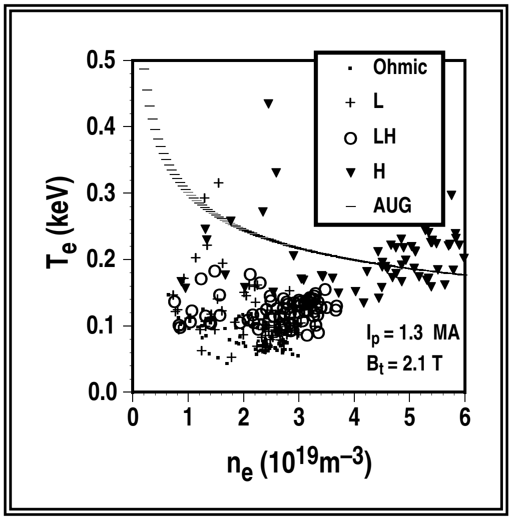

Fig. 2.2.3–4. Operational space diagram evaluated 2 cm inside the separatrix. A fit to LH data from ASDEX-Upgrade is shown for comparison.

on 3D simulations of edge turbulence by Rogers and Drake [8], the threshold condition is parameterized in terms of α_{MHD} and α_{DIAM}, both of which contain edge gradients. Figure 2.2.3–5 shows that α_{MHD} may provide a better separation of the L–mode and pre-transition data than edge n_e and T_e in Fig. 2.2.3–4, indicating it may be important for the L-H transition. Due to the lack of separation of the data with α_{DIAM}, the importance of this parameter is not clear. Quantitative comparisons will require improvements in the model to include realistic geometry.

In another model of the L-H transition based on the stabilization of Alfvén drift waves by O. Pogutse *et al*. [9], the threshold condition is parameterized by a normalized beta and collision frequency such that $\beta_n > \beta_{crit} = 1 + \nu_n^{2/3}$. Figure 2.2.3–6 shows data evaluated at the maximum edge density gradient on the $\beta_n - \nu_n$ plane. The value of β_n has about the right magnitude but no clear distinction exists between points just before the L-H transition and points that remain L–mode or Ohmic. H–mode points, taken just after the L-H transition, are well above the threshold condition in both these models. Therefore, comparison of the edge gradients between L– and H–mode is not particularly useful in distinguishing among these models.

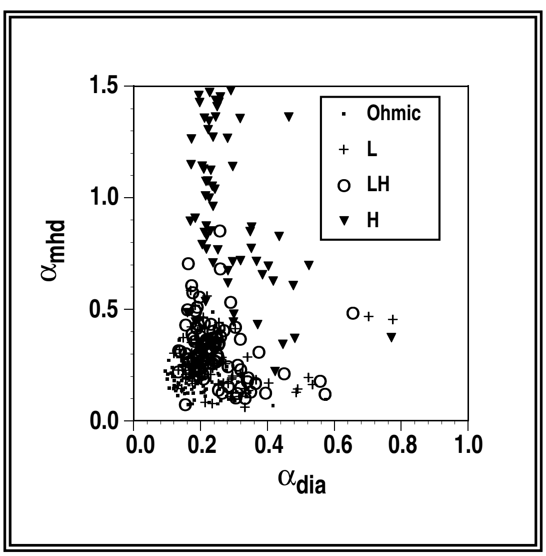

Fig. 2.2.3–5. Operational space diagram for critical parameters of Ref. 8.

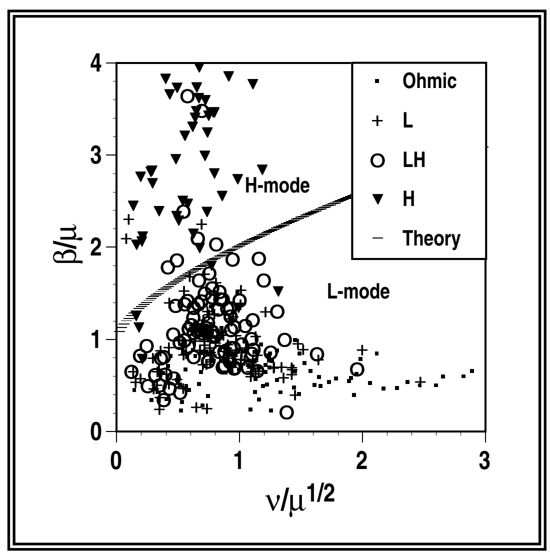

Fig. 2.2.3–6. Operational space diagram for critical parameters of Ref. 9.

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2.2.4. RI–MODE INVESTIGATIONS IN THE DIII–D TOKAMAK WITH NEON AND ARGON INDUCED RADIATING MANTLES[6]

The RI–mode regime, with high radiating power fractions from 0.5 to 0.9, energy confinement enhancements, H_{89P}, over ITER89-P L–mode scaling greater than 1.6, and operation at or above the Greenwald density limit (n_{GW}) is an attractive operating scenario for future fusion burning plasma devices. The TEXTOR tokamak has demonstrated this scenario in a limiter device with steady state conditions, $\Delta t_{RI\text{-}mode}/\tau_E > 100$ [1]. Studies have been initiated on the DIII–D tokamak with the goals of: (a) extending these results to a larger non circular machine (providing size and shape scaling), (b) investigating the underlying physical mechanisms of RI–mode with a complementary diagnostic set to that on TEXTOR, and (c) using non-intrinsic impurities, e.g. neon and argon, to obtain high performance diverted discharges, ($\beta_N H_{89P} > 6$) in support of the DIII–D advanced tokamak (AT) program, where $\beta_N = \beta_T/(I_p/aB_T)$ and β_T, I_p, a, and B_T are toroidal beta (in %), plasma current (MA), minor radius (m), and toroidal magnetic field (T) respectively. We define $P_{radLCFS}$ as the radiated power inside the LCFS and note that nearly all of this radiation occurs in the mantle region $0.6 < \rho < 1.0$, i.e., $P_{mantle} \approx P_{radLCFS}$. Three types of DIII–D discharges where mantle radiation plays a significant role are discussed in this paper: (i) ELMing H–mode "puff and pump," (ii) limiter L–mode, and (iii) high performance.

The first type of discharge with a high fraction of mantle radiation is obtained by puffing deuterium above the midplane into a lower single null shape, where the outer strike point is positioned at the entrance to the DIII–D toroidally continuous cryopump. With argon injection into the divertor region, this technique, termed "Puff and pump" [2], produced ELMing H–mode discharges and under certain conditions there was a significant fraction of radiated power from within the LCFS at high normalized densities, $n_e/n_{GW} <= 0.95$ as shown in Fig. 2.2.4–1. In this discharge, there is a marked increase in the rate of density rise and an increase in the fraction of radiation beginning at 3000 ms, although external parameters are constant, e.g. gas flow, I_p, B_T, q_{95}, P_{NB}, and discharge shape. Both density and confinement continue to increase until the argon puff is terminated at 4000 ms. Energy confinement shown in Fig. 2.2.4–1, f_{H93}, has been normalized to the ITER H93 ELM free confinement scaling relation. Coincident with the decrease in confinement and density, an increase in MHD activity is observed which generally coincides with the end of the density increase in the radiating mantle phase of argon "puff and pump" discharges and in most cases occurs

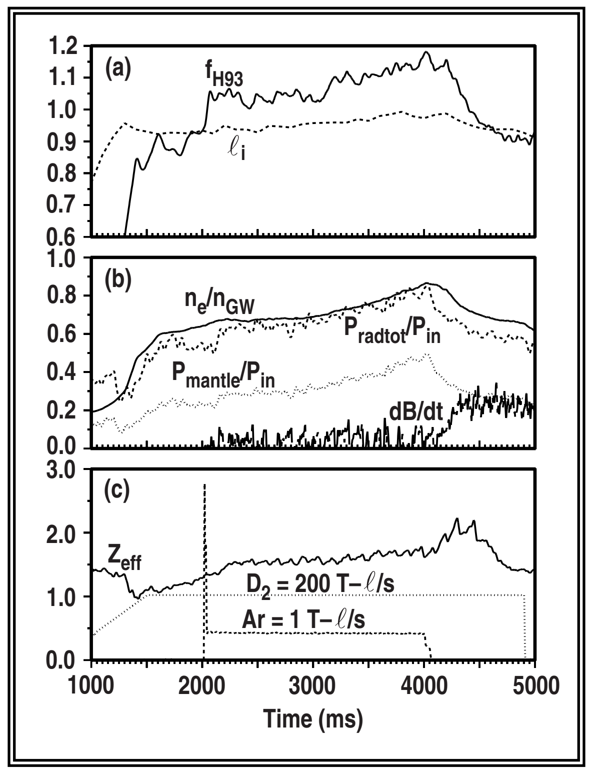

Fig. 2.2.4–1. Temporal evolution of a high density radiating mantle discharge. An increase in the rate of density rise begins at $t \approx 3000$ ms. Argon is injected from 2000 to 4000 ms (I_p=1.35 MA, B_t = 2.1 T, P_{NB} = 6.4–7.3 MW).

[6]Jackson, G.L., G.M. Staebler, M. Murakami, et al., "RI–mode Investigations in the DIII–D Tokamak with Neon and Argon Induced Radiating Mantles," Proc. 25th European Conf. on Controlled Fusion and Plasma Physics, June 29–July 3, 1998, Zofin, Praha, Czech Republic, Vol. 22C, p. 810 (European Physical Society, 1998); General Atomics Report GA–A22911 (1998).

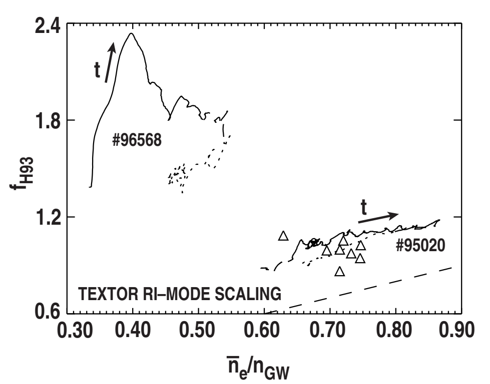

Fig. 2.2.4–2. Normalized confinement as a function of normalized density for the 3 types of discharges: "puff and pump" temporal behavior (#95020), IWL L–mode (points from several discharges), and high performance H–mode (#96568). Solid lines are times during impurity puffing, dashed are before or after, and dots indicate times of significant MHD. The TEXTOR RI–mode scaling relation is also displayed.

before the termination of the argon puff. This increase in MHD is most likely associated with the onset of the m/n=3/2 neoclassical tearing mode [3], and further experiments are planned to examine this in more detail.

The temporal behavior of confinement as a function of normalized density is shown in Fig. 2.2.4–2 for #95020 (same as Fig. 2.2.4–1). Similar response has also been observed in TEXTOR [1], although confinement in these ELMing H–mode divertor DIII–D discharges is substantially above the TEXTOR limiter RI–mode scaling, $\tau_E = \tau_{93H} \bullet n_e/n_{GW}$ at the same normalized density, plotted in Fig. 2.2.4–2. After the increase in MHD fluctuations, shown in Fig. 2.2.4–2, the normalized confinement enhancement markedly decreases.

The increase in the rate of density rise in these puff and pump discharges is usually accompanied by increased toroidal rotation, shown in Fig. 2.2.4–3. This radiation increase is similar to the spin-up first observed in VH–mode discharges [4], although the absolute magnitude is lower in these radiating mantle discharges. The MHD activity increases at the time the toroidal velocity begins decreasing, t ≈ 3700 ms.

The puff and pump discharges described here exhibit a heat flux reduction of more than a factor of 2 to the divertor tiles when compared to similar discharges with no impurity radiation. We also note that these ELMing H–mode discharges were not detached. However there is a reduction of the edge pedestal electron pressure as the fraction of mantle radiation increases, shown in Fig. 2.2.4–4.

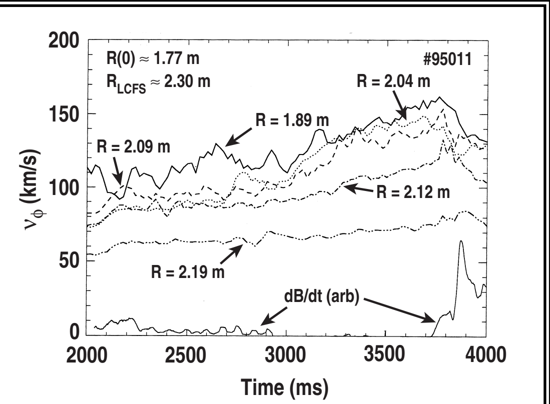

Fig. 2.2.4–3. Toroidal plasma rotation from C^{+6} CER. Discharge conditions are similar to Fig. 1. A "spin up" begins at t=2900 ms. Rotation begins to decrease at approximately the time that MHD activity (also shown) increases, t ≈ 3700–3800 ms.

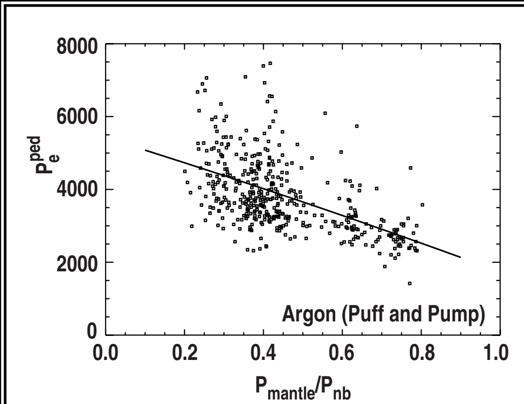

Fig. 2.2.4–4. Edge electron pedestal pressure derived from fits to the H–mode edge pressure profile [8] measured by Thomson scattering. All points are from puff and pump discharges during the time argon was injected. The solid line is a least squares second order polynomial fit to the data.

Normalized confinement compares favorably with discharges without neon or argon radiation, shown in Fig. 2.2.4–5, where radiating mantle discharges are compared with a subset of the DIII–D ELMing H-mode ITER database with similar parameters (I_p = 1.2–1.4 MA, B_T > 1.65 T, P_{NB} > 3 MW). The addition of non-intrinsic impurities allows an extension to higher density and confinement generally higher than the comparative ITER database discharges. Some of the discharges with neon injection shown in Fig. 2.2.4–5 had transient VH–mode and ELM free phases which generally exhibit higher confinement. The argon puff and pump series were either stationary ELMing H–mode, similar to those in the DIII–D ITER database, or slowly evolving ELMing H–mode, such as in Fig. 2.2.4–1.

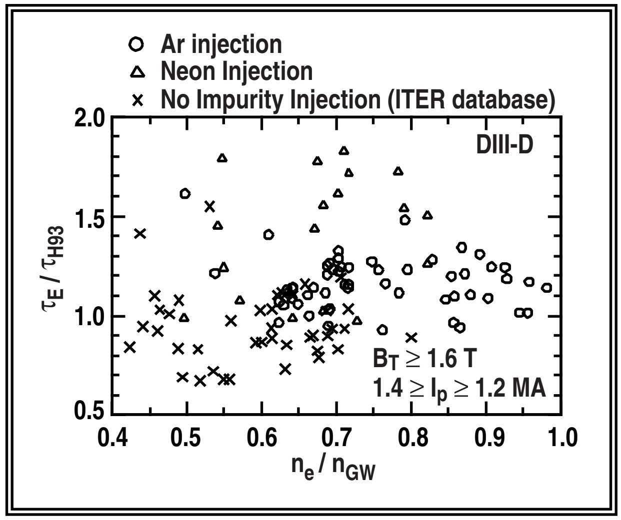

Fig. 2.2.4–5. A comparison of confinement with impurity puffing to ELMing H–mode discharges from the DIII–D ITER database discharges.

The second type of DIII-D discharge with a radiating mantle is inner wall limited L–mode. Neon was injected (above the midplane) into L–mode inner wall limited (IWL) discharges (I_p =1.2–1.4 MA, B_T = 1.7-2.1T, q_{95} = 2.8–3.8, P_{NB} = 6–9 MW). Due to technical constraints, operation on the DIII–D outer bumper poloidal limiters is severely limited so the TEXTOR configuration could not be reproduced exactly (TEXTOR RI–mode discharges are limited on the ALTII pumped limiter located below the outer midplane). Nevertheless, this series of DIII–D discharges had features similar to the TEXTOR RI–mode [5]. To date, MARFing has limited the maximum density achieved in DIII–D L–mode IWL discharges to $n_e/n_{GW} \leqq 0.75$ which is the lower normalized density range observed by TEXTOR. This is probably a consequence of the differences in the limiting surfaces between the two machines. As shown in Fig. 2.2.4–2, confinement at the same normalized density is higher than the TEXTOR scaling relation plotted in Fig. 2.2.4–2. We also note that long duration L–mode IWL discharges without MARFing have not yet been demonstrated in DIII–D.

The third type of discharge where mantle radiation can be important is high performance H– and VH–mode discharges. Impurity seeding has been used to obtain high performance discharges with $\beta_N H$ > 10 for 0.55 s and $\beta_N H$ > 6 for 1.6 s (#96568). In the latter case, operation at the DIII–D empirical stability limit [6], $\beta_N \approx 4\ell_i$ was maintained for 1.4 s, where ℓ_i is the normalized internal plasma inductance. Neon injection has also allowed VH–mode at the highest target density ever achieved, $6{\times}10^{19}$ (#93450), nearly a factor of 2 higher than the normal target density for the L to H transition and VH–mode confinement [5].

Mantle radiation in these high performance discharges is substantially lower than the discharges described previously. For example, in the two discharges mentioned above, the fraction of mantle radiation was $\lesssim 0.25$. However, this is still substantially higher than standard VH–mode discharges, where $P_{mantle}/P_{in} < 0.1$. Although $Z_{eff}(0)$ is low in these discharges, <2, higher impurity concentrations are observed in the mantle region. This increase in Z_{eff} can provide a stabilizing effect for electron temperature gradient (ETG) modes, allowing increases in particle and energy confinement. Such suppression has been observed in some DIII–D discharges with neon puffing [7].

In conclusion, non-intrinsic impurities have been used in DIII–D to simultaneously obtain good confinement ($\tau_E/\tau_{ITER\text{-}93H}$ >1), high density (n_e/n_{GW} > 0.8), significant mantle radiation (P_{rad}/P_{in} > 0.4), and reductions in peak heat flux to the walls of more than a factor of 2. L–mode limiter discharges have been obtained under similar conditions, but at lower normalized density. In addition, discharges with $\beta_N H$ of greater than 6 have been observed with impurity radiation. We have focused in this paper on the role of impurities inside the separatrix flux surface. Radiating divertor discharges have also been achieved in DIII–D and have been reported elsewhere [2].

The role of increased toroidal rotation in the argon puff and pump experiments can contribute to enhanced confinement, similar to the VH–mode. The increased rotation has also been observed in some DIII–D L–mode IWL discharges such as those discussed in this paper. Whether this "spinup" is a fundamental characteristic of the TEXTOR RI–mode and radiating mantle DIII–D discharges is currently under investigation. The plasma rotation may also be important in maintaining a low central Z_{eff}. For example, impurity concentrations in the argon puff and pump experiments were sufficiently low that $Z_{eff} \lesssim 2$ during the phase of increasing density. However the onset of the m/n=3/2 MHD activity and a decrease in rotation shown in Fig. 2.2.4–3 is accompanied by a rapid increase in Z_{eff}.

Future work will focus upon obtaining stationary radiating mantle discharges, identifying the underlying physical mechanisms, and determining the feasibility and advantages of employing radiating mantle operation in future fusion burning plasma devices such as ITER.

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2.3. STABILITY AND DISRUPTION PHYSICS

2.3.1. OBSERVATION AND CONTROL OF RESISTIVE WALL MODES[7]

Stabilization of low-n kink modes by a conducting wall is crucial for high beta, steady state “advanced tokamak” scenarios. Operation at high beta allows a more compact and economical fusion plasma with a large fraction of bootstrap current. Good alignment of the bootstrap current with the equilibrium current density profile, important for minimizing the requirements on external current drive systems, is achieved with broad current density profiles and broad pressure profiles. Such broad profiles have a low beta limit in the absence of a wall, but strong coupling to a nearby conducting wall can improve the stability limit by as much as a factor of 2 or 3 [1–3].

Two approaches to achieving long-time scale stabilization with a real, finite conductivity wall are being considered: plasma rotation and active feedback control. Ideal MHD theory predicts that for a plasma which would be stabilized by an ideal wall, non-zero wall resistivity leads to an unstable “resistive wall mode” with a growth time on the order of the wall’s magnetic field penetration time τ_W and a real frequency $\omega \sim \tau_W^{-1}$, and which is not stabilized by sub-Alfvenic plasma rotation. However, more detailed theories show that the addition of dissipation in the plasma allows stabilization by sub-sonic plasma rotation [4,5]. Furthermore, external kink modes can drive islands in a resistive plasma, allowing stabilization by plasma rotation frequencies as low as $\Omega \sim \tau_W^{-1}$ [6,7].

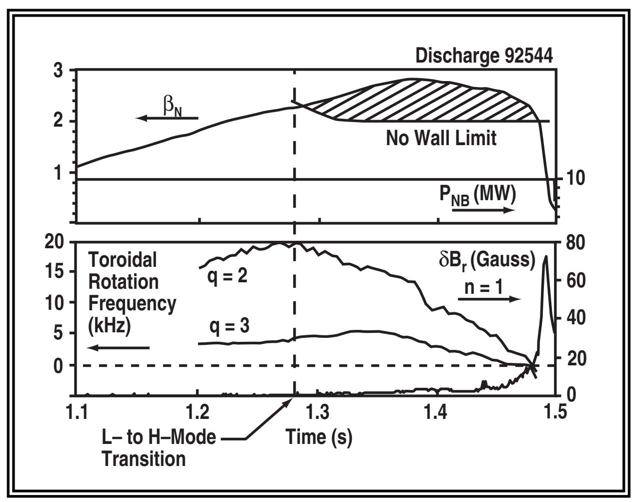

Fig. 2.3.1–1. Time evolution of a wall-stabilized DIII–D discharge (92544). (a) Normalized beta, bN=b(aB/I) and neutral beam power. (b) Plasma rotation frequency from charge exchange recombination spectroscopy at two radial locations, and the Br amplitude of the non-rotating n=1 mode from the saddle loop array.

DIII–D experiments [8,9] confirm many of the important qualitative features of these more recent theories. In discharges with broad current density profiles, beta values reach up to 1.4 times the ideal n=1 kink mode limit calculated without a wall, but remain within the stable range calculated with an ideal wall at the position of the DIII–D vacuum vessel. Beta greater than the no-wall limit has been sustained for up to 200 ms, much longer than the wall penetration time $\tau_W \leq 6$ ms, which indicates that the resistive wall mode has been stabilized [Fig. 2.3.1–1(a)]. As the rotation slows, these plasmas are typically terminated by an m=3, n=1 mode which has a growth time of 2–8 ms and a real frequency $\omega \sim \tau_W^{-1}$, as expected for a resistive wall mode. The mode typically begins to grow as the plasma rotation at the q=3 surface decreases below 1–2 kHz, consistent with a loss of rotational stabilization [Fig. 2.3.1–1(b)].

In many cases, temperature profiles measured with electron cyclotron emission show an ideal-like mode structure, without islands (Fig. 2.3.1–2), as expected for an ideal kink mode which has lost its wall

[7]Strait, E.J., A.M. Garofalo, M.E. Austin, et al., “Observation and Control of Resistive Wall Modes,” presented at 17th IAEA Fusion Energy Conference, October 19–24, 1998, Yokohama, Japan, to be published in a special issue of Nucl. Fusion; General Atomics Report GA–A22994 (1998).

stabilization. The broad displacement of the T_e profile in Fig. 2.3.1–2 is consistent with a global kink mode structure, and the radial displacement of ~1 cm in the outer part of the profile is consistent with the measured mode amplitude of ~50 G at the wall. By itself, a single T_e profile measurement cannot conclusively rule out the existence of an island. However, in this and other discharges, the outer portion of the T_e profile rises or falls consistent with an ideal MHD mode structure, given the toroidal phase inferred from magnetic measurements. The temperature perturbation profile agrees well with predictions by the GATO stability code. The growth of a stationary mode in the presence of significant plasma rotation also indicates the absence of islands. (In some cases, electron cyclotron emission and beam emission spectroscopy measurements do show evidence of stationary island formation, but at beta below the ideal no-wall limit.)

Rotational Stabilization. Plasma rotation is one possible means for long time-scale stabilization by a resistive wall. Vacuum field measurements show that the DIII–D vacuum vessel wall penetration time for an imposed n=1 radial magnetic field can be approximated by a 2-pole response with time constants of 7 ms and 1–3 ms. This agrees well with calculations using the SPARK 3D electromagnetic code which show that the time constant for the lowest n=1 eigenmode of the DIII–D vacuum vessel is about 5.8 ms, followed by about 3 ms for the next eigenmodes. Stabilization for longer times in the experiment indicates that plasma rotation is important.

The existence of a critical rotation frequency for stabilization is clearly demonstrated by a series of reproducible discharges in which the rotation rate was modified through magnetic braking by an applied magnetic error field. As the magnetic braking field was increased [Fig. 2.3.1–3(a)], the plasma rotation decelerated more rapidly, and the onset of the resistive wall mode occurred earlier, corresponding to a fixed value of the rotation [Fig. 2.3.1–3(b)].

The experimental data allow us to distinguish at least qualitatively between predicted mechanisms for stabilization. The observed critical rotation frequency $\Omega=2\pi f\sim10^4\ s^{-1}$ at the q=3 surface disagrees with

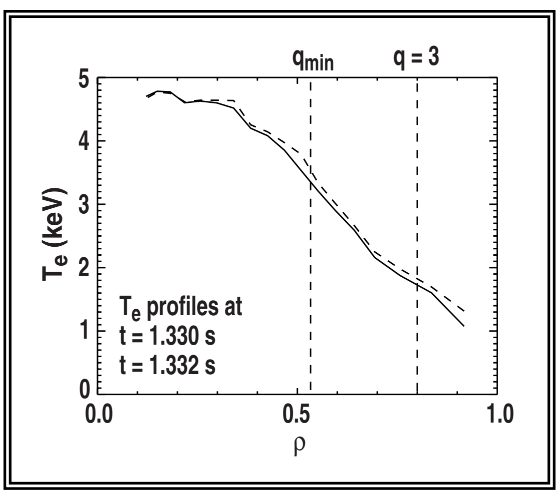

Fig. 2.3.1–2. Electron temperature profiles from electron cyclotron emission before (broken line) and during (solid line) the growth of a resistive wall mode (96519). Magnetic data indicates that the maximum inward displacement at the plasma edge is near the toroidal location of the ECE diagnostic.

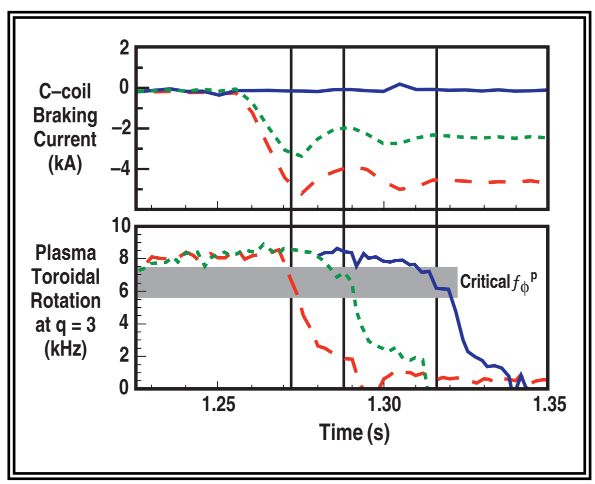

Fig. 2.3.1–3. Three discharges with varying amounts of magnetic braking (96514, 96518, 96515). (a) C–coil current applying an n=1 magnetic perturbation. (b) Plasma rotation frequency at the q=3 surface. The onset time of the resistive wall mode in each discharge is shown by a vertical line. The inferred critical rotation frequency is shown as a horizontal shaded band.

the predictions $\Omega \sim \tau_w^{-1} \leq 3\times10^2$ s^{-1} of theories which include driven islands, and $\Omega \sim \tau_A^{-1} > 10^6$ s^{-1} of ideal MHD theory.The agreement is somewhat better with predictions $\Omega \sim 0.05\,\tau_A^{-1} \sim 10^5$ s^{-1} of theories where the ideal mode is stabilized by dissipation which occurs through coupling to sound waves. The observed critical rotation speed is typically at least 10% of the ion acoustic speed, and thus may be consistent with coupling to sound waves.

We speculate that the much more rapid central rotation of $\Omega \sim 1$–2×10^5 s^{-1} could also contribute to stabilization. Sound wave coupling and dissipation occur at resonant surfaces, and strong shaping and toroidicity couple poloidal modes so that all integer q surfaces are important in this global *n*=1 instability. To date, the discharges which significantly exceed the no-wall beta limit have $q_{min}<2$, placing the q=2 surface in a region of strong rotation (Fig. 2.3.1–1, for example). Discharges with $q_{min}>2$ and hence no q=2 surface tend to have a resistive wall mode onset at lower beta and larger rotation, indicating that rotational stabilization is less effective. The discharges in Fig. 2.3.1–3, for example, have $q_{min}\approx2.3$ and develop an RWM at $\beta_N \sim 2.2$ with a rotation frequency greater than 6 kHz.

The plasma rotation is observed to gradually slow in discharges which exceed the no-wall limit, eventually leading to loss of rotational stabilization as in Fig. 2.3.1–1. Comparison of timing in several discharges shows that this slowing does not correlate with the presence of rotating MHD activity, the H-mode transition, or the onset of ELMs. Possible explanations include electromagnetic drag due to a resistive wall mode saturated at small amplitude or drag due to the continuum resonances of a stable resistive wall mode [3]. Further experimental and theoretical work is needed to determine whether this represents an inherent problem for rotational stabilization.

Active Control. The slow growth and rotation of the resistive wall mode should permit active feedback stabilization by non-axisymmetric coils outside the vacuum vessel, without the need for plasma rotation. Active suppression of resistive wall modes may also help to maintain rotation. Several approaches have been proposed, including the “smart shell” [10,11] where the feedback control is designed to maintain a net zero change in radial magnetic field at the resistive wall, and the “fake rotating shell” [12] in which a phase shift applied to the response mimics the effect of a rotating wall. These schemes will be tested in active control experiments which are planned for DIII–D, initially using the existing error field coil (C–coil). A set of six midplane saddle loops for mode detection have recently been installed, matched in geometry to the six toroidal segments of the C–coil.

A preliminary experiment in open-loop control has been performed, with encouraging results for feedback control experiments. A series of discharges was established having a resistive wall mode at a reproducible onset time and spatial phase. Then the C-coil was programmed to produce a static n=1 magnetic perturbation with a spatial phase opposing the mode, beginning at the anticipated onset time. (The lack of bipolar power supplies required this n=1 perturbation to be superimposed on a constant n=3 bias field; other experiments established that this n=3 field has no detectable effect on plasma stability.) As seen in Fig. 2.3.1–4, in the stabilized discharge the electron temperature, beta, and plasma rotation hesitate at the anticipated onset time, then continue at constant or increasing values. In contrast, these parameters decrease rapidly in the comparison shot without the stabilizing n=1 field. These results suggest that the resistive wall mode was stabilized by the opposing n=1 field. Although complicated by the rapidly changing applied fields, analysis of the saddle loop data indicates that the instability was delayed by at least 20 ms.

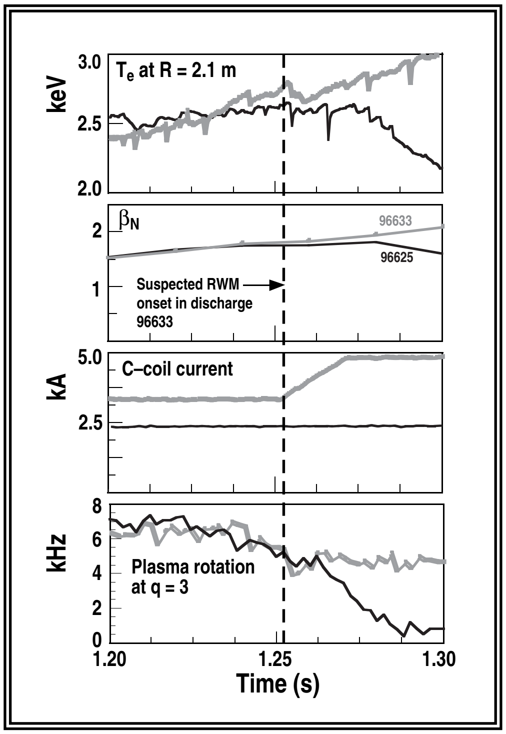

Fig. 2.3.1–4. Comparison of a discharge with a static n=1 perturbation applied to oppose the resistive wall mode (gray, upper curves, discharge 96633) and a discharge without the perturbation (solid, lower curves, discharge 96625). (a) Electron temperature near the q=3 surface. (b) Normalized beta. (c) C-coil current. (The non-zero dc level represents an n=3 bias field.) (d) Plasma rotation frequency at the q=3 surface.

Closed-loop feedback experiments in the near future will be aimed at comparing control algorithms and demonstrating improved stability. New bipolar power supplies to be procured in 1999 and 2000 will increase the power available for feedback stabilization. Numerical modeling with the VALEN 3D electromagnetic code [13] indicates that feedback stabilization using the existing 6-segment C-coil can produce a measurable (~20%) increase in beta over the no-wall limit. Modeling also shows that an extension of the C-coil with additional segments above and below the midplane can double the margin over the no-wall stability limit by allowing better coupling to the helical mode structure. Experimental validation of the models with the existing midplane coil set will provide support for the design of the extended coil set.

Summary. DIII–D experiments have shown that a resistive wall can stabilize a rotating plasma at beta values well above the ideal no-wall limit, for durations much longer than the resistive wall penetration time for n=1 magnetic fields. The predicted resistive wall mode has been observed as the plasma rotation decreases below a critical value of a few kHz, and the ideal structure of the mode has been confirmed. The critical rotation frequency for stabilization may be consistent with theories which include dissipation by coupling to sound waves to provide stabilization in the absence of islands. Long-duration sustainment of wall-stabilized plasmas has been hindered by a slowing of rotation as beta exceeds the no-wall limit. We conjecture that the slowing may result from drag caused by a small-amplitude resistive wall mode or by continuum resonances of the stabilized resistive wall mode. Modeling predicts that feedback stabilization using non-axisymmetric coils can provide a significant increase over the no-wall beta limit. In a preliminary open-loop experiment, the onset of the resistive wall mode was postponed for several wall penetration times, an encouraging result for closed-loop feedback experiments.

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2.3.2. EFFECTS OF PLASMA SHAPE AND PROFILES ON EDGE STABILITY IN DIII–D[8]

One of the major goals of advanced tokamak research is to develop plasma configurations with good confinement and improved stability at high β. In DIII–D, high performance discharges with a high confinement edge and various plasma shapes and current profiles have been produced. All these discharges exhibit enhanced confinement in the plasma edge region, leading to a large edge pressure gradient P′ and bootstrap current density J_{BS}. These edge conditions typically drive edge instabilities which terminate the discharge high performance phase, often accompanied by a permanent loss of the discharge internal transport barrier. An improved understanding of these edge instabilities is essential to optimize and sustain the discharge performance. Furthermore, the performance of future tokamak devices such as ITER is sensitive to the magnitude of the edge pressure pedestal, which is limited by edge instabilities. An improved understanding of the edge instabilities will also provide a more accurate prediction of the performance of future tokamak devices.

Edge instabilities often appear as cycles of edge localized mode (ELM) [1,2] with varying amplitude and frequency depending on the edge conditions, the power loss from the core, and the plasma shape. The effect of ELMs on the discharge performance varies with the ELM amplitude and frequency. An ELM of large amplitude can substantially degrade the plasma performance and result in large energy flux to the divertor. One of the major issues facing advanced tokamak research is the control of edge P′ and J_{BS} which drive these instabilities. In this paper, the results of recent experimental and theoretical studies concerning the effects of plasma shape and current and pressure profiles on edge instabilities in DIII–D are presented. Here, we explore the use of plasma shape as a means to control the edge P′ and J_{BS}, as well as a means to improve our understanding of these instabilities. Since these instabilities are sensitive to details of edge P′ and J_{BS}, most of the studies make use of the recently upgraded 35-channel Motional Stark Effect (MSE) current profile diagnostic [3] and recent improvements to our equilibrium and stability analysis tools to allow a more definite comparison with theory.

In DIII–D discharges with moderate squareness, prior to the first giant type I ELM, magnetic oscillations with toroidal mode number $n \approx 2$–9 and a fast growth time $\gamma^{-1} = 20$–150 μs are often observed. Ideal stability calculations using simulated and experimental equilibria are in general consistent with various observed features of these instabilities. High n ballooning stability results show that the edge region of these discharges is in the second ballooning stability regime, and that the edge P′ substantially exceeds the first ballooning stability limit [4]. Low n stability results show that discharges with large edge P′ and current density J are more unstable to $n > 1$ modes [5–7]. These results indicate that edge instabilities may be the outcome of a complex interaction among the high n ballooning modes, the low n kink/ballooning/peeling modes and the edge P′ and J. The results also suggest that the large edge P′ and J may be controlled by reducing the ballooning second stability access in the edge, thereby providing a means to control edge instabilities. Indeed, calculations and experimental results show that ELM amplitude and frequency can be varied by controlling access to the ballooning second stability regime at the edge through variation of the squareness of the discharge shape [8–10]. Motivated by these results from the squareness experiments, recent calculations show that a high order local perturbation of the plasma shape in the outboard bad curvature region can also reduce and eliminate second ballooning stability access in the plasma edge region. Since the perturbation is local, these configurations tend to retain many of the favorable low n stability property of Dee-shaped plasmas.

[8]Lao, L.L., J.R. Ferron, R.L. Miller, et al., "Effects of Plasma Shape and Profiles on Edge Stability in DIII–D," presented at 17th IAEA Fusion Energy Conference, October 19–24, 1998, Yokohama, Japan, to be published in a special issue of Nucl. Fusion; General Atomics Report GA–A22993 (1998).

In Section 2, the general features of edge instabilities are introduced together with a discussion of magnetic fluctuations. The role of edge P′ and current density J on high *n* ballooning and moderate *n* kink/ballooning/ peeling modes is discussed in Section 3. This is followed by a discussion of the effects of plasma shaping on ELMs in Section 4. A discussion and a summary is given in Section 5.

Edge Instabilities and Magnetic Fluctuations. The performance of DIII–D discharges with a high confinement edge is typically limited by edge instabilities. This is illustrated in Fig. 2.3.2–1, where the time evolution of an H–mode discharge is shown. This discharge carries a plasma current I_P of 1.9 MA and has a vacuum toroidal magnetic field B_T of 2.1 T at the vacuum vessel center. To produce a negative central magnetic shear, up to 14.5 MW of neutral beam power P_B is injected into the plasma during the current ramp up phase. The discharge makes a transition into the H–mode phase at 1.58 s, as can be seen by the rapid increase in the edge electron temperature T_e near $\rho \approx 0.9$ shown in Fig. 2.3.2–1(b). The normalized toroidal beta β_N increases to 2.5 and remains at that value as P_B is reduced from 14.5 MW to 12.1 MW. At 1.876 s, a giant type I ELM occurs as shown by the spikes in the divertor D_α radiation and the outboard mid-plane Mirnov signal given in Figs. 2.3.2–1(b) and 2.3.2–1(c), respectively. The giant ELM causes a rapid drop of the edge T_e and a decrease in the global β_N. As subsequent ELMs occur , the edge T_e and β_N continue to decrease. The magnetic oscillation which initiates the giant type I ELM has a toroidal mode number $n \approx 5$ and a fast growth time $\gamma^{-1} \approx 150$ μs.

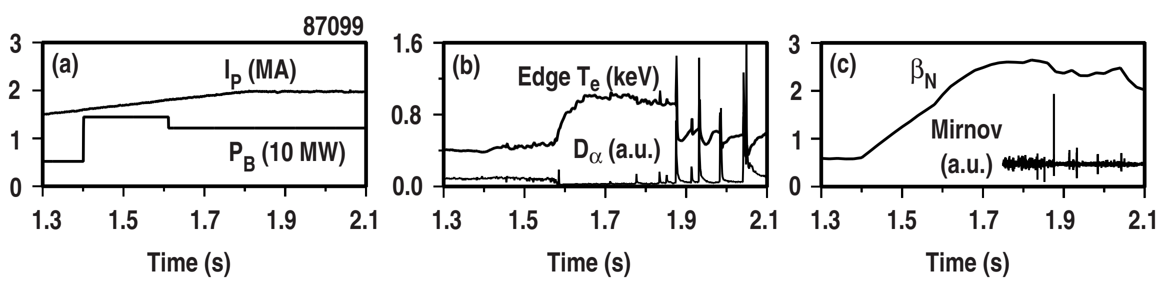

Fig. 2.3.2–1. Time evolution of DIII–D H–mode discharge 87099. (a) Plasma current and injected neutral beam power; (b) edge electron temperature near $\rho \approx 0.9$ and divertor D_α radiation; (c) normalized toroidal beta and Mirnov oscillation in the outboard midplane region.

These type I ELMs and edge instabilities have been observed in DIII–D H– and VH–mode discharges with various poloidal cross sections including single- and double-null divertors, Dee and crescent shapes. Prior to the first giant type I ELM, magnetic oscillations with toroidal mode number $n \approx$ 2-9 are often observed. This is illustrated in Fig. 2.3.2–2. All these discharges have broad pressure profiles and moderate amount of squareness in the plasma shape. The magnetic precursors are localized poloidally in the bad curvature region as well as toroidally with a fast growth time γ^{-1}= 20–150 μs [5,11]. They usually rotate in the electron diamagnetic drift direction, which is consistent with a location near the plasma edge where the $\boldsymbol{E} \times \boldsymbol{B}$ drift is dominated by the diamagnetic drift associated with the large edge P′. They have been observed in discharges with various current profiles and over a wide range of β_N = 2.0–5.0. The attainable beta values decrease with the fraction of plasma current contained in the edge region [7] and are consistent

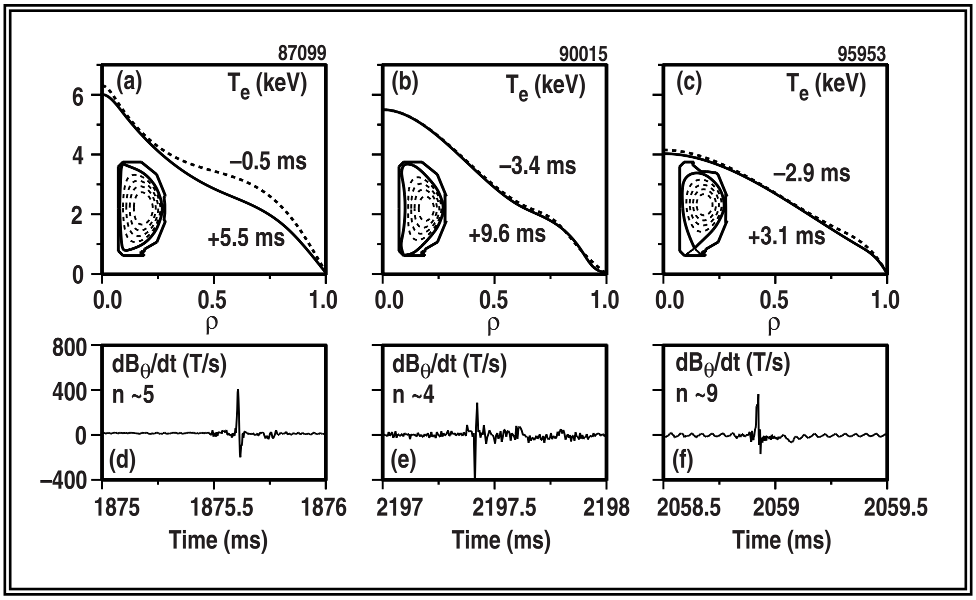

Fig. 2.3.2–2. Magnetic precursors and radial electron temperature profiles before (dashed) and after (solid) an edge instability for (a) a double-null divertor discharge, (b) a crescent-shaped discharge, and (c) a lower single-null divertor discharge.

with the previously observed operational beta limit of $\beta_N \approx 4\ \ell_i$ [12,13]. As shown in Figs. 2.3.2–2(a–c), the instabilities can have global effects, ranging from a slight decrease of edge T_e with a saturation of β_N, to a drop of T_e across the entire plasma with a decrease in β_N. The transport barriers observed in VH–mode and negative central magnetic shear discharges are usually destroyed. These moderate to low *n* edge instabilities have many features similar to the outer modes observed in the hot-ion H–mode in JET [14]. However, the outer modes observed in JET generally have $n = 1$, whereas in DIII–D modes with n = 2–9 are observed. After this first giant ELM, the discharge usually evolves into a quasi-stationary phase at similar or lower β_N values. The low to moderate n =2–9 magnetic perturbations are rarely observed during this phase. Modes with $n > 9$ are difficult to resolve with the existing DIII–D magnetic probes, which suggests that these later ELMs may be driven by edge instabilities with significantly higher *n*.

Localized reflectometer measurements of density fluctuations at the outboard mid-plane show that the magnetic precursors coincide with, or in some cases are preceded by, bursts of increased density perturbation localized to the plasma edge. In cases where a radial propagation can be discerned, the perturbation initiates in the high pressure gradient edge region, and propagates outward into the scrape-off layer.

Role of P′ and J in High *n* Ballooning and Moderate *n* Modes. High *n* ballooning stability analyses show that prior to the onset of the edge instabilities the discharges often have access to the second ballooning stability regime in the outer edge region and the edge P′ substantially exceeds the first ballooning stability limit. This is illustrated in Fig. 2.3.2–3 for a double-null and a lower single-null divertor discharge. The equilibria used in the stability analysis are fully reconstructed from equilibrium analysis using external magnetic data, MSE data, kinetic profile data, and the EFIT code [15]. Stability to the high *n* ballooning modes is evaluated using the BALOO code [16], which now employs a local equilibrium representation [17]. As shown in Figs. 2.3.2–3(a) and 3(b), the double-null divertor discharge, which has a n ≈ 5

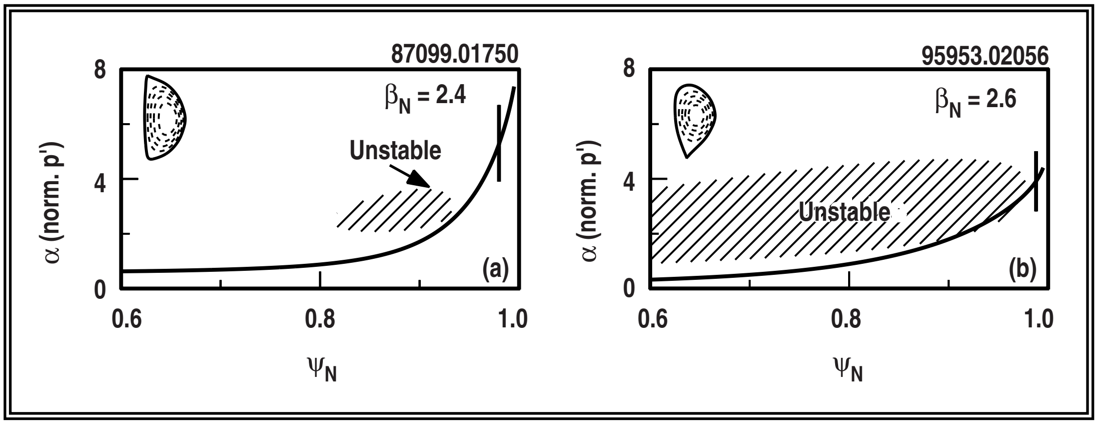

Fig. 2.3.2–3. Ideal ballooning stability of two DIII–D H–mode discharges 87099 at 1750 ms and 95953 at 2056 ms showing the edge regions have second ballooning stability access.

magnetic precursor prior to the first giant ELM [Fig. 2.3.2–2(a)], has a much larger edge P′ and a much wider edge second ballooning regime access than the lower single-null divertor discharge which has a $n \approx 9$ magnetic precursor [Fig. 2.3.2–2(c)]. Access to the second ballooning stability regime in the outer edge is necessary to allow buildup of a large edge P′ often observed in the DIII–D H– and VH–mode discharges. The radial extent of the region with second ballooning stability access depends on the plasma shape and the edge J. An increase in the edge J_{BS} due to an increase in the edge P′ will lead to a further opening of the second ballooning stability zone, in turn allowing a further increase in the edge P′ and J_{BS}. Similar results on ballooning stability have also been reported in Ref. 18. The effects of the plasma shaping on the ballooning stability will be discussed in the next section.

Results of low n stability analyses using simulated and experimental equilibria are consistent with various observed features of the experiments and show that equilibria with broad pressure profiles and large edge P′ and J are more unstable to modes with $n > 1$. This is illustrated in Fig. 2.3.2–4 for a

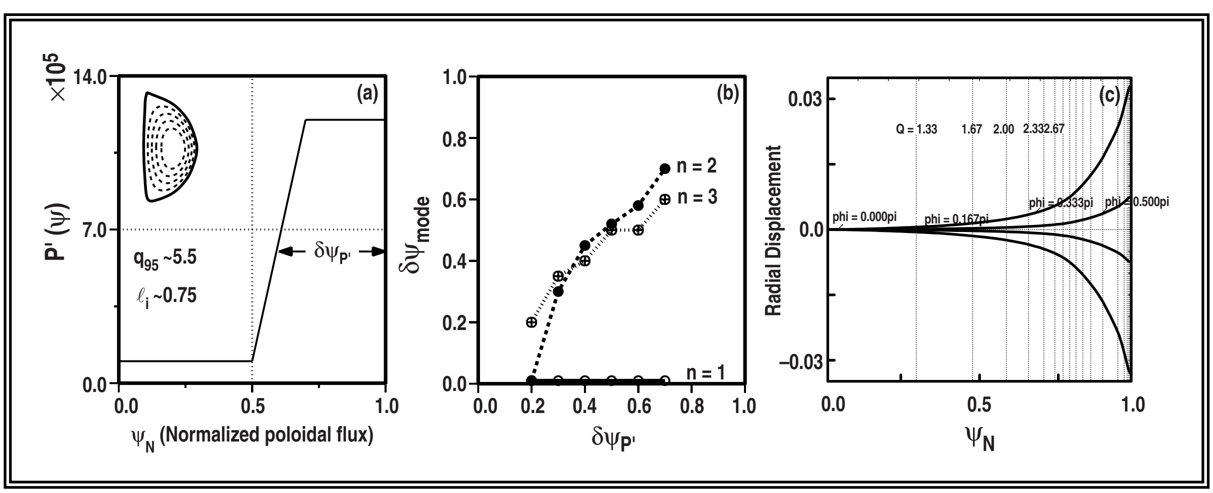

Fig. 2.3.2–4. (a) Equilibrium flux surface and radial profile of pressure gradient used in the low n stability analysis, (b) variation of external radial width of unstable modes with the width of the large edge P′ region, (c) radial structure of a n=3 unstable mode.

sequence of simulated equilibria based on the experimental discharge shown in Fig. 2.3.2–1 but with a simpler pressure gradient profile that has the shape of a step function [Fig. 2.3.2–4(a)]. The stability to the ideal n = 1–3 modes is evaluated using the GATO code [19] with a conducting wall at the surrounding vacuum vessel. As the radial width of the large edge P′ region $\delta\Psi_{P'}$ is increased, the n = 3 modes become unstable first. These unstable modes are kink/ballooning modes with a large peeling component. With a further increase in $\delta\Psi_{P'}$, the n = 2 modes then become unstable. As shown in Fig. 2.3.2–4(c), the radial structure of the unstable modes exhibits a large peeling component in the edge. As expected, the external radial width of the unstable modes $\delta\Psi_{mode}$ increases with $\delta\Psi_{P'}$. The n = 1 modes are stable in all cases. These low n =2,3 modes are driven by both the edge P′ and J [5,6].

Effects of Plasma Shaping on ELMs. The effects of plasma shaping on edge instabilities and ELMs are considered in this section. Theoretical calculations suggest that second ballooning stability access in the outer edge region is reduced at low and high squareness [8]. This is illustrated in Fig. 2.3.2–5, where the edge current density (normalized to the collisionless bootstrap current at the edge) required to gain second stability access in the plasma edge as a function of the squareness of the plasma shape is shown. With $C_{boot} = (J/J_{BS})|_{edge} = 1$ the current density needed to gain access is the bootstrap current density based on the collisionless model [20], with $C_{boot} = 0$ no current density is required. As shown in Fig. 2.3.2–5, at low and high squareness C_{boot} becomes greater than 1 suggesting that second ballooning stability access is more difficult at these squareness values. The calculations are done using self-consistent pressure and bootstrap current density profiles. This reduction of second ballooning stability access is due to the increased weighting of the magnetic field lines in the outboard bad curvature region at high or low squareness.

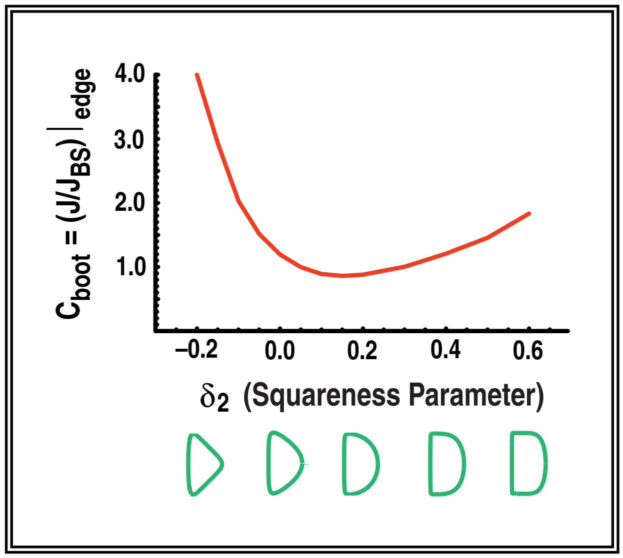

Fig. 2.3.2–5. Collisionless bootstrap current multiplier C_{boot} = (J/J_{BS})/edge required to gain second stability access in the plasma edge region varies with the squareness of the plasma shape, $R(\theta) = R_0 + a\cos[\theta + \sin^{-1}(\delta\sin\theta)]$, $Z(\theta) = \kappa\, a \sin[\theta + \delta_2 \sin 2\theta]$.

Consistent with the results from the theoretical calculations, experimental results show that ELM amplitude and frequency can be varied by controlling access to the second ballooning stability regime at the edge through variation of the squareness of the discharge shape [9]. This is illustrated in Figs. 2.3.2–6 and 2.3.2–7 where the ELM frequency and amplitude as indicated by the divertor D_α radiation frequency and the change in edge T_e at various values of the squareness of the plasma shape are compared. As shown in Fig. 2.3.2–6, the plasma shape is rectangle-like at high squareness and triangle-like at low squareness. At low and high squareness the ELM frequency is strongly increased. The effects of the ELMs on the edge T_e are compared in Fig. 2.3.2–7. At high squareness, the ELM amplitudes are strongly reduced and there is no detectable change in the edge T_e. The ELM behavior at low squareness (not shown) is similar. The ballooning stability boundary of the high squareness discharge is also compared to

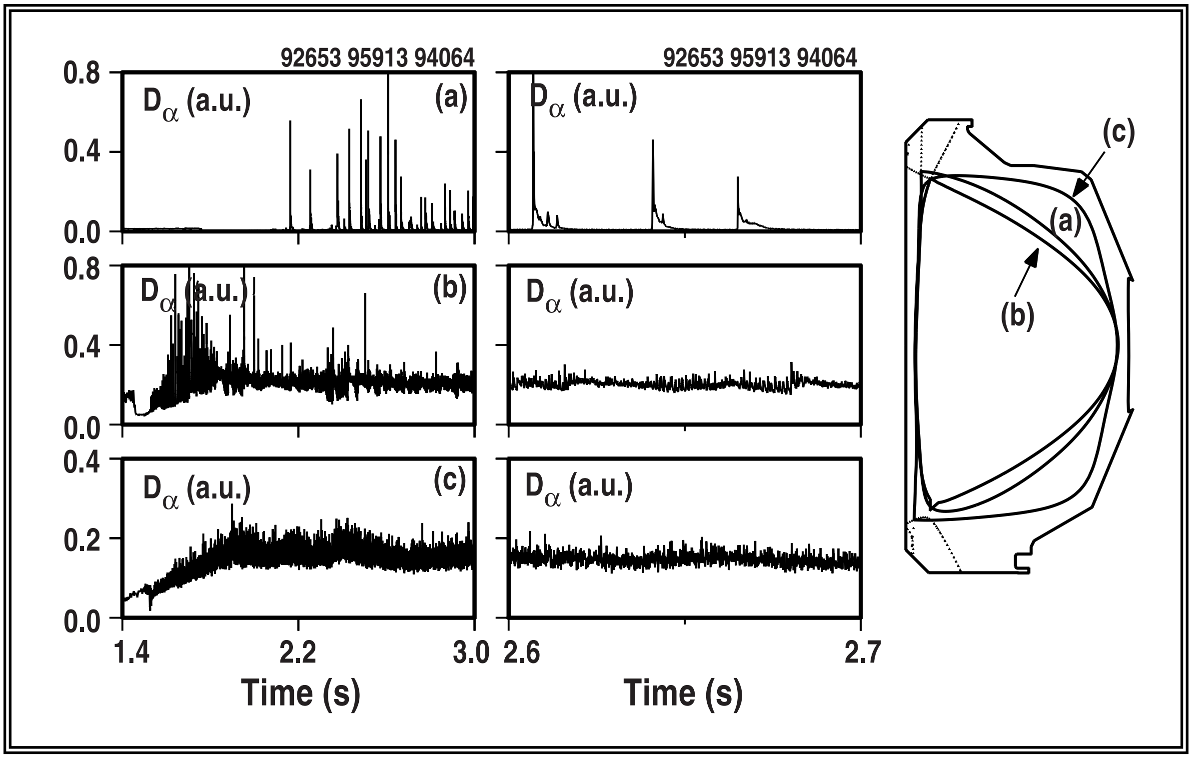

Fig. 2.3.2–6. Comparison of divertor D_α signals for three DIII–D discharges with (a) moderate squareness, (b) low squareness, and (c) high squareness.

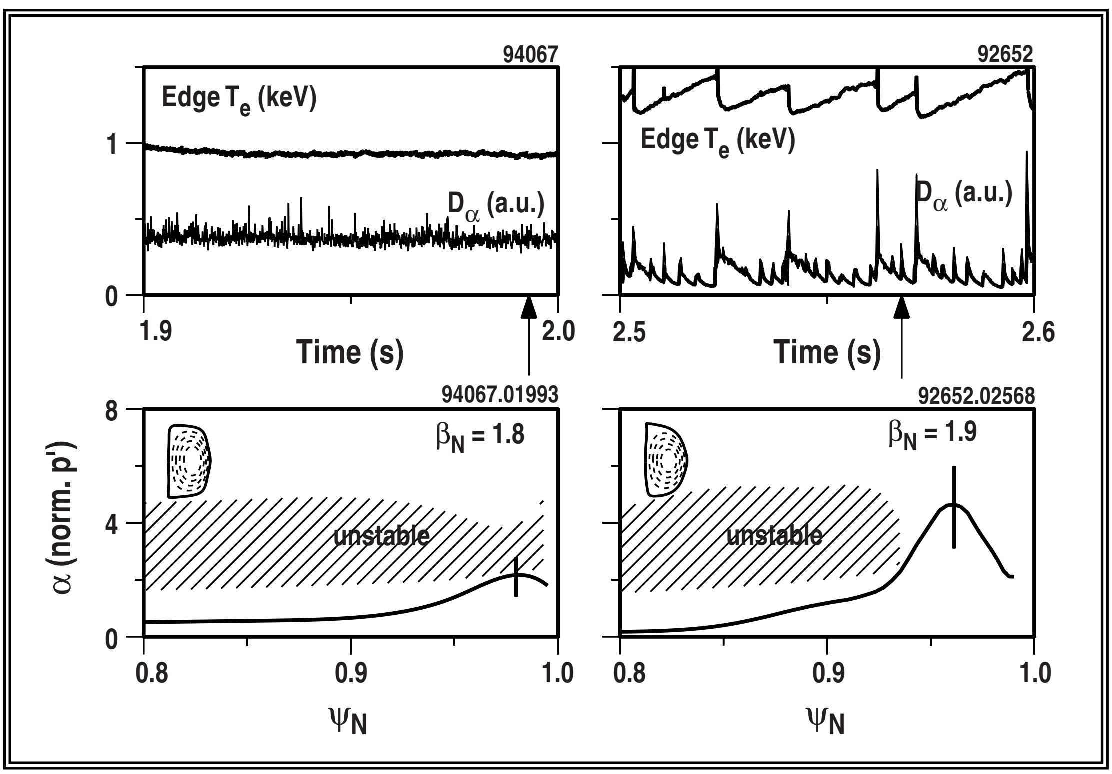

Fig. 2.3.2–7. Comparison of ideal ballooning stability, divertor D_α signals, and edge electron temperatures for a high squareness and a moderate squareness discharge. Arrows show time of stability calculations.

that of the moderate squareness discharge in Fig. 2.3.2–7. At high squareness, access to the ballooning second stability regime is eliminated.

The results from the squareness experiments show that edge instabilities can be controlled by limiting the edge P′ and J through elimination of the second ballooning stability access in the edge region. Motivated by these results, calculations have been carried out to evaluate the effects of a local perturbation of the plasma shape on the outboard bad curvature region on the ballooning stability. The results show that such a high order local perturbation of the plasma shape can also reduce and eliminate second ballooning stability access in the edge region. This is illustrated in Fig. 2.3.2–8. As the perturbation in the outboard region is increased, the second ballooning stability access is reduced and then eliminated. Since the perturbation is local, many of the favorable low *n* stability properties of Dee-shaped plasmas are also retained. Thus, localized shape perturbations may provide a means of obtaining plasmas with good overall β stability but with small benign ELMs. New experiments are being proposed in DIII–D to test the idea.

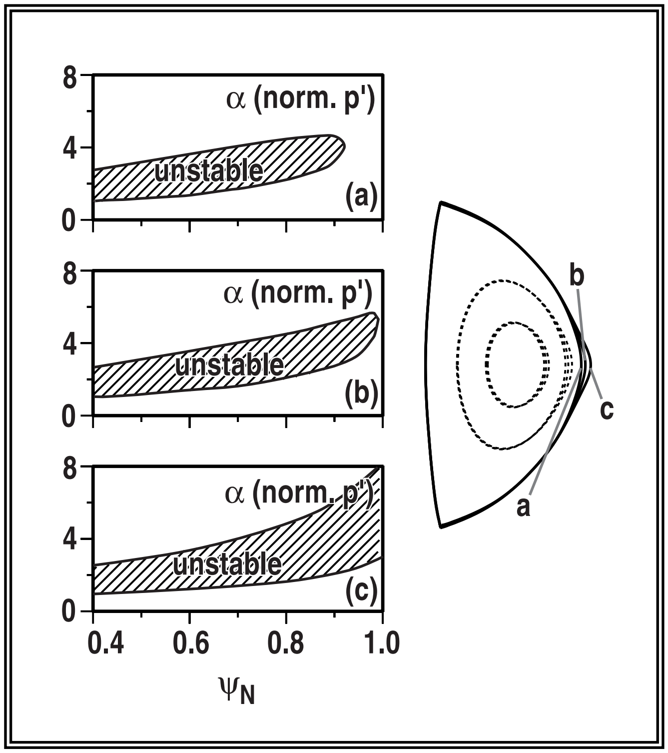

Fig. 2.3.2–8. Comparison of the ideal ballooning stability boundary for three simulated equilibria with (a) no perturbation of the plasma shape in the outboard bad curvature region, (b) moderate perturbation, and (c) large perturbation.

Discussion and Summary. As shown in the previous sections, the performance of DIII–D H– and VH–mode discharges are limited by edge instabilities driven by the large edge P′ and J. Low to moderate n =2–9 magnetic precursors are often observed prior to the first giant type I ELM. Ideal stability analyses suggest that discharges with large edge P′ and J are more unstable to n > 1 modes as observed experimentally and that second ballooning stability access enhances the instabilities by facilitating the development of large edge P′ and J. The observed edge instabilities cannot be explained by a simple picture of instability to the high *n* ideal ballooning modes. Rather, the experimental and theoretical results suggest that they may be the outcome of a complex interaction among the high *n* ballooning modes, the intermediate to low *n* kink/ballooning/peeling modes, the edge P′, and the edge J. The results from the squareness experiments show that plasma shaping can provide an useful means to control edge instabilities.

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2.3.3. DISRUPTION MITIGATION STUDIES IN DIII–D[9]

During a tokamak disruption, the rapid and complete loss of the thermal and magnetic energy result in high thermal and electromagnetic loads on the vessel and internal components and sometimes generate intense high energy runaway electron beams [1–4]. Critical to the tokamak concept along with the operation of future devices, is the development of techniques to terminate the discharge safely and mitigate the destructive effects of disruptions. A disruption in the International Thermonuclear Experimental Reactor (ITER) [5] could result in the rapid loss of 1 GJ of energy leading to damage and reduced lifetime of the first wall of the vessel [5]. Research on the DIII–D [6] tokamak and elsewhere has shown that some of these disruption phenomena are highly nonaxisymmetric, giving rise to local thermal and electromagnetic loads that are much higher than the average [1,7–14]. In a series of dedicated disruption experiments described here, further data have been obtained on the thermal loads to the first wall, the structure and amplitude of the halo currents, and runaway electrons. Major disruptions and vertical displacement events (VDEs) have been investigated and techniques to mitigate the disruptions while minimizing runaway electron production have also been evaluated.

The results presented here extend the work previously reported [7–10,15–16]. The paper is organized as follows: the disruption phenomenology is discussed in Section II, the halo current data and modeling results are reported in Section III, the results of mitigation experiments on DIII–D are reported in Section IV, and the summary and conclusions are presented in Section V.

Disruption Phenomenology. Disruption experiments on DIII–D were performed in deuterium discharges, typically lower single-null divertors with plasma current I_p = 1–1.5 MA, toroidal field B = 1.8–2.1 T, major radius R = 1.7 m, minor radius 0.6 m, and elongation 1.2–1.8. Deuterium neutral beam heating was generally used.

Disruptions, including the experiments performed here, can generally be divided into one of two basic categories: major disruptions and VDEs that lead to a disruption. These two types differ in the sequence of disruption events. In a major disruption, the plasma first becomes unstable due to reaching an operational limit, such as a density limit or beta limit, that leads to the growth of a large magnetohydrodynamic (MHD) mode. The large MHD causes a loss of nested confinement surfaces. The thermal energy is rapidly lost (thermal quench) and the current profile flattens causing a drop in the plasma inductance and a corresponding spike up in the current. Finally, the high resistivity of the cold plasma results in a rapid decay of the plasma current (current quench). Frequently, as a consequence of the change in the current profile, the plasma energy, and the plasma radial position, the vertical position in a major disruption is lost after the thermal quench. In a VDE, the results are similar but the sequence is different. The first event is a loss of the vertical position, and the plasma moves vertically with the cross section and edge safety factor, q, decreasing as the plasma scrapes off against the first wall. The plasma then disrupts: the thermal quench occurs (typically, there is no current spike) followed by the current quench. Most of the experiments reported here were performed using VDEs because of the ability to reproducibly trigger the disruption to occur at a time where the diagnostics are optimized. Disruption effects evaluated include the current decay rate, the halo current, the halo current toroidal asymmetry, the power radiated, and the heat flux to the first wall. The two major effects investigated and mitigated in these experiments are the halo currents and the heat flux.

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Halo currents are currents generated during the disruption that flow along open field lines surrounding the plasma, in what is known as the “halo” region, and return poloidally through the vessel [1,9–10]. Large forces on the vessel components can result when these poloidal halo currents interact with the toroidal field. Note that the toroidal halo currents do not contribute to forces on the vessel since they do not flow in the vessel wall. A number of experiments have also observed significant toroidal asymmetries [7,11–12] in the halo currents that can result in concentrated local forces larger than the average. The halo current measured on DIII–D for a typical triggered VDE [Fig. 2.3.3–1(b)] reaches ~20% of the predisruption plasma current [7]. The halo current is toroidally asymmetric and the degree of asymmetry is quantified in terms of the toroidal peaking factor (TPF) which is defined as the ratio of the peak of the toroidal current distribution to the toroidally averaged value. The time dependence of the TPF is also shown in Fig. 2.3.3–1 for two halo current monitoring arrays in the divertor floor at different major radii. Note that at the time of peak halo current, the TPF is reduced. Results from a number of DIII–D experiments show that at the time of peak halo current, TPFs approaching 3 are seen and halo currents range up to 35% of the initial plasma current.

The second disruption effect of interest is the heat flux. Detailed measurements of the heat flux to the divertor floor in DIII–D and the energy flow across a closed surface surrounding the plasma have been previously reported [8,15–16]. The stored thermal energy is lost during the thermal quench via radiation and conduction while the stored magnetic energy is lost during the current quench largely by radiation. For three disruptions, a VDE, a high beta disruption, and a disruption due to a large injection of argon gas an energy balance can account for the total energy loss during the disruption to within 15%. The energy input to the disruption (Fig. 2.3.3–2) includes the initial stored thermal energy and magnetic energy along with the energy added during the disruption by auxiliary neutral beam heating. The energy out includes the energy lost via radiation and conduction during the disruption and the residual magnetic energy at the final analysis time. Note that the initial energy stored in the magnetic field flows into the plasma during the current quench and is lost as radiation.

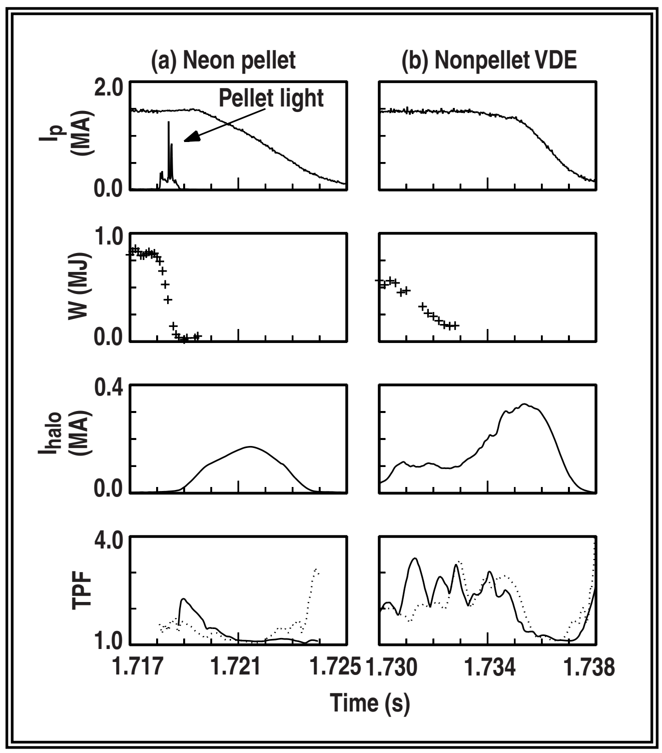

Fig. 2.3.3–1. Comparison of a VDE discharge mitigated with a neon pellet [(a) discharge 88826] versus an unmitigated VDE [(b) discharge 88810]. Time evolution of plasma current, pellet light, stored thermal energy calculated from equilibrium fitting code, the total poloidal halo current and TPF of the halo current measured by divertor floor toroidal arrays at major radius R = 1.08 m, 1.20 m (dotted).

Disruption Modeling. The phenomenology of the axisymmetric component of the halo current, both its origin and evolution, is now well understood. The halo current time evolution in DIII–D disruptions have been accurately modeled with the DINA code [17], a time-dependent, 1.5-dimensional (1.5-D), axisymmetric, resistive MHD and transport plasma simulation code. A simple analytic model, however, can be used to describe and explain the halo currents [18]. This model simulates the plasma and halo interaction via a simple circuit equation:

$$L_h \frac{dI_h^{tor}}{dt} + R_h I_h^{tor} = -M_{hp} \frac{dI_p^{core}}{dt} - \frac{1}{q_h} \frac{d\phi_h}{dt} \quad ,$$

where I_h^{tor} is the toroidal halo current (current on open field lines); I_p^{core} the core plasma current (current contained within the last closed flux surface); ϕ_h the toroidal flux linked by the halo current; L_h, R_h, and q_h are the effective inductance, resistance, and safety factor of the halo region; and M_{hp} the mutual inductance between the core and halo plasmas. The toroidal currents in the halo are induced by the decay of the toroidal core plasma current and by changes in the enclosed toroidal flux (i.e., changes in the plasma geometry). The model shows that the transfer of toroidal current from the core to the halo depends on the characteristics of the halo and core plasmas during the disruption. The toroidal halo current is approximately proportional to the ratio of the core current decay rate to the halo current decay rate which is, in turn, proportional to the ratio of the core-to-halo resistivities (lower temperature halos have lower halo currents). The model also shows that in VDEs, the vertical instability is a factor with the halo current proportional to the ratio of the vertical instability growth rate to the core current decay rate (γ_z/γ_p). While the current decay drives toroidal halo current, a poloidal current is also produced since in the halo region the plasma is force free (∇ p = J × B ~ 0) and the current flows along the open field lines. From the force-free constraint, a simple relation between the poloidal and toroidal halo currents can be derived showing they are proportional and related by the q of the field lines $\left(I_h^{pol} = I_h^{tor}/q\right)$ [18]. An experimental scan of the vertical instability growth rate γ_z was performed by varying the plasma beta and elongation. The poloidal halo current increases with γ_z and the model predictions are in good agreement with both the experimental measurements of the peak halo current (Fig. 2.3.3–3) and with the time history measurement of the halo current in DIII–D VDEs [8,18]. The relation $\left(I_h^{pol} = I_h^{tor}/q\right)$ shows that high poloidal halo currents can arise when the edge q is low. To understand this evolution of the poloidal halo current, we examine VDE discharges in two regimes, $\gamma_z/\gamma_p < 1$ and $\gamma_z/\gamma_p > 1$ (Fig. 2.3.3–4). When γ_z is large [Fig. 2.3.3–4(b)], the cross section decreases faster than the core current decay causing the edge safety factor q (q ~ a^2/I_p, a the plasma minor radius) to decrease and thus the poloidal halo current to be larger. As we shall see, reduction of the poloidal halo currents results

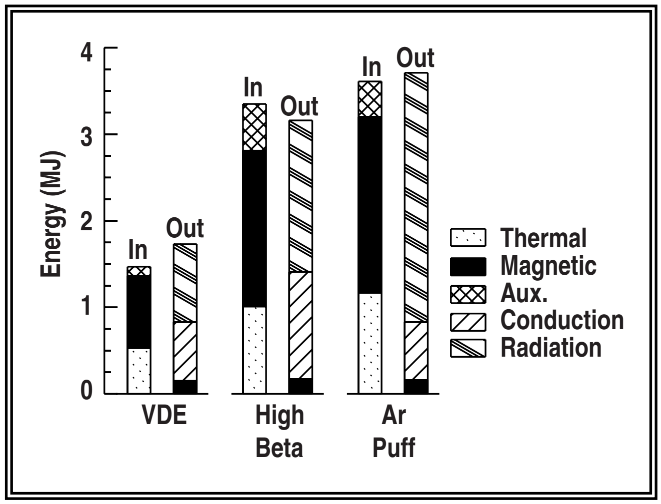

Fig. 2.3.3–2. Energy balance accounting of the energy input to the disruption (IN), which includes initial stored thermal energy, initial magnetic energy, and energy input during the disruption from auxiliary heating, versus the energy lost during the disruption (OUT) which includes radiation and conduction losses and the residual magnetic energy at the final analysis time. Results for three types of disruptions: VDE, high beta, and argon gas puff.

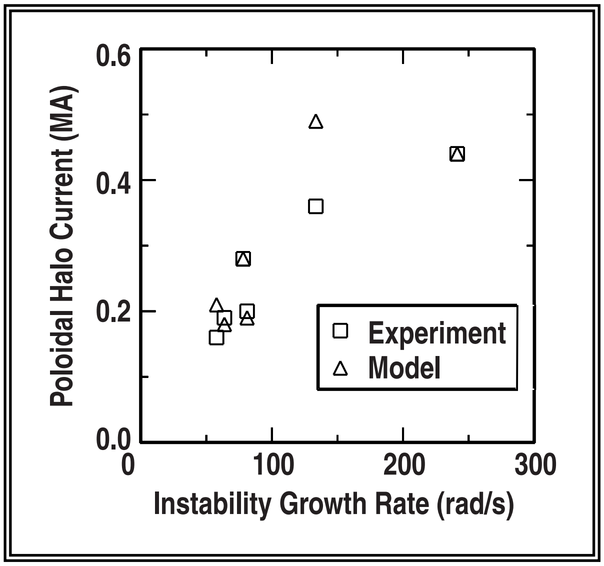

Fig. 2.3.3–3. Halo current versus vertical instability growth rate with fixed current decay rate as experimentally measured and predicted by a model.

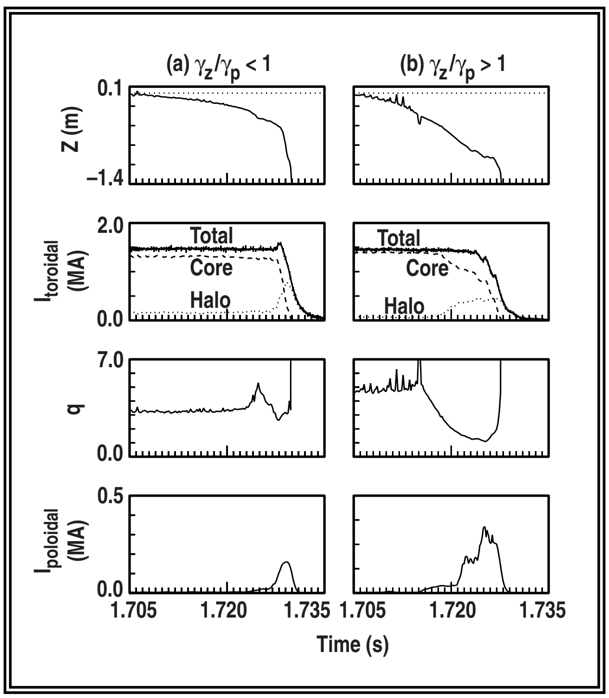

Fig. 2.3.3–4. Comparison of two VDE discharges with a ratio of $\gamma_z/\gamma_p < 1$ [(a) discharge 90204] and $\gamma_z/\gamma_p > 1$ [(b) discharge 90219]. Time evolution of vertical position, total toroidal current, core toroidal current (dashed), halo toroidal current (dotted), edge safety factor q, and total poloidal halo current.

from keeping q large so that the poloidal halo current is reduced.

Disruption Mitigation Results. The deleterious effects of disruptions, i.e., large halo currents and heat fluxes along with large toroidal asymmetries, require methods to reduce these effects. Experiments on DIII–D that have successfully mitigated disruptions include injection of impurity "killer" pellets of neon and argon and injection of a massive gas puff of helium. Solid pellets of neon and argon with sizes of 1.7, 2.8, and 4 mm were injected with a typical velocity of 500 m/s using a pneumatic injector [19] and penetrated to a normalized radius $\rho = r/a$ (r the minor radius) between $0.2 < \rho < 0.5$. The massive gas puff of helium was accomplished using a fast acting valve developed for the DIII–D pellet injector propellant [20]. The valve mounted 0.5 m from the plasma edge somewhat above the midplane connects a 300 ml reservoir filled to 1000 psi with helium directly to the tokamak. A 10-ms wide gas burst is produced with an average flow rate of 4×10^5 T-l/s compared to a typical discharge gas fueling rate of ~70 T-l/s; a total of ~3400 T-l of helium is injected.

The pellet injection phenomenology was described previously along with initial mitigation results [8]. Experiments with neon pellets, argon pellets, or massive helium gas puff reduce the force on the vessel by up to 50% [21]. The mitigation of a VDE by injection of a neon impurity pellet is compared in Fig. 2.3.3–1 to a similar discharge without a pellet. The pellet ablates in ~0.7 ms and most of the stored thermal energy is lost during the pellet ablation time. There is a reduction of the poloidal halo current and TPF. Similar behavior is seen in experiments with argon pellets (Fig. 2.3.3–5). Argon, because of its higher Z, is a better radiator and as expected, this results in a faster cooling with lower halo currents. The MHD for both the pelleted and nonpelleted shot locks and a large n = 1 mode appears ($\delta B/B_T \sim 1\%$).

However, the mode growth with the pellet is much more rapid and occurs early in the pellet ablation. Almost all the stored energy is again lost during the pellet ablation time. The pellet reduces the heat flux to the divertor by ~40% and increases the power radiated. The halo current is reduced but for this disruption case, the halo current asymmetry at the time of peak halo current is already relatively low and there is little reduction in the TPF. The experiments with neon and argon pellets have successfully reduced both the halo currents and TPFs and examination of the time history of the signals shows both have been reduced throughout the entire disruption phase.

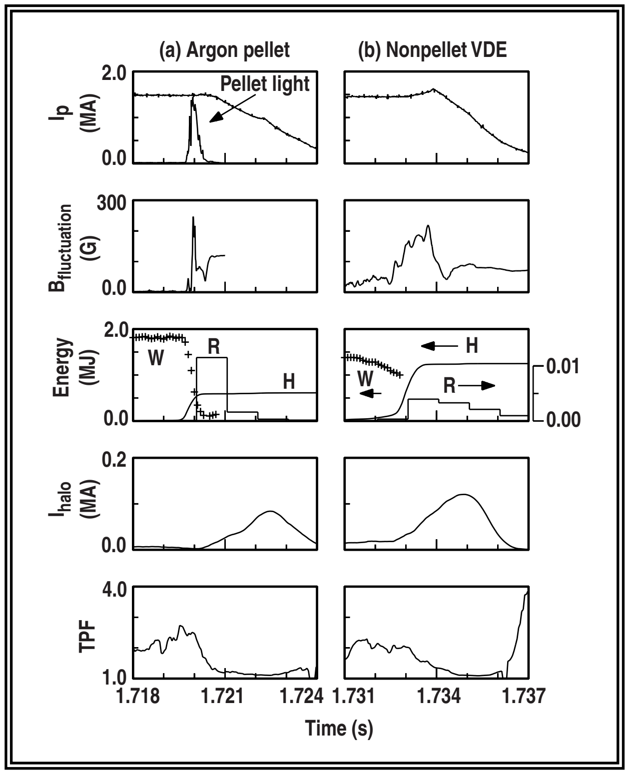

Fig. 2.3.3–5. Comparison of a VDE discharge mitigated with an argon pellet [(a) discharge 90206] versus an unmitigated VDE [(b) discharge 90205]. Time evolution of plasma current, pellet light, poloidal magnetic fluctuation, stored thermal energy (W) calculated from equilibrium fitting code, energy radiated (R) in the time interval represented by each step, the integrated heat to the divertor floor (H), the total poloidal halo current, and the TPF. Note the increased scale for the radiated energy (R) for the unmitigated VDE.

The mitigation of the axisymmetric poloidal halo current by the pellet is explained by applying the model presented previously to the time evolution of the plasma as shown in Fig. 2.3.3–6 (same neon discharges as in Fig. 2.3.3–1). The pellet injection at 1.718 s occurs before the plasma has moved far off axis, when the cross section is still large, and it triggers the thermal quench within 1 ms followed by the current quench. Although the total current decay with the pellet is slower, the core current decay is faster (1.719 to 1.722 s) than the nonpellet VDE (1.730 to 1.735). The larger minor radius a and the smaller core plasma current both combine to keep the edge safety factor q (q ~ a^2/I_p) higher. Thus in the pellet case, although its toroidal halo current is similar to the nonpellet case, the poloidal halo current is reduced .

The rapid and almost complete loss of the stored thermal energy during the pellet ablation occurs even though the pellet is fully ablated before penetrating to the core of the plasma. For the argon pellet injection (Fig. 2.3.3–5), 0. 6 ms after the pellet injection when the ablation of the pellet is complete, the pellet has only penetrated to ρ ~ 0.4 and the stored energy is reduced by 93%. We hypothesize that this energy quench of the plasma inside ρ = 0.4 results from an anomalous rapid transport of the pellet material into the plasma core ahead of the pellet which then causes sufficient impurity radiation to dissipate the plasma's thermal energy [22]. This increased dissipation reduces the heat flux to the divertor. Figure 2.3.3–5 shows a 40% reduction of the integrated heat flux to the divertor.

The results of the impurity pellet injection mitigation experiments have been modeled by a time-dependent 1–D code (KPRAD) that includes pellet ablation and impurity radiation [23]. The code calculates the radiation and energy balance during the pellet ablation on each flux surface assuming no radial

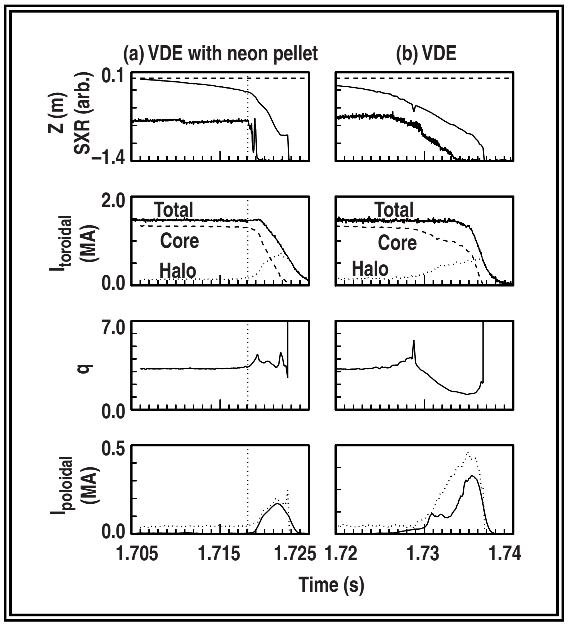

Fig. 2.3.3–6. Comparison of a VDE discharge mitigated with a neon pellet [(a) discharge 88826] versus an unmitigated VDE [(b) discharge 88810]. Time evolution of vertical position, soft x-ray, total toroidal current, core toroidal current (dashed), halo toroidal current (dotted), edge safety factor q, and total poloidal halo current measured (solid) and calculated (dotted). The dotted vertical line in (a) marks the pellet injection time.

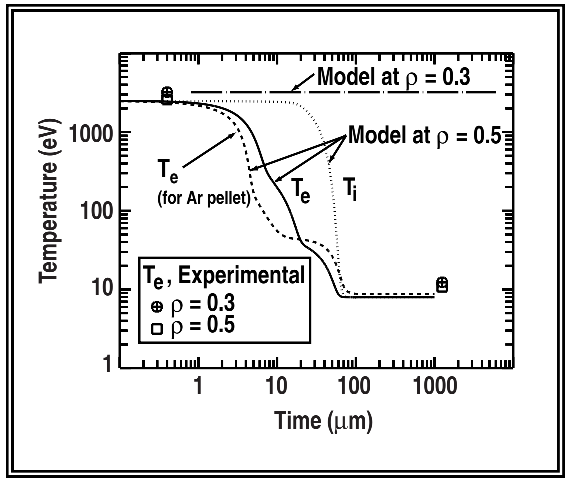

Fig. 2.3.3–7. Time evolution of the electron temperature (solid) and ion temperature (dotted) during a neon pellet ablation as calculated from the KPRAD code at normalized radii of r/a = 0.5 (for discharge 88806) and r/a = 0.3 (dash-dot). Experimental measurements of the electron temperature (from Thomson scattering) before and after the pellet injection at r/a = 0.3 (circle w cross) and r/a = 0.5 (square). Also shown is a KPRAD code calculation of the electron temperature evolution if the pellet material was argon instead of neon (dashed).

conduction. The time evolution of the electron and ion temperature and the electron density are among the output results of the code. Figure 2.3.3–7 shows the calculated temperature for a neon pellet discharge. Both the ion and electron temperature decrease in less than 0.1 ms, at $\rho = 0.5$. This very rapid cooling is due to the large amount of deposited neon impurity ions (~half the initial electron density). They radiate sufficiently in the lower charge states ($Z_{neon} < +8$) that the electron temperature drops fast enough to halt further ionization of the neon to the less radiative higher charge states [23]. A simulation of the cooling due to this neon pellet with the TSC code, [24] a 2–D time-dependent axisymmetric MHD code, agreed with the KPRAD result when TSC assumed no change in the thermal diffusivity and no radial redistribution of the ablated pellet material. The experimentally measured electron temperatures at $\rho = 0.5$ agree with the KPRAD code prediction, but the code does not predict the measured collapse of the central temperature ($\rho < 0.4$) (Fig. 2.3.3–7).

However, the central temperature collapse can be explained by anomalous penetration of the pellet impurity material into the core. Evidence of this anomalous transport includes an increase in the core density within 1 ms of the pellet injection and spectroscopic measurements of pellet material impurity radiation originating in the core. Both the level of the magnetic fluctuations ($\delta B/B_T \sim 1\%$ in Fig. 2.3.3–5) and their early appearance in the pellet ablation indicate that MHD may be the cause of this anomalous radial transport [22].

Although the impurity pellet successfully mitigates the halo current, TPF, and heat flux it frequently has the undesirable effect of generating runaway electrons. Runaway electrons are evident in bursts or continuous hard x-rays, nonthermal electron cyclotron emission (ECE), and a plateau in the current decay due to current carried by the runaway electrons (Fig. 2.3.3–8). Signatures from runaway electrons have been observed on all the argon pellet discharges, many of the neon pellet discharges, but none of the nonpellet comparison discharges. The production of runaway electrons cannot be explained by the classical Dreicer runaway mechanism where electrons with energies above a critical energy runaway due to lack of collisions [25–27]. Calculations using the KPRAD code show that the ratio of the critical energy (E_{crit}) for runaway generation to the thermal energy (T_e) of the bulk plasma drops from ~2000 to ~100 during the pellet injection. This results in an insignificant runaway current («1 A/m^2). The failure of the classical Dreicer runaway electron generation mechanism was also confirmed in a DINA simulation of a discharge with a clear runaway current plateau (Fig. 2.3.3–8). No runaway current was generated in the DINA simulation which included a runaway current model of "Dreicer" acceleration along with a collisional avalanche mechanism [28–29] term to amplify any Dreicer runaway seed current.

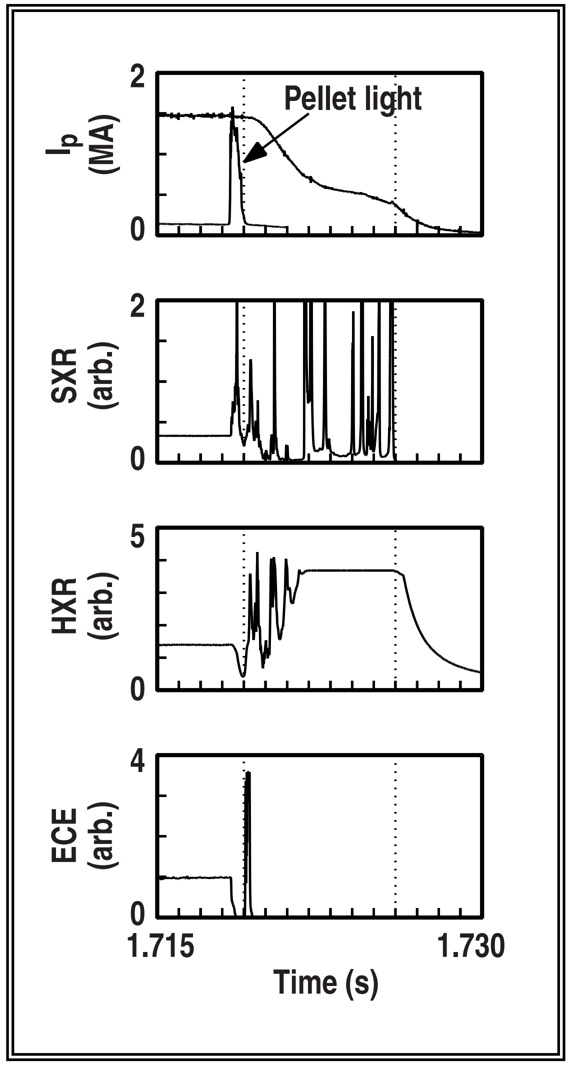

Fig. 2.3.3–8. Time evolution of plasma current, argon pellet ablation light, central soft x-ray radiation, hard x-ray scintillator outside vessel, and RF ECE emission (suffers from density cutoff later in time) for discharge 95180. Evidence of runaway electrons seen after pellet at 1.719 to 1.726 s (times marked by vertical dotted lines).

Two modifications of the standard Dreicer process can provide the observed runaways. Large temperature and pressure gradients across the pellet ablation region can lead to instabilities and, as part of that mixing process, hot electrons from the core can be dumped into the cold, thermally collapsed plasma. These electrons will have a ratio of $E_{crit}/T_e \sim 1$ and will run away. The second modified runaway mechanism is due to the rapid cooling caused by the pellet. KPRAD calculations of the cooling rates (for the discharge in Fig. 2.3.3–7), as a function of the electron energy, indicate that the cooling time of ~0.03 ms for the bulk electrons is too rapid for the electrons in the tail of the energy distribution function to participate due to the finite collisional coupling times. Test particle calculations with KPRAD show that electrons with energies of 12 times the initial thermal temperature of the bulk plasma or larger will runaway to relativistic energies while electrons with energies of 11 times or less will be cooled [22–23]. A Fokker-Planck code, CQL3D, [30–31 simulation of the discharge in Fig. 2.3.3–7 has also verified this runaway generation mechanism. For the neon pellet, the calculated runaway current density is ~1.1 × 10^5 A/m^2 (compared to the initial current density of ~10^6 A/m^2). This runaway production mechanism is very sensitive to the cooling time of the bulk electrons. Using a KPRAD simulation for argon instead of neon results in a more rapid cooling

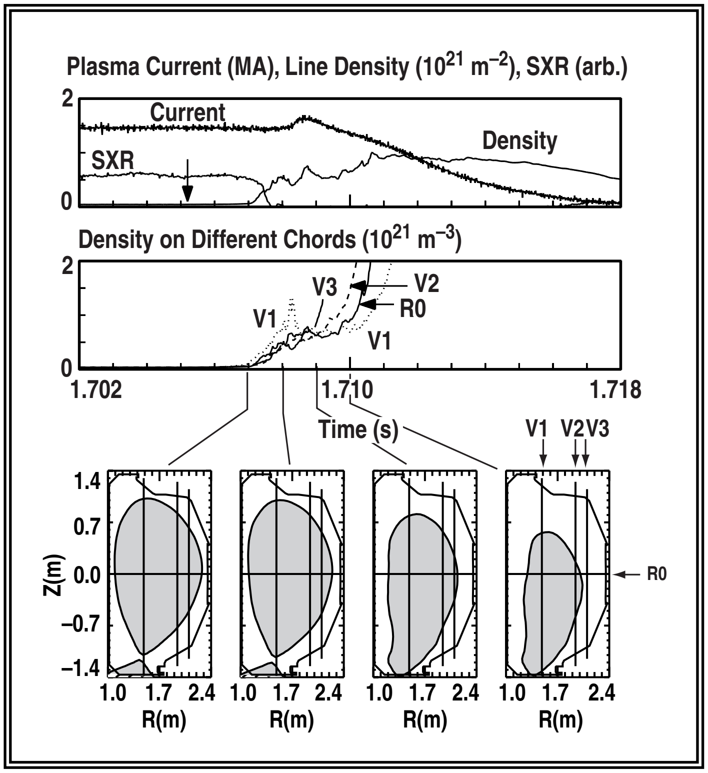

Fig. 2.3.3–9. Massive helium gas puff phenomenology. Plasma current, line density, and average density calculated from the line density along three vertical chords V1, V2, V3 at major radii R = 1.486, 1.942, 2.099 m and 1 horizontal chord at z = 0 m for discharge 96764. Flux plots for four times 1.707 to 1.710 s are also shown with the density chord positions marked. The time the puff valve opens is marked by the arrow.

(Fig. 2.3.3–7) and thus a runaway current density nine times higher (~9×10^5 A/m^2). This increase in runways with the faster cooling argon impurity pellets explains the experimental observation stated above that runaways are observed in all the argon pellet experiments but only some of the neon experiments. These runaway generation mechanisms provide an initial seed of runaway electrons that can then be amplified by the collisional avalanche mechanism [28–29] to produce the observed runaway current levels (~0.3 MA in Fig. 2.3.3–8) [32].

Results similar to the impurity pellet mitigation, but without the generation of runaway electrons, are obtained in a set of preliminary experiments where a massive puff of helium gas is injected. The phenomenology of the massive gas puff is shown in Fig. 2.3.3–9. In this discharge, there is a large density increase, a thermal quench, and then the beginning of the current quench following ~2, 2.2, and 3 ms, respectively, after the opening of the fast puff valve. The increase in the average density to ~1×10^{21} m^{-3} measured along two vertical chords and one horizontal chord initially track (until 1.709 s) indicating that the gas has rapidly penetrated to the center and that the density is uniform across the plasma. One vertical chord (V1), passing through the divertor region, is initially larger than the others indicating a large density in the divertor region until the plasma has limited on the floor after which all the chords give the same density (1.7085 to 1.709 s). Only 10% to 20% of the ~10^{23} helium atoms injected are ionized. Later in time, the average density along the chords, which is calculated assuming all the density is in the plasma core, start to diverge due to significant density in the halo region.

The massive helium gas puff mitigates a VDE (Fig. 2.3.3–10) with a reduction of both the halo current and TPF but shows no evidence of runaway electrons [soft x-ray signal (Fig. 2.3.3–9)]. The heat flux in these ohmic discharges is also significantly reduced by the helium puff but has to be inferred since the direct measurement by the IR camera during the helium puff is precluded by line radiation from the helium in the same region of IR used by the camera. The helium puff results in the radiated power increasing by 45% and the fraction coming from the core plasma as opposed to the divertor region increases from 0.64 to 0.82 (Fig. 2.3.3–11). The total energy radiated in the helium puff discharge is equal (within 5%) to the sum of the energy radiated and the energy deposited as heat flux to the divertor in the unmitigated

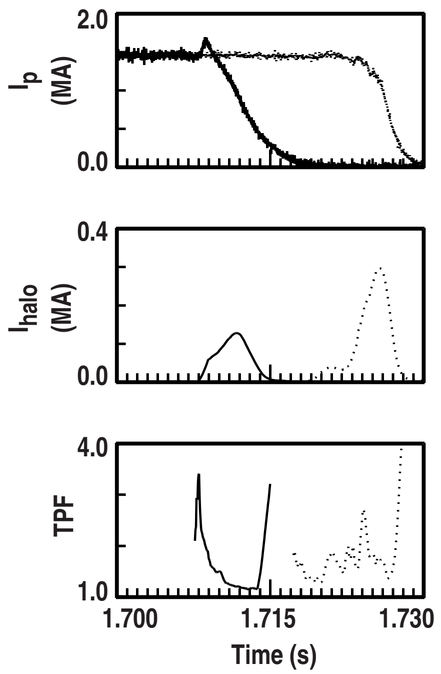

Fig. 2.3.3–10. Comparison of VDE discharge mitigated by a massive helium gas puff (solid, discharge 96764) with an unmitigated VDE (dotted, discharge 96759). Time evolution of plasma current, total poloidal halo current, and TPF of the halo current.

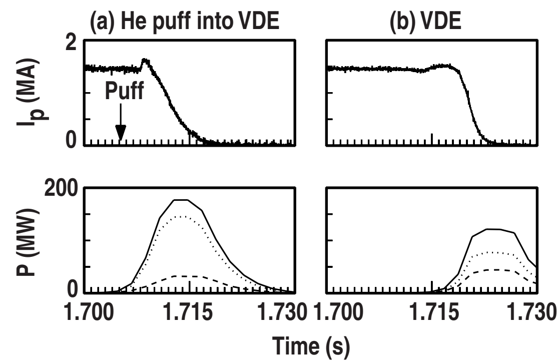

Fig. 2.3.3–11. Comparison of VDE discharge mitigated by a massive helium gas puff [(a) discharge 96762] with an unmitigated VDE [(b) discharge 96757]. Time evolution of the plasma current, total power radiated (solid), core power radiated (dotted), and divertor power radiated (dashed).

VDE (Fig. 2.3.3–12). Assuming there is no energy conducted to the divertor in the helium puff discharge, the total energy accounting shows the radiation and residual magnetic energy balance to within 5% with the initial thermal and magnetic energy (Fig. 2.3.3–12). Thus since essentially all the energy is appearing as radiation, there is none remaining to be deposited as heat flux on the divertor. Several factors combine to leave unresolved whether the thermal energy is dissipated via radiation or via conduction to the floor. These factors include: a thermal energy in these ohmic discharges that is well below the dominant magnetic energy, the slow time resolution of the bolometer, and the lack of a reliable IR measurement of the conduction to the floor.

A KPRAD simulation provides an understanding for the heat flux mitigation. The temperature dilution from the increasing density, along with energy losses from ionization and radiation decrease the electron temperature to ~6 eV in 2 ms. The plasma becomes dominated by volume recombination (recombination time for He^{+2} ~2 ms for these conditions) and by radiation losses from the hydrogenic (He^{+1}) charge state which has a radiative cooling rate of $\sim 3 \times 10^{-35}$ W–m^3 at 6 eV. This results in ~30 MW/m^3 (or 0.6 GW total) radiated power loss density, in good agreement with experimental radiation measurements (maximum measured 35 MW/m^3). The measured UV continuum radiation from the recombination of He^{+2} confirms that the plasma temperature is ~6 eV at the beginning of the current quench in agreement with the modeling. This low electron temperature effectively halts any further ionization of the helium (ionization potential 24.6 eV) and explains why only 10% to 20% of the injected helium is ionized.

Experiments were also performed to see if results similar to the mitigation of disruptions following VDEs occurred for major disruptions. The massive helium gas puff and argon pellet were used to preemptively terminate a discharge near a density limit major disruption (Fig. 2.3.3–13). This figure shows

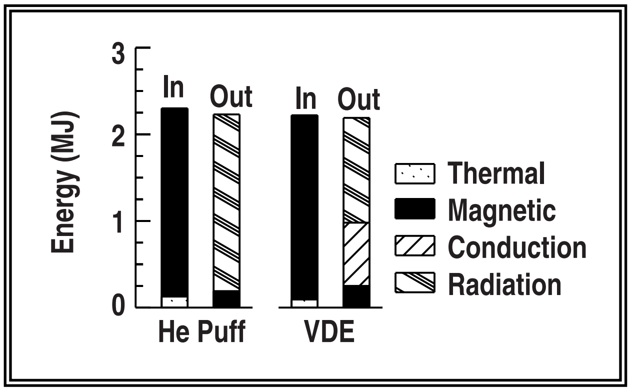

Fig. 2.3.3–12. Energy balance accounting of the energy input to the disruption (IN), which includes initial stored thermal energy and initial magnetic energy (there is no auxiliary heating in this discharge), versus the energy lost during the disruption (OUT) which includes radiation and conduction losses and the residual magnetic energy at the final analysis time. Results for the helium gas puff mitigation and unmitigated VDE discharges in Fig. 2.3.3–11. The conduction for the helium puff is not measured.

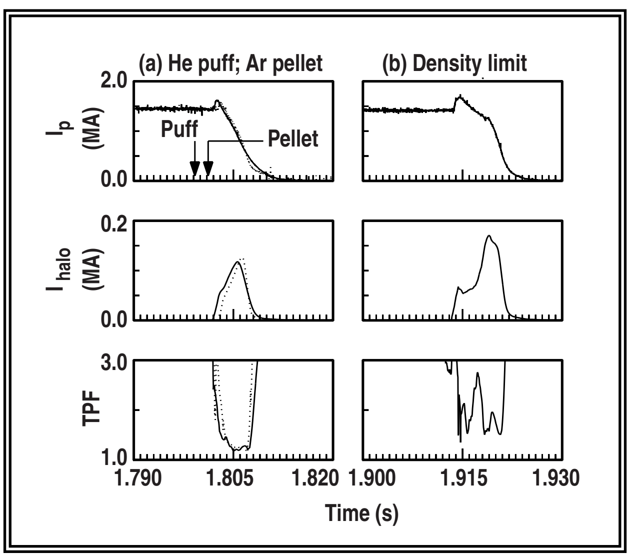

Fig. 2.3.3–13. Comparison of a massive helium gas puff injection [(a) solid, discharge 96767] and an argon pellet injection [(a) dotted, discharge 96768] into repeat of a discharge with a major density limit disruption [(b) discharge 96766]. Time evolution of plasma current, total poloidal halo current, and TPF of the halo current. The injection times of the helium puff and argon pellet are as marked.

that the results of a helium puff and an argon pellet have a very similar mitigation of the halo current and TPF. The force on the vessel was also reduced in the helium puff case, similar to the pellet mitigation results.

The mitigation using the massive helium gas puff injection is encouraging for liquid jet mitigation of disruptions in future large devices such as ITER. In such devices, with large currents (>10 MA) a small initial "seed" of runaway electron current is expected to be amplified by the collisional avalanche mechanism [28–29 to the point that the dominant part of the many mega-ampere plasma current is carried by multi-MeV runaway electrons. To prevent such a generation of runaways, it is necessary to increase the density sufficiently that the increased electric field during the disruption remains below the critical field for the avalanche process [29,33]. In ITER, a 50 to 100 fold density increase would be required and a single pellet or multiple pellets would not be sufficient. Calculations indicate a pulsed liquid jet can meet the mass injection requirement [33]. A fast liquid jet of hydrogen or helium also has a number of advantages for disruption mitigation. During jet injection, there will be isobaric dilution cooling, allowing deep penetration and possibly avoiding MHD driven instabilities due to an unchanged pressure profile. Liquid jets will rapidly cool the plasma (dilution cooling, bremsstrahlung, and recombination radiation cooling) which induces the rapid current decay; thus mitigating the halo current and heat flux load. The high density will also inhibit the formation of runaway electrons.

Summary and Conclusions. We have reported on the progress of understanding the mitigation of both major disruptions and vertical displacement events (VDEs) on the DIII–D tokamak. Halo currents with up to 35% of the predisruption plasma current, toroidal peaking factors approaching 3, and heat fluxes of up to

100% of the predisruption thermal energy have been observed in unmitigated discharges. The halo current origin and scaling is well understood and well predicted by a simple analytical model and simulation codes such as DINA and TSC.

DIII–D experiments achieved a similar level of mitigation of both major disruptions and the disruptions following VDEs by preemptively terminating the discharge by injection of impurity pellets of neon and argon and a massive helium gas puff. Injection of impurity pellets into DIII–D plasmas after the initial loss of vertical position have effectively reduced the halo currents, toroidal asymmetry of the halo current, and the heat flux conducted to the divertor that occurred during the disruption at the end of the VDE. Production of runaway electrons, however, have been observed in many of the pellet injection experiments — particularly those using argon. The rapid cooling of the plasma can be explained by a modeling code (KPRAD) and the dissipation of the stored energy inside of the pellet burnup radius can be explained by the observed anomalous rapid transport of the pellet material into the plasma core. The runaway generation can be understood in terms of a modified Dreicer mechanism occurring during the pellet penetration. The first experiments with the injection of a massive amount of helium gas have been explored as an alternative to the impurity pellets in order to eliminate the undesirable runaway electrons. These experiments have shown effective mitigation of the halo currents, halo current asymmetry, and heat flux and have avoided the generation of runaway electrons. Both the impurity pellets and the massive helium gas puff have also been shown to preemptively mitigate a major disruption. Calculations for liquid jet mitigation of future large machines show the promise of the liquid jet idea.

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2.4. DIVERTORS AND EDGE PHYSICS

2.4.1. RADIATIVE DIVERTOR AND SOL EXPERIMENTS IN OPEN AND BAFFLED DIVERTORS ON DIII–D[10]

We present in this paper progress towards an increased understanding of the relevant physical processes that are important in controlling: (a) the divertor heat flux with divertor and core radiation (Section 2), (b) impurity and deuteron concentrations with plasma flows and cryopumping (Section 3), and (c) erosion and redeposition at material walls in the divertor (Section 4). We have performed new experiments comparing operation with open and baffled divertors, and have demonstrated core density control in both low- and high-triangularity plasmas (Section 5). Core density control is an important tool for DIII–D where active profile control is planned with electron-cyclotron-current drive.

We have used new diagnostic measurements of key physics parameters in concert with state-of-the-art computational models (UEDGE-EIRENE-DEGAS, B2.5-EIRENE) to further our understanding of the fundamental physical processes. Shown in Fig. 2.4.1–1 is the DIII–D diagnostic set in the lower divertor, along with the quantities that are measured by each instrument. Recent improvements include: (a) 2-D measurements of carbon line and deuterium recombination radiation, including CIV (1550 Å), (b) measurement of plasma flows with spectroscopy and a Mach probe, and (c) erosion and redeposition measurements with a DiMES probe.

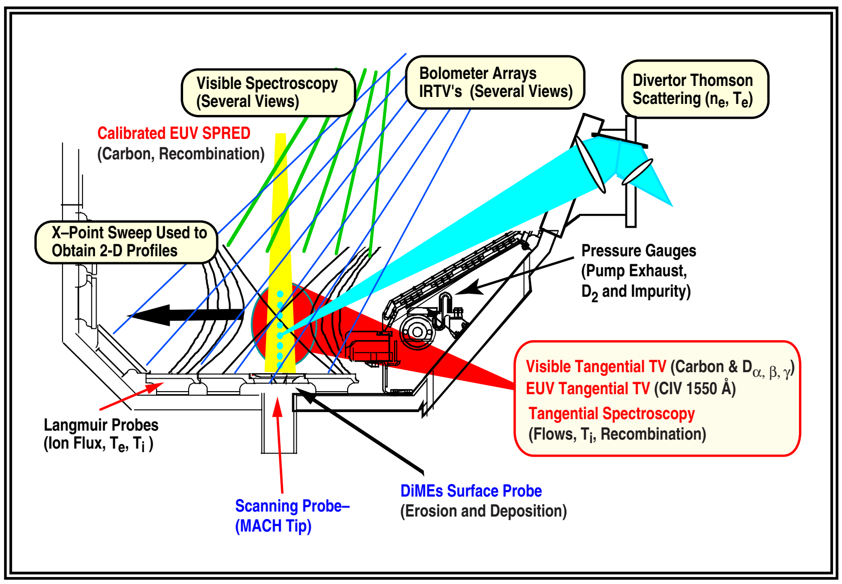

Fig. 2.4.1–1. The array of diagnostics (with quantity measured) in the lower divertor in DIII–D includes: Bolometer (48-channel, radiated power distribution), Scanning mach probe (deuterium flow), Infrared TV's (7-IRTV's-lower divertor heat flux at two toroidal locations, upper divertor, also inside wall), Tangential spectroscopy(neutral, impurity and deuteron flows, T_i, Recombination), Tangential TV (simultaneous inverted 2-D profiles of two lines: CII, CIII, D_α, D_β, D_γ, neutral density), DiMES surface probe (erosion and redeposition on carbon and tungsten samples), Target Langmuir probes (T_e, n_e, Γ_i-ion flux), EUV SPRED spectrograph with absolute intensity calibration(CIV, L_α, L_β, other impurity emissions), an EUV tangential camera (CIV 1550 Å 2-D images), Neutral pressure gauges (deuterium and impurity partial pressures in the edge plasma and pumping plenum), and an 8-channel divertor Thomson scattering system DTS (T_e, n_e the plasma is swept to obtain 2-D profiles).

Physics of Divertor Heat Flux Reduction. Starting from an empirical observation of the reduction of divertor peak heat flux with deuterium puffing [1], we have systematically measured the important terms in the heat and particle transport equations with new diagnostics; at present we have a 2-D characterization of much of the divertor region. Shown in Fig. 2.4.1–2(a) is a representative partially detached divertor (PDD) discharge [2], showing the reduction in peak heat flux,

[10]Allen, S.L., N.H. Brooks, R. Bastasz, et al., "Radiative Divertor and SOL Experiments in Open and Baffled Divertors on DIII–D," presented at 17th IAEA Fusion Energy Conference, October 19–24, 1998, Yokohama, Japan, to be published in a special issue of Nucl. Fusion; General Atomics Report GA–A23004 (1998).

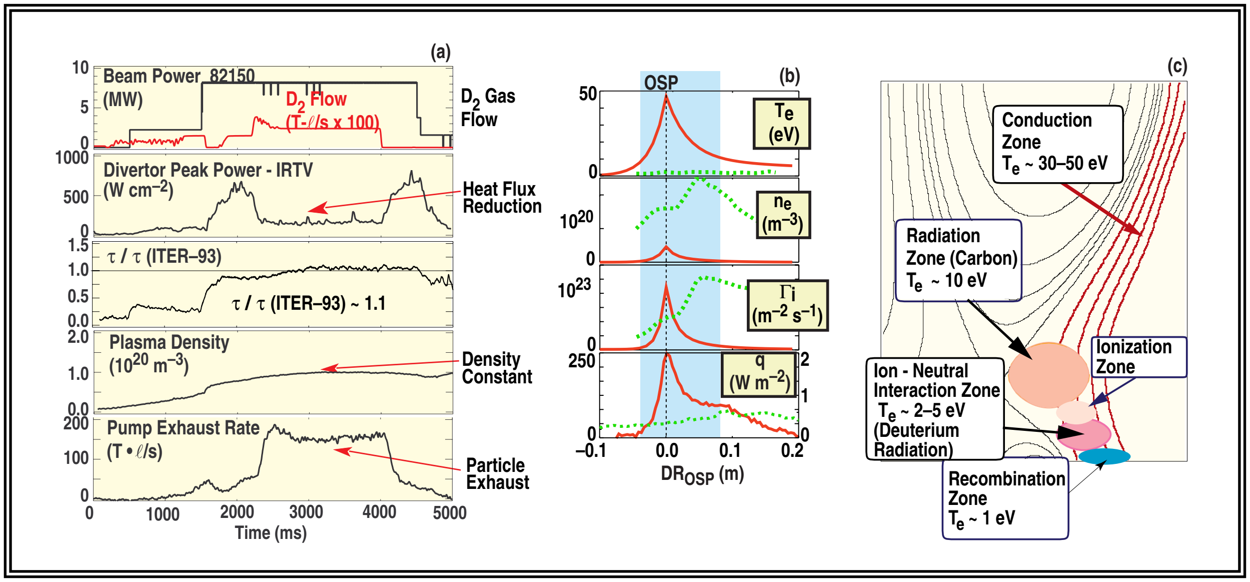

Fig. 2.4.1–2. (a) A representative PDD discharge (D_2 puffing) showing heat flux reduction, H–mode confinement, and controlled core density. (b) The T_e, n_e, Γ_i, and heat flux at the outer strike point for a representative attached and PDD (dashed) discharge. (c) The model that has been developed to explain the physical processes in the PDD discharge.

constant ELMing H–mode confinement, and controlled core density with cryopumping. Scaling experiments have shown that the outer strike point (OSP) peak heat flux can be reduced to nearly the same value, 1 MW m^{-2}, independent of input neutral beam power up to $P_{inj} \sim 14$ MW [3,4]. In Fig. 2.4.1–2(b) are plasma profiles at the OSP for a typical discharge set of attached and detached (dashed) discharges. The q_{div} (heat flux), Γ_i (ion flux), and P_e (electron pressure) are reduced at the OSP, but Γ_i increases outboard, hence the name partially detached divertor. The T_e from Divertor Thomson Scattering (DTS) is dramatically reduced to 1–2 eV, and this has been a key measurement in the interpretation of much of the divertor data.

A 2-D model of the PDD is shown in Fig. 2.4.1–2(c). To facilitate discussion of the changes of the parallel heat flow from an ELMing H–mode to PDD operation, we use a 1-D form of the parallel energy flow equation [5], where the first term is identified as the classical parallel heat conduction, the second term is convection, and $v_{//}$ is the parallel plasma fluid velocity. The losses S_E include radiation, ionization, and recombination:

$$\frac{d}{ds}\left\{\left(-\kappa T_e^{5/2}\frac{dT_e}{ds}\right) + n v_{\parallel}\left[\frac{5}{2}\left(T_i + T_e\right) + \frac{1}{2} m_i v_{\parallel}^2 + I_o\right]\right\} = S_E \quad . \tag{1}$$

In the PDD, conduction carries the exhaust heat along the field line from the core to a region near the divertor. Here, carbon radiates strongly (10–12 eV), as indicated by calibrated EUV spectroscopy of the divertor region. Inversions of 2-D visible (CII, CIII) and ultraviolet (CIV 1550 Å) emissions indicate carbon radiates primarily near the X–point. The total radiation (mostly UV) from spectroscopy agrees well with corresponding chords of the bolometer. Ratios of calibrated multi-chord [6] and TV images in the visible (Balmer$_\alpha$/B$_\beta$) [7] indicate that recombination occurs near the divertor plate in the PDD as indicated in the model in Fig. 2.4.1–2(c) [8]. Corresponding ratios in the UV (Lyman$_\alpha$/Ly$_\beta$) show similar changes, but often suggest absorption of the Ly_{α} emission [9]. A radially-integrated form of Eq. (1) has shown that the measured T_e from DTS, the bolometer radiation, and q_{div} (from IRTVs) are consistent with conduction in ELMing H–mode, but convection is required in the PDD [10,11]. The implied radially-averaged $v_{//}$ is consistent with direct measurements of M ~ 0.8 in the divertor region [12] from the scanning Mach probe

The diagnostic set has enabled several other interesting observations: (a) there is a critical upstream separatrix density for PDD onset that scales roughly linearly with the input power [13]; (b) target Langmuir probe measurements indicate that electron kinetic effects are not important in ELMing H–mode or PDD plasmas [14], (c) the neutral density near the X–point has been measured to be ~10^{17} m^{-3} in L–mode discharges from tangential TV views [15], and (d) the effect X–point geometry effects on boundary potentials and turbulence have been examined [16].

We have also performed "puff and pump" experiments with SOL deuterium puffing (up to 2.5×10^{22} D^0 s^{-1}), exhaust with cryopumping, and impurity injection. In trace injection experiments, exhaust enrichment up to 17 was obtained with argon, and enhanced radiation in the divertor and SOL has also been obtained with argon [17]. For the conditions on DIII–D, we find that the argon and carbon radiation in the divertor increase by equal amounts, but models [18] indicate that this need not be the case in future machines operating in different temperature regimes.

We explored the source of carbon by comparing L–mode helium with L–mode deuterium plasmas, and the dramatic decrease in the carbon radiation suggests that chemical sputtering plays a role in the source. However, the total divertor radiation in these discharges was unchanged because the carbon radiation was replaced by helium radiation. Modeling with the MCI Monte-Carlo code suggests that standard chemical sputtering models overestimate the amount of carbon we measure experimentally [19].

Plasma and Impurity Flows and Comparison with Models. We have studied plasma flows both to further our understanding of the convective heat flow discussed above and to understand impurity transport in the divertor and SOL. Active control of impurities, like enrichment achieved with "puff and pump" techniques discussed above, is an approach to increase the operational window of radiative divertor plasmas and a method to control core impurity content. We have made recent improvements to the UEDGE code, including realistic models of chemical and physical sputtering [20], to model impurity transport for comparison with data. The UEDGE code has been benchmarked with a variety of DIII–D discharges [21], and it calculates multi-species impurity transport. In Fig. 2.4.1–3 are the poloidal distributions of the $C4^+$ ion density (the charge state flowing into the core) and net parallel particle force. Impurity accumulation or depletion is determined by the force gradient

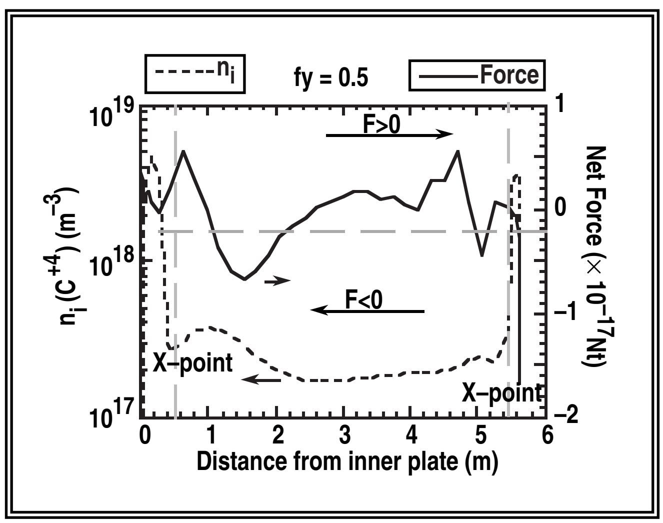

Fig. 2.4.1–3. The ion density and net force on $C4^+$ ions (right) from UEDGE, the gradient in the force near a force null (e.g., near 0.4 and 1 m) determines whether the ion density builds up or is depleted.

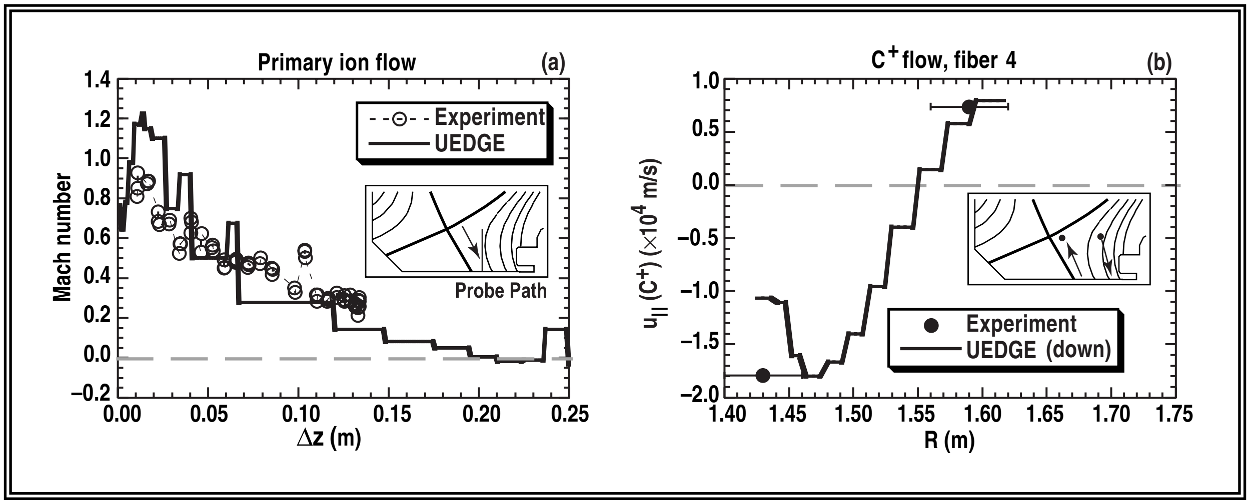

Fig. 2.4.1–4. Comparisons of measurements of (a) primary ion flow from the Mach probe and (b) C+ velocity from spectroscopy with calculations from the UEDGE code for an (OSP-attached) ELMing H–mode.

near a force null, and the position of this null is determined by the details of the parallel ion temperature gradient and the flow pattern of the fuel ion species (i.e. the frictional drag term).

One strength of the UEDGE modeling is the ability to calculate predicted diagnostic data directly (e.g., the Mach probe does not scan along a field line). In Fig. 2.4.1–4(a), we compare direct measurements of the deuteron Mach number from a scanning probe [12,22] with a UEDGE calculation of the data. The data were obtained along a vertical cut through the plasma in the SOL, and the UEDGE results were calculated for the same discharge and plasma shape. The flow in the SOL is towards the plate. In Fig. 2.4.1–4(b) is a comparison of the C^+ ion velocity distribution [6,23] from a tangential spectrometer chord and the UEDGE calculation. There is usually a flow away from the plate near the separatrix, and a flow towards the plate farther out in the SOL. As shown in Fig. 2.4.1–4, there is reasonable agreement between the data and the model in attached discharges. In detached discharges, at the location of the Mach probe, we do not see a large change in the flow velocity, but the 2-D UEDGE calculation indicates that there are strong gradients and more 2-D data is required for a better comparison. The measured carbon velocities tend to be more towards the plate in detached discharges: the flow away from the plate decreases, and the flow towards the plate increases. Detailed comparisons with multiple chords of spectroscopy are in progress.

We have also identified new, interesting physics in the interface between the SOL and the “private flux” region below the x-point. A reversal in the plasma parallel flow has been observed by the Mach probe, as shown in Fig. 2.4.1–5(a). Potential measurements from the probe also indicate large radial electric fields. The corresponding E×B drifts can drive an appreciable fraction of the recycling flux from the outer leg poloidally across the private region to the inner leg. The DTS system also measures large plasma density in the private flux region. We have also observed significant recombination radiation in the private flux region during the transition to outer leg detachment. The UEDGE model can now include drifts in the calculation, and comparisons with data are in progress [24].

Erosion and Redeposition Measurements and Modeling. To predict the lifetime of plasma-wall surfaces in future machines, we must understand the physical processes taking place between the divertor plasma and

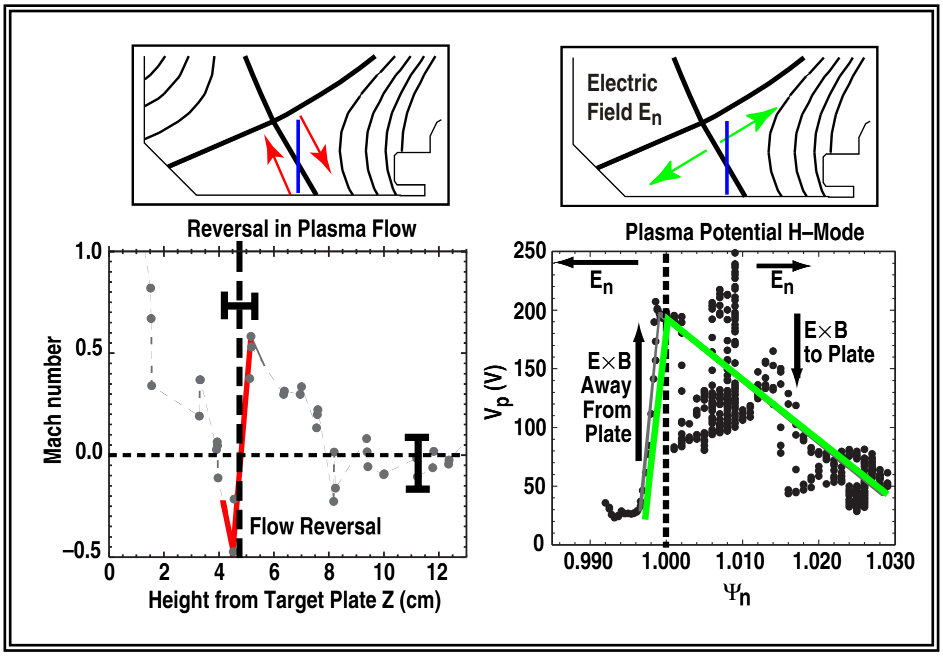

Fig. 2.4.1–5. We are observing new physics at the separatrix between the outer SOL and the private flux region. There is a reversal in the main ion flow (left) and a sharp change in the potential (right), implying an electric field. The corresponding E×B drifts can drive appreciable particle currents.

the target plate material. We have made direct measurements of erosion and redeposition at the OSP with an insertable DiMES probe [25] for comparison with models. A key to the comparison of these data with experimental models is the characterization of the plasma near the probe with the diagnostics shown in Fig. 2.4.1–1 and the data in Fig. 2.4.1–2(b), as the determination of quantities such as the input ion flux,

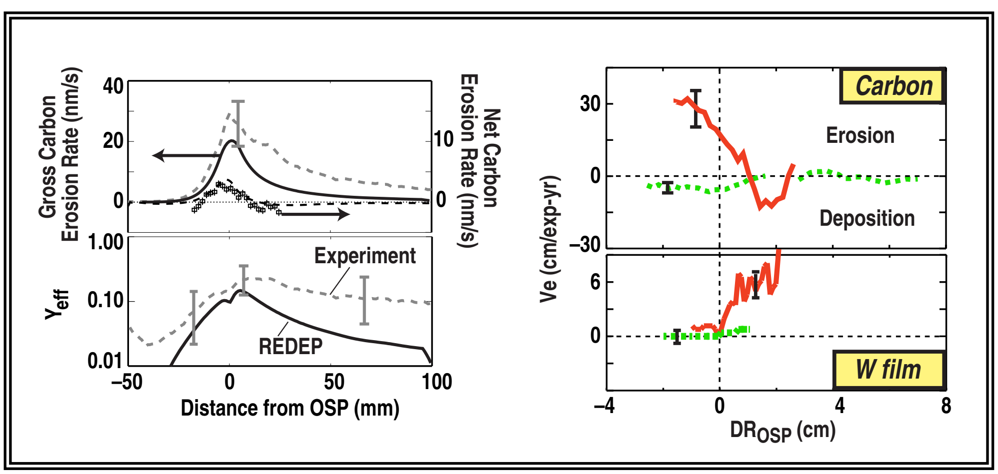

Fig. 2.4.1–6. (Left) The gross and net erosion from the DiMES probe (dashed) compared with the REDEP code (solid) in attached plasma operation. The effective sputtering yield agrees well with the code prediction. (Right) The effective sputtering yield for carbon (top) and tungsten (bottom) for attached (solid) and detached (dashed) plasma operation; note the reduction in erosion for the PDD case.

prompt redeposition, and self-sputtering require these data. Shown in Fig. 2.4.1–6 is a comparison of the gross and net erosion obtained from a carbon DiMES sample in an attached plasma, along with a calculation from the REDEP code [26]. Also shown (right) are the erosion for attached (solid) and PDD (dashed) plasma operation for carbon and tungsten DiMES samples. Note the dramatically reduced erosion near the strike point in the carbon sample for the PDD case. The erosion rates for the attached cases are greater than 10 cm/exposure-year, even with incident heat flux <1 MW/m^2. In this case, measurements and modeling agree for both gross and net carbon erosion, showing the near- surface transport and redeposition of the carbon is well understood. Self-sputtering and oblique incidence are important, and the effective sputtering yields exceed 10%. The private flux wall is measured to be a region of net redeposition with attached divertor plasmas. Divertor plasma detachment eliminates physical sputtering, spectroscopically measured chemical erosion yields are also found to be low ($Y(C/D^+) \le 10^{-3}$). This leads to suppression of net erosion at the outer strike-point, which becomes a region of net redeposition (~4 cm/exposure-year). Leading edge erosion, and subsequent carbon redeposition, caused by tile gaps can account for half of the deuterium codeposition in the DIII–D divertor.

Particle Control with Baffling and Cryopumping. We have recently installed and conditioned a new upper divertor baffle and cryopump whose geometry is matched for particle control in high-δ plasmas (δ ~ 0.7). The shape of the baffle was designed with a combination of UEDGE and DEGAS modeling [27], and the width is a optimum between a very closed baffle for neutrals (which can introduce recycling directly into the plasma core), and a open baffle that allows neutral leakage (with little recycling). A detailed comparison of unpumped open/closed divertor operation was carried out with carefully matched plasmas as shown in Fig. 2.4.1–7; the ion ∇B drift was towards the plate in each case. We observed that the line-average density was very similar in the two cases, but the midplane H_α (a tangential view just inside the separatrix) was reduced for the closed case. The density gradient at the separatrix, a measure of the core ionization rate, is reduced in the closed geometry. Transport modeling indicates that the ratio of the (open/closed) core ionization source was reduced by a factor of about F = 2.6; we normalized to the target plate current for each discharge. No changes in energy confinement during ELMing H–mode operation were observed, but the density at which the PDD occurred was decreased by 20% in the closed divertor.

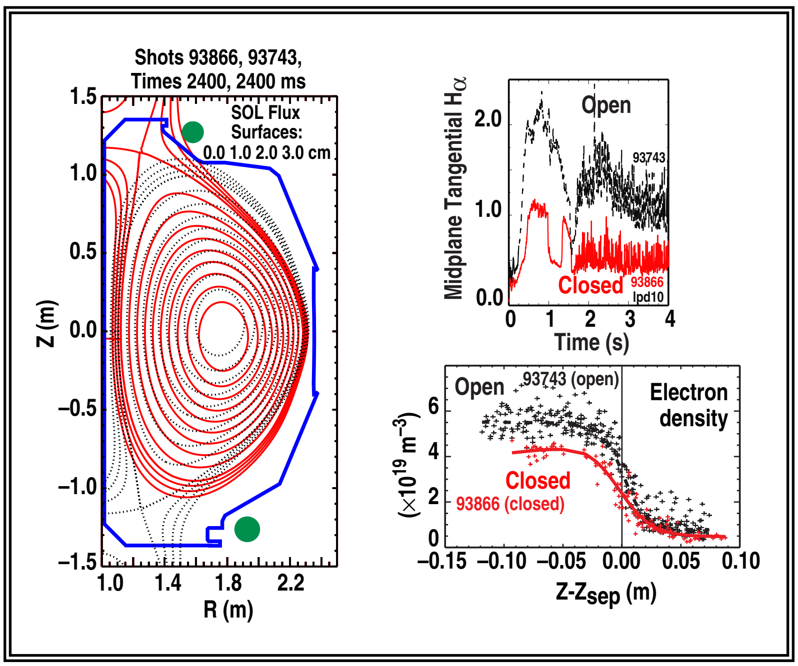

Fig. 2.4.1–7. Carefully matched plasmas were used for comparison of open (dashed) and baffled (solid) divertor configurations. We observed a drop in the edge tangential H_α signal, and a change in the edge density profile, implying a net reduction in the core ionization source with the baffled divertor.

We compared the measured reduction in core ionization with two computational models. A UEDGE model, in which the density profile is self-consistently calculated with a fluid neutrals model results in an estimate of F = 2. We have also calculated F = 3.8 for a fixed UEDGE core plasma model and a full Monte-Carlo neutrals calculation with DEGAS [27,28]. Coupled self-consistent UEDGE-EIRENE calculations are in progress. Using the second technique, we estimate for a more closed divertor with a private-flux space "dome" and inside baffle that F = 9 for SN operation. A full double null installation is required for F = 9 for DN. The UEDGE modeling has also shown that the shape of the baffle can influence the plasma flows, and we are optimizing the shape to see if more effective enrichment of impurities is possible in a closed divertor.

With the upper cryopump on, we achieved active density control with n_e/n_{GW} = 0.27 (fraction of the Greenwald density), which is close to the 0.22 achieved with the lower pump. This establishes an important particle control tool for high-δ plasma operation in DIII–D. The particle exhaust of the upper pump is similar to the lower pump, except at low $n_e < 5\times10^{19}$ m^{-3}. We are currently determining if this low exhaust at low density is due to better wall conditioning or reduced pumping effectiveness. Wall conditioning has controlled impurities in the baffled divertor so that $Z_{eff} \sim 2$. We routinely injected trace neon impurities to measure the core impurity confinement in several configurations. We found that at high density, the core neon decay time was similar for upper and lower divertors, but at low density, the time was significantly longer for the upper baffled pump.

Double Null Divertor Plasma Operation. We have continued research in double null divertor plasma operation. Previously, we showed that a balanced up/down heat flux required an unbalanced magnetic configuration [4]. Recent density control experiments with the upper high-δ cryopump have shown that the pump exhaust (with one pump in a balanced DN configuration) is about 50% of that in a single null plasma. As shown in Fig. 2.4.1–8, a nearly upper single-null plasma (with drsep = 5 cm, the distance between the two separatrices mapped to the midplane) is required to obtain maximum exhaust with one pump.

We have also observed that the role of neutrals may be different between DN and SN discharges [29], in that D_2 gas puffing in DN discharges produced only modest reductions in P_e, and core MARFEs were not observed, even at n_e approaching n_{GW}.

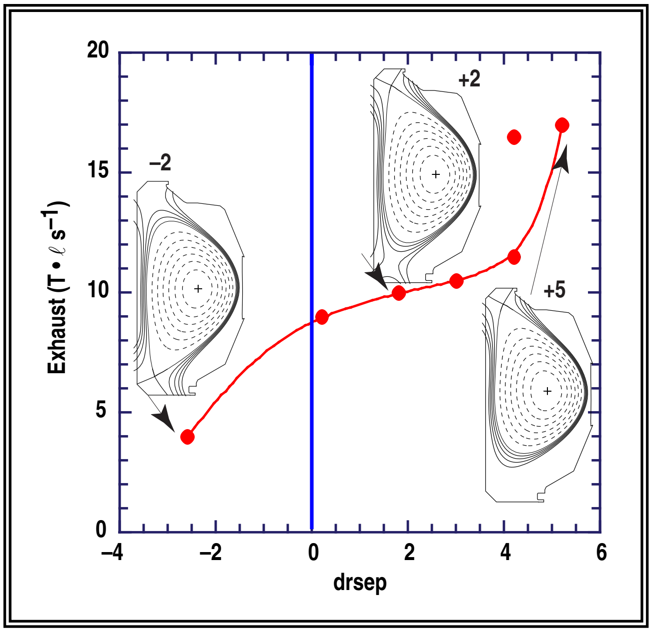

Fig. 2.4.1–8. The pump exhaust to the upper cryopump as a function of SN and DN plasmas. The abscissa, drsep, is the distance between the upper and lower separatrix measured at the midplane, drsep = 0 is a magnetically balanced double null plasma.

Conclusions and Future Work. In conclusion, we have used a comprehensive diagnostic set and state-of-the art computational models to gain a better understanding of convective heat flow, recombination processes, deuteron and impurity ion flows, and the effect of divertor geometry and baffles. In 1999, we will install a third divertor cryopump, private-flux-space dome, and inside baffle which will is expected to provide particle control at the inner strike point and increase our particle control capability in high-δ plasmas. New diagnostics, such as a Penning gauge and spectroscopy chords in the upper divertor, will be added to study the impurity effects in the more baffled divertor.

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2.4.2. PHYSICS OF THE DETACHED RADIATIVE DIVERTOR REGIME IN DIII–D[11]

Abstract. This paper summarizes results from a 2-dimensional (2D) physics analysis of the transition to, and stable operation of, the partially detached divertor (PDD) regime induced by deuterium injection in DIII–D. The analysis [1] shows that PDD operation is characterized by a radiation zone near X–point at T_e ~ 8–15 eV which reduces the energy flux into the divertor and thereby also reduces the target plate heat flux, an ionization zone below the X–point which provides a deuterium ion source to fuel parallel flow down the outer divertor leg, an ion-neutral interaction zone in the outer leg which removes momentum and energy from the flow, and finally a volume recombination zone above the target plate which reduces the particle flux to the low levels measured on the plates and thereby also contributes to reduction in target plate heat flux.

The peak heat flux striking divertor target surfaces in future tokamak devices must be reduced from that which is predicted based on ELMing H–mode operation of present experiments (see for example estimates of the local peak heat flux to surfaces in the ITER divertor [2]). The heat flux reduction technique which has received the most detailed experimental investigation on the DIII–D tokamak is to induce detachment of both divertor legs and large radiated power in the divertor away from the target plates by strong deuterium gas puffing. Frequently the outer leg plasma after gas injection is only detached from the target plates on flux surfaces near the separatrix and remains attached on flux surfaces farther out in the outer leg scrape-off-layer (SOL) [3,4]. This Partially Detached Divertor (PDD) is attractive because the peak heat flux near the separatrix is reduced typically by factors of 3–5, which would be sufficient for ITER requirements, and an attached plasma remains in the outer SOL from which helium ash could be pumped in a tokamak reactor. To understand the physics that produces the radial structure of this operating mode requires detailed two dimensional (poloidal and radial) measurements and computer simulations; these were done on DIII–D and are summarized below.

The paper is organized as follows. A one-dimensional model of the key physics mechanisms in a PDD plasma is described in Section 2. Section 3 summarizes key results from a comparison of conditions in the divertor before gas injection and after establishing the steady PDD conditions [1]. Signatures of the initiation and evolution of the transition to PDD conditions are described briefly in Section 4. In Section 5 possible source mechanisms for the carbon needed to initiate and sustain the PDD are described. Conclusions are given in Section 6.

One Dimensional Model of Outer Leg During PDD. A one dimensional (1D) flux tube model of the outer leg during PDD operation has five regions from the midplane to the target, each dominated by different physics processes: 1) a zone dominated by thermal conduction, 2) a radiation zone, 3) a deuterium ionization region, 4) a volume dominated by ion-neutral interactions, and 5) a region dominated by deuterium volume recombination (Fig. 2.4.2–1). At the midplane T_e is high (~100 eV), the SOL screens the plasma at the separatrix from thermal neutrals and the energy transport is governed by the conductivity equation, $q_{||} \sim T_e^{5/2}\, dT/ds$ where s is the parallel length coordinate along the field line. In the region near the X–point measurements show high radiation levels during PDD operation. This radiation dissipates energy from the

[11]Fenstermacher, M.E., J. Boedo, R.C. Isler, et al., "Physics of the Detached Radiative Divertor Regime in DIII–D," Proc. 25th European Physical Society Conf. on Controlled Fusion and Plasma Physics, June 29–July 3, 1998, Zofin, Praha, Czech Republic; General Atomics Report GA–A22909 (to be printed).

flux tube and T_e is reduced substantially. The measured electron density is not substantially increased so the drop in T_e produces a drop in pressure. Below the X–point the measured T_e is low enough (~5 eV) and the neutral density is high enough that an ionization region forms. This produces a source which leads to poloidal flow of primary ions toward the target plates. Farther downstream ion-neutral interactions begin to dominate at the low T_e and high neutral density found in the lower part of the outer leg. These collisions can remove parallel momentum across the field lines from the plasma flow. This effectively reduces the flow velocity toward the target plate. For the region in which the flow velocity is low enough that the transit time through a volume is comparable with the recombination time, substantial recombination takes place. This occurs in the volume above the target surfaces and effectively reduces the ion current striking the plates near the OSP to substantially lower values than in the attached plasma conditions. The combination of low ion current recombining in the plate and reduced energy transport to the plate, due to the high radiation near the X–point, produces the observed low peak target heat flux near the separatrix strike point. The measured energy transport and radiation in the regions of the flux tube below the ionization front can not be accounted for by thermal conduction; convection by Mach~1 plasma flow is consistent with the measurements [5–8].

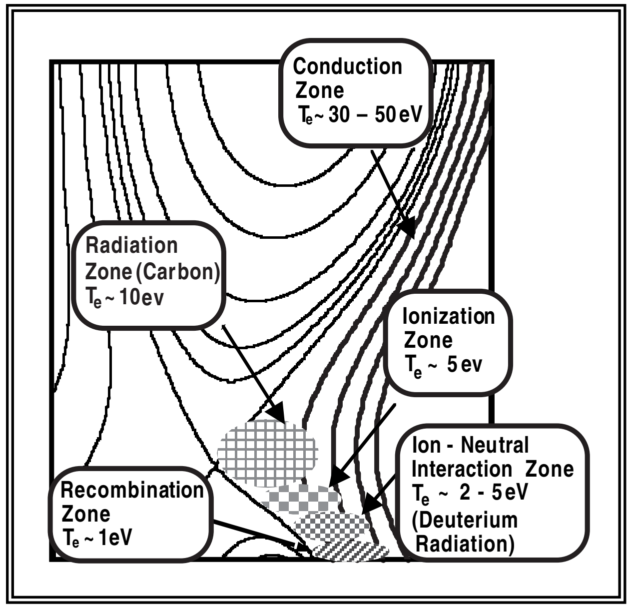

Fig. 2.4.2–1. Schematic diagram for the one dimensional model of PDD conditions in the outer divertor leg of DIII–D. Regions dominated by radiation (carbon) near the X–point, deuterium ionization and recombination, and ion-neutral interactions are shown. Conduction dominates the energy transport above the X–point; convection dominates below the ionization region.

Comparison of Conditions in Attached and Detached Divertor. Deuterium gas injected at a high rate (~200 Tℓ/s or ~27 Pa m^3/s) for several hundred milliseconds into a lower single-null ELMing H–mode plasma (∇B toward the divertor) followed by a reduced steady injection rate (~ 50 Tℓ/s or ~7 Pa m^3/s) is used to establish PDD conditions in the divertor while retaining good H-mode confinement ($H_{ITER89P}$ = $t_E/t_{ITER89P}$ ~ 1.6) and low impurity content [$Z_{eff}(\rho=0)$ ~ 1.2, $Z_{eff}(\rho=0.7)$ ~ 1.6] in the core plasma (Fig. 2.4.2–2). The peak heat flux is reduced by about a factor of 5 near the outer strike point (OSP), compared to the attached conditions prior to the gas injection. Total radiated power increases by 80% to P_{rad}/P_{inj} = 0.75 consistent with the reduction in total heat flux to the target plates, with most (60%) of the increase occurring in the divertor and little (<10%) inside the last closed flux surface, LCFS. With continued low gas injection, the PDD conditions can be sustained for the remainder of the discharge (10–20 energy confinement times).

Experimental Observations. Comparisons of 2D bolometer reconstructions with 2D profiles of deuterium and carbon radiation from the tangential visible TV show that in PDD operation the majority of the radiation is along the outer divertor leg separatrix with carbon radiating near the X–point and deuterium radiating nearer to the OSP. Images of the *difference* between the pre-injection and the PDD states are shown for

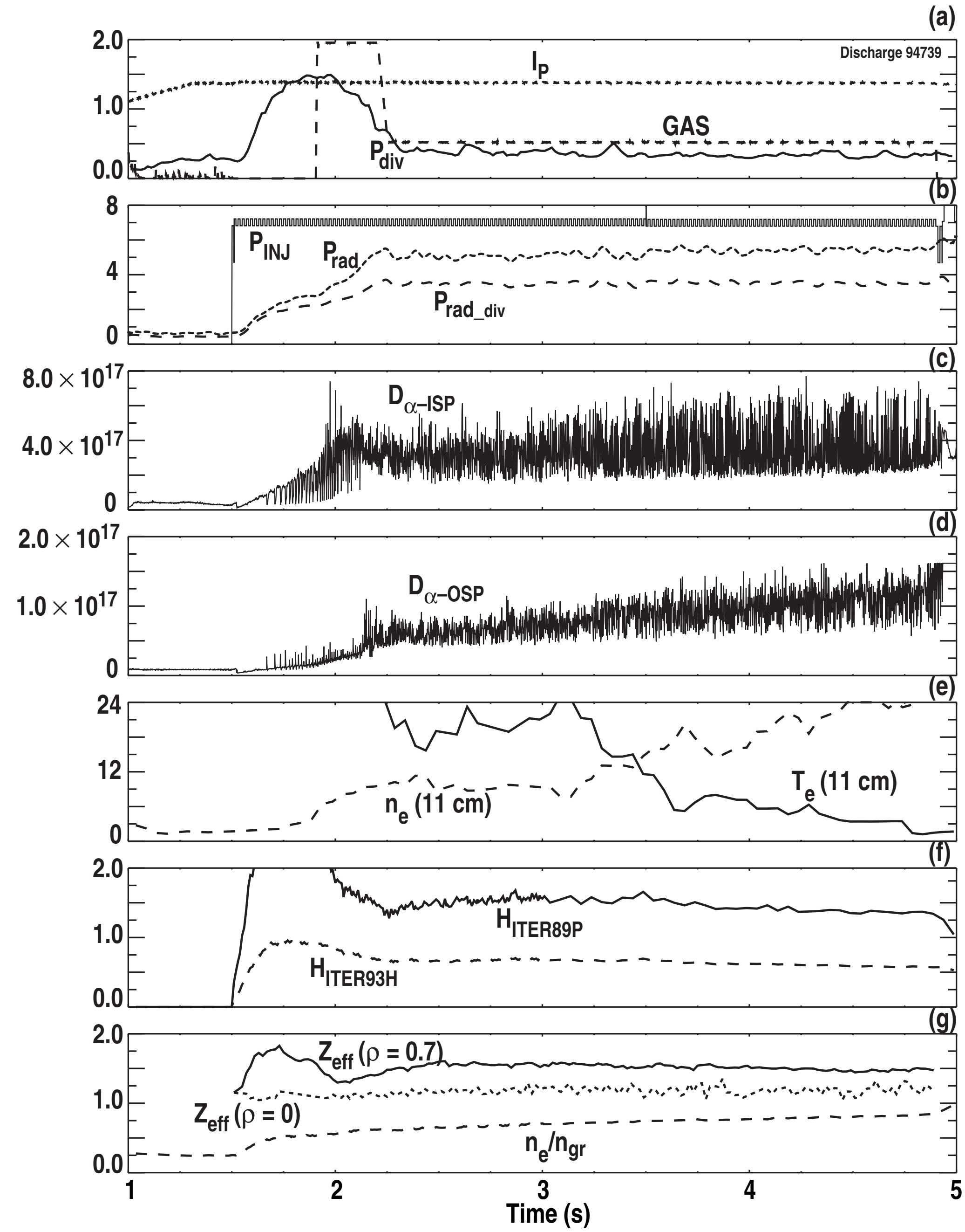

Fig. 2.4.2–2. Time history of a typical LSN discharge with a PDD phase beginning at 2350 ms. Traces shown are (a) the plasma current (I_p), deuterium gas injection (Gas), peak outer divertor leg heat flux (P_{div}), (b) neutral beam injection power (P_{inj}), total and divertor radiated powers (P_{rad} and P_{rad_div}), (c) D_α emission at the ISP and (d) at the OSP, (e) electron temperature (T_e) and density (n_e) at 11 cm above the divertor floor from divertor Thomson scattering, (f) energy confinement enhancement factors normalized to ITER89P and ITER93H scalings, and (g) core line averaged density normalized to the Greenwald density, and Z_{eff} at $\rho = 0$ and $\rho = 0.7$.

the total radiated power [Fig. 2.4.2–3(a)], the CIII emission [Fig. 2.4.2–3(b)], and the deuterium (Balmer_α) emission [Fig. 2.4.2–3(c)]. Taken together these profile changes are consistent with a model in which the inner leg has completely detached and cooled up to the X–point to temperatures below which neither carbon nor deuterium radiate substantially (T_e <~ 1 eV). In addition, the profiles are consistent with outer leg temperatures of 1–3 eV for deuterium to radiate, and at the X–point, T_e ~ 8–15 eV for substantial radiation from CIII and CIV.

Vertically integrated spectral measurements from the divertor VUV SPRED spectrograph [9] have confirmed, by direct measurement of the strongest radiating lines, that carbon is the main radiator in DIII–D detached plasmas (80% of the total radiated power) with deuterium contributing the remaining 20% of P_{rad}. The highest power line is CIV at 155 nm. Substantial contributions from CIII at 117.5 and 97.7 nm, and from CII at 133.4 nm are also observed [1] as is deuterium Ly-α emission (121.6 nm).

Comparisons of temperature, density, ion flux and heat flux profiles on the outer target plate before and during PDD operation [10] show that in PDD the temperature is significantly decreased across the profile, the density peaks radially outboard of the separatrix, the heat flux reduction is largest near the OSP and the ion flux is reduced near the OSP but increased farther out in the outer leg SOL. These are the signature characteristics of PDD operation; reduced heat and particle flux near the separatrix strikepoint (detachment) with high particle flux and finite heat flux radially outboard of the OSP.

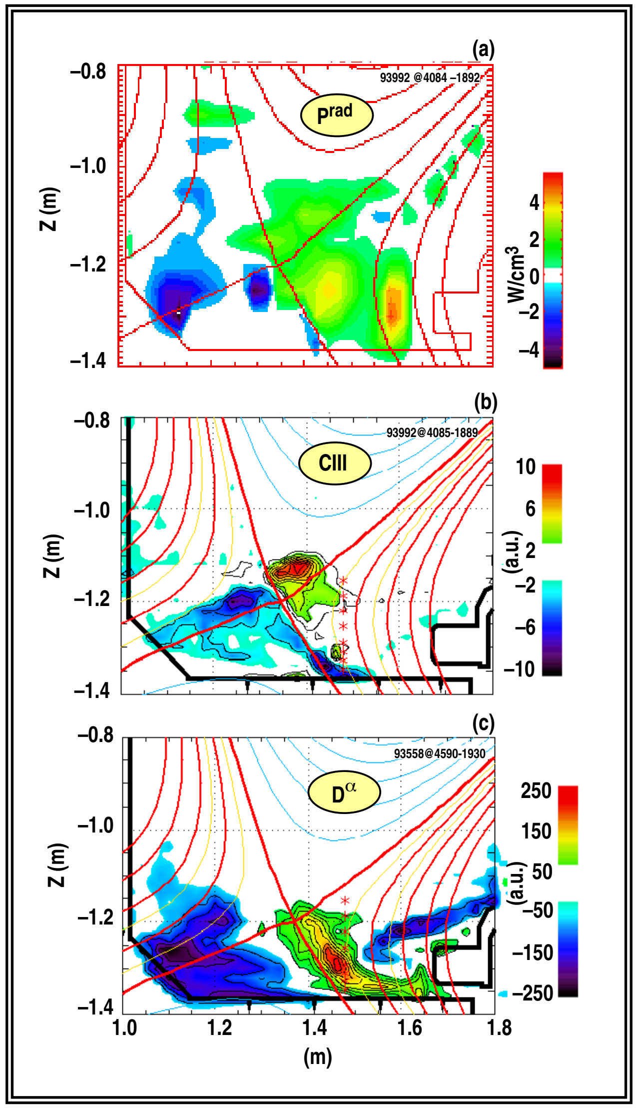

Fig. 2.4.2–3. Profiles of the *difference* between the 2D profiles during the PDD phase and the profiles during the pre-puff phase for (a) total radiated power reconstructed from the 48-channel bolometer array, (b) CIII (465 nm) visible emission and (c) D_α (656.1 nm) emission, both reconstructed from the tangentially viewing visible video system. In all three cases the signals decrease in the inner leg region during PDD. The total radiation increases along the entire outer leg and at the X–point. The radiation in the lower part of the leg is from deuterium; radiation near the X–point is from carbon.

Reconstructions of divertor Thomson data from shots with radial sweeps of the X–point [11] produce 2D profiles of n_e and T_e showing that in the pre-injection phase [Fig. 2.4.2–4(a,b)] the electron pressure on flux surfaces is nearly constant, within a factor of 2, in contrast to PDD operation [Fig. 2.4.2–4(c,d)] in which there are large pressure variations both along flux surfaces (X–point to target) and radially in the SOL. In the pre-puff phase the electron temperature, and density (and therefore pressure) are nearly constant, within a

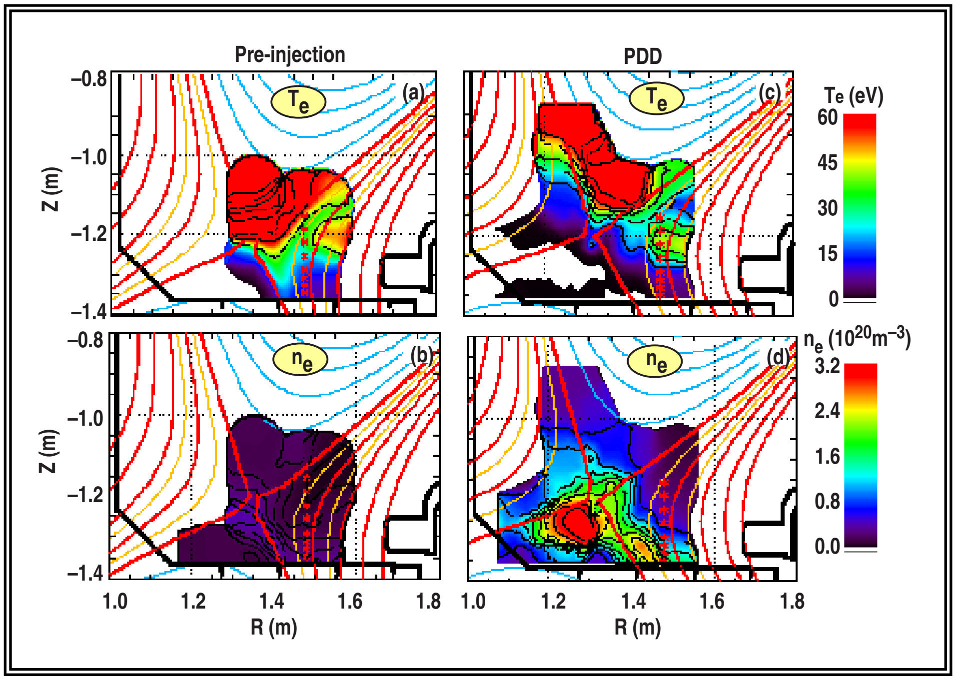

Fig. 2.4.2–4. Reconstructions of divertor plasma parameters from divertor Thomson scattering data in the pre-injection phase (a–b) and in the PDD phase (c–d). The electron temperature T_e is given in (a) and (c), and the density n_e in (b) and (d). In the pre-injection phase the temperature and density are constant along flux surfaces near the separatrix within a factor of 2. In the PDD phase T_e is reduced to low values (1–3 eV) throughout the divertor and n_e in the outer leg peaks radially outboard of the separatrix. A high density, low temperature MARFE-like structure is measured below the X–point.

factor of 2, on flux surfaces in the outer leg SOL. During the PDD phase temperature is low throughout most of the divertor; T_e ~1–3 eV. The density in the SOL peaks radially outboard of the separatrix. This gives rise to a drop in pressure along flux surfaces of a factor of 3–5 from the X–point to the target plate. Finally, a high density (n_e = 2–4 $\times 10^{20}$ m^{-3}), cold (T_e = 1–2 eV) MARFE-like structure is observed below the X–point in the private flux region in many of the PDD plasmas (see Ref. 1).

The D_α emission attributable to recombination, obtained by taking the ratio of Balmer_α to Balmer_γ emission [12], changes from a distributed region across the inner target plate in the pre-injection phase [Fig. 2.4.2–5(a)] to a region near the outer leg separatrix and target plate during PDD operation [Fig. 2.4.2–5(b)]. The profile before gas injection (ELMing H–mode) gives an indication that the inner leg plasma is nearly detached. During the PDD phase the profile in the outer leg SOL shows recombination occurs farther up the leg toward the X–point on surfaces close to the separatrix; emission farther out in the SOL is close to the target. This is consistent with theories which emphasize the importance of neutrals in triggering detachment [13], especially those models which propose that the neutrals enter the outer leg SOL by migrating across the separatrix from the private flux region [14].

The change in the D_α emission due to ionization [12] is from a poloidally narrow layer upstream of the recombination zone in the inner leg prior to gas injection to a zone in the outer leg below the X–point in the PDD phase [Fig. 2.4.2–5(c,d)]. These profiles in both phases are consistent with a model [15] in which an ionization layer near the T_e=5 eV surface provides a source which drives plasma flow toward the plates and a recombination region at lower temperature and higher density provides the ion sink

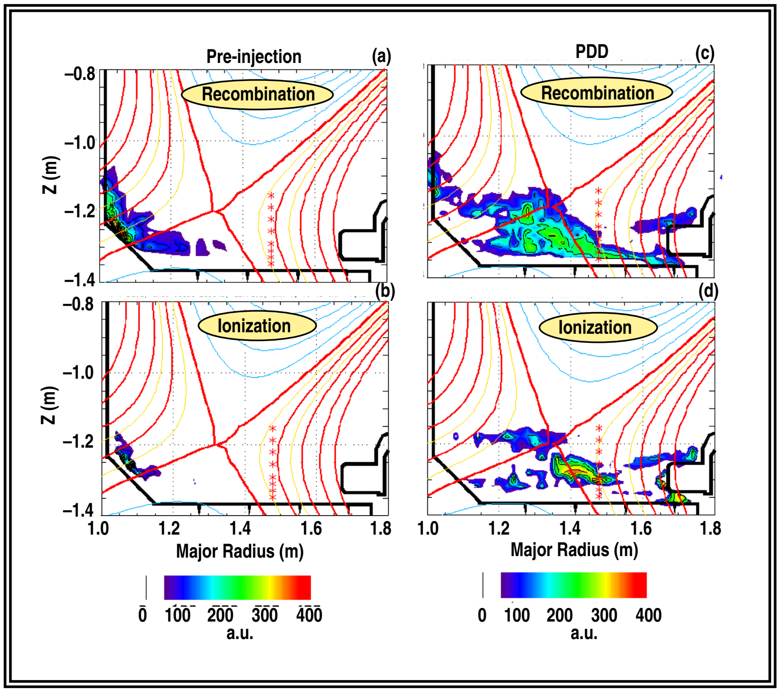

Fig. 2.4.2–5. Reconstructions of TTV D_α data in a poloidal plane in the pre-injection phase (a), (b) and the PDD phase (c), (d). The D_α emission due to recombination is shown in (a) and (c); the D_α emission from excitations in an ionizing plasma is shown in (b) and (d). In the pre-injection phase recombination is near the inner target; ionization a few cm upstream poloidally. During PDD operation recombination is in the lower outer leg and private flux region; ionization poloidally upstream in the mid-outer leg and in the inner SOL near the X–point.

which keeps the ion flux to the target plates low.

Measurements of carbon ion (C1+ and C2+) flows [16,17] show significant differences between PDD operation and pre-puff conditions; the *change* in the flow velocities in all regions of the divertor which have strong CII and CIII emission is in the direction toward the outer target plate during PDD operation. The flows of C1+ derived from measurements along chord T4 [Fig. 2.4.2–6(a)] are shown in Fig. 2.4.2–6(b). Analysis of the Zeeman splitting of the spectrum components indicates that in the outer SOL during the attached phase the flow is toward the target plate (forward) and near the separatrix the flow is away from the target (reversed). When the PDD is established (2500 ms in the figure) both the forward and reversed components of the C1+ flow pattern change in the direction of increased flow toward the outer target. The forward flow increases almost a factor of 2 and the magnitude of the reversed flow decreases by a factor of 4. The measured velocities compare favorably with UEDGE modeling results [Fig. 2.4.2–6(c) and discussion below].

Two Dimensional Modeling. Simulations of PDD plasmas with the multi-fluid UEDGE code [18] now include the effect of gas puffing to produce the detachment and a more realistic carbon sputtering model [19] than was used previously. As in the previous work, many of the macroscopic features of PDD conditions can be reproduced simultaneously in a single calculation with the new carbon model including: 1) the midplane n_e and T_e profiles, 2) the heat flux profile at the targets, 3) the low ion flux at the targets, 4) carbon radiation near the X–point and deuterium radiation near the outer target. The best matches to the data are for simulations with less than (typically 50% of) the full sputtering yield predicted by the Haas model [19]. This is within the uncertainty of the model for the low temperatures in the PDD divertor plasma.

Initial simulations of carbon (C1+) flow profiles [20] reproduce the essential feature of flow measurements by the tangential spectroscopy diagnostic, namely: the *change* in the carbon flows is in the direction of the plates in PDD compared with the pre-puff phase. The comparison of C1+ ion flow with the spectroscopy measurements [Fig. 2.4.2–6(c)] shows that, for attached plasma conditions, the simulations reproduce both the forward flow toward the outer divertor plate in the outer regions of the SOL (data at

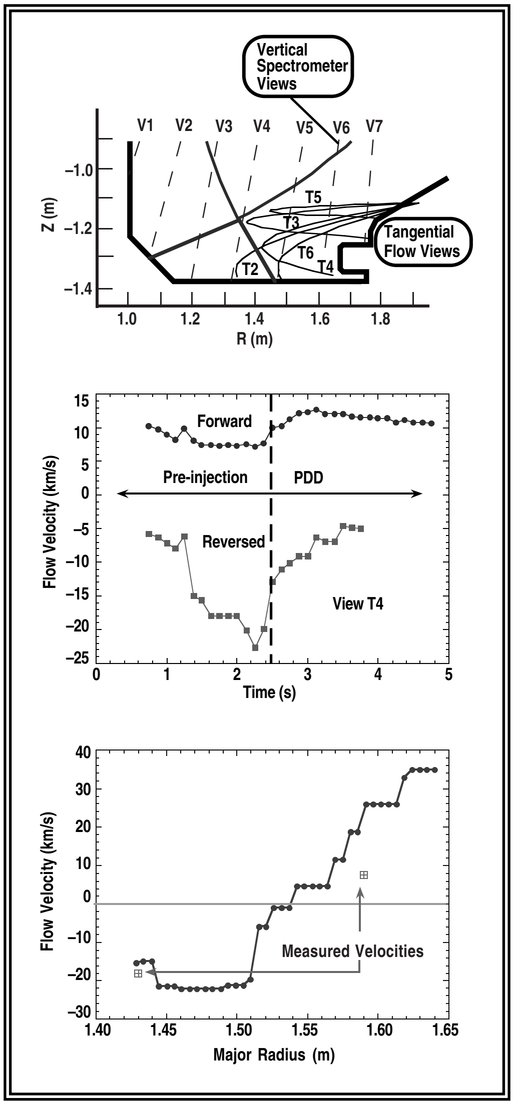

Fig. 2.4.2–6. Flow velocities of C^+ ions from the tangentially viewing flow diagnostic using Doppler shift spectroscopy of CII 657.9 nm emission and simulation results from UEDGE. In (a) a schematic diagram of the views of the flow spectrometer chords [vertical (V1-V7) and tangential (T2–T6)] projected onto a poloidal plane and the location of the reciprocating probe plunge (arrow), in (b) the "positive" C^+ flow toward the outer target plate and "negative" flow away from the plate flow both show a change in the direction of the target plates when the PDD conditions are established. In (c) a profile along the T4 line of sight (line with dots) calculated from UEDGE simulations of the pre-injection phase shows good agreement, within a factor of 2, with both the positive and negative measured velocities.

R=1.59 m) and the reversed flow in regions closer to the separatrix (data at R=1.43 m).

The essential features of the deuterium flow measurements by the reciprocating probe [7,8] and the tangential spectrometer [17] are also reproduced by UEDGE [20], namely: 1) deuterium flow away from the plates near the separatrix in the upper leg and toward the plates in the outer SOL in the pre-puff phase, and 2) deuterium flow only toward the plate near the OSP in PDD.

Initiation and Evolution of the Transition from Attached to Detached Conditions. Reconstructions of emission from the constituent radiators at the initiation of the transition to PDD conditions [12] show a sharp increase in D_α emission in the private flux region near the ISP and increased carbon radiation both along the outer divertor leg and in the inner SOL near the X–point. The carbon emission at the start of the transition increases in the entire outer leg and in the X–point region in addition to the inner leg SOL near the X–point. As the transition progresses the CIII profile becomes more localized near the X–point. The D_α emission from recombination is reduced near the ISP at initiation, increases radially inboard of the inner separatrix (in the inner leg SOL) near the target plate and increases in the private flux region near the ISP. As the transition progresses the emission moves across the private flux region and also up to the inner SOL near the X–point. During the slow evolution later in this discharge (Fig. 2.4.2–2) the carbon radiation gradually becomes more localized at the X–point (peak intensity unchanged but less emission in the lower leg), and deuterium recombination emission spreads radially outward near the outer target and poloidally up along the outer separatrix. These observations are consistent with Divertor Thomson measurements showing a gradual cooling of the plasma from the outer target plate upward during this evolution.

Carbon Sources. There is evidence of contributions from both chemical and physical sputtering by neutrals in the divertor and possibly of carbon sublimation sources. Sputtering by neutrals, either by chemical processes or direct physical sputtering by fast charge exchange particles, may be the dominant source of carbon in the DIII–D divertor during PDD operation. This indication was obtained by comparing the carbon content in the core and divertor plasmas in deuterium versus helium PDD [9,10,21–23].

The response of the core carbon content in helium plasmas compared with that in deuterium plasmas is consistent with a carbon source in the divertor from neutral sputtering. There are indications [23] that the carbon sputtered by chemical processes may be generated in a location from which it has a higher escape probability from the divertor into the core than that from neutral physical sputtering. There is also evidence that the source of the carbon that controls the core content is likely not from charge exchange neutral sputtering of the walls around the main plasma chamber, but instead is more likely from divertor sources. Simulations of attached plasmas with UEDGE support the hypothesis that the carbon source controlling the core carbon content is from the divertor region [20].

Conclusions. The data presented above show that PDD formation and sustainment are inherently two dimensional phenomena; although the 1D model of outer leg detachment provides a good framework for understanding the plasma properties in the poloidal regions along a divertor flux tube, the 2D measurements show that the critical sources and sinks for particle and energy depend on behavior in other regions of the divertor. The PDD regime is characterized by reduced target plate heat flux and ion current near the strikepoints, enhanced upstream impurity radiation, and low plasma temperature (T_e=1–3 eV) in much of the divertor, leading to volume recombination of the plasma before reaching the target plates. Reconstructions of impurity radiation profiles show that carbon is the strongest radiating constituent near the X–point; deuterium radiation dominates near the OSP. The data also shows that the key physics processes are: 1) the buildup of neutral deuterium in the private flux region by radial diffusion from the inner leg, 2) transport of carbon from generation at the graphite walls to radiation near the X–point, and 3) diffusion of neutrals to an upstream ionization region in the outer leg. This ionization region is poloidally separated from a volume recombination region downstream. The separation of ionization and recombination zones leads to parallel flow in the outer leg approaching Mach 1, and convective energy transport toward the target plates which may lengthen the radiation region along the divertor leg compared with that implied by conductive energy transport alone [5,6]. Measurements of the 2D profiles averaged over ELMs during the transition to PDD operation show a sequence of events after deuterium gas injection: 1) inner leg detachment up to the X–point, 2) the resulting migration of neutrals across the private flux region and 3) initiation of outer leg detachment by energy and momentum losses due to plasma interactions such as charge exchange with neutrals entering the SOL from the private flux region.

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2.4.3. IMPURITY CONTROL STUDIES USING SOL FLOW IN DIII–D[12]

In a high-power density fusion device, controlling impurities in the scrape-off-layer (SOL) is essential in obtaining high radiative power fractions in the SOL to protect the divertor surfaces from unacceptable heat loads and material erosion while maintaining acceptably low impurity contamination in the core plasma. A key issue in this regard is whether external control of impurities is possible through tailoring of the main ion flow in the SOL. The tailoring of the main ion flow through strong D_2 gas injection and simultaneous divertor exhaust has been an integral part of the DIII–D program since 1994 [1–4]. These experiments have demonstrated that induced SOL flow offers several potential advantages and two primary results have been obtained: (1) induced SOL flow improves impurity enrichment in the divertor plasma in both open and closed divertor configurations with argon enrichment values as high as 17 obtained in the highest flow cases; and (2) plasmas with $P_{rad}^{tot}/P_{input} > 80\%$ with a high SOL radiation fraction ($P_{rad}^{SOL} + P_{rad}^{div} > 60\%$ of P_{rad}^{tot}) have been obtained in ELMing H–mode plasmas with confinement ~1.0 $\tau_{ITER93H}$, $n_e = 0.75\ n_{eGW}$, and $Z_{eff} < 1.9$ using argon as the seeded impurity.

Impurity Enrichment Studies. The premise of the "puff and pump" technique used on DIII–D is that through simultaneous deuterium injection near the midplane and divertor exhaust one can augment the bulk plasma ion flow in the SOL sufficiently to overcome the thermal gradient force, which acts to drive impurities toward the core plasma [5,6]. Additional benefit may be gained through lowering of the SOL and divertor ion temperature, which increases the frictional force and reduces the thermal gradient force. Experiments have demonstrated that this technique is effective in increasing the enrichment of impurities in the DIII–D divertor [1,2] with the improvement being substantial for higher Z impurities [3]. To assess the effectiveness of this process in entraining impurities in the SOL, the typical figures of merit are the exhaust enrichment ($\eta_{exh} \equiv f_{exh}/f_{core}$) and compression ($C_{exh} \equiv n_Z^{exh}/n_Z^{core}$). On DIII–D, direct measurement of these quantities are made using charge-exchange recombination (CER) spectroscopy for the core impurity content and a modified Penning gauge for the exhaust gas impurity content. The results from these studies are summarized in Table 2.4.3–1. The observed exhaust enrichment η_{exh} is observed to be consistently higher in the top fueling case, indicating a beneficial effect of SOL flow on divertor retention

Table 2.4.3–1
Measured Enrichment (η_{exh}) and Compression (C_{exh})
for Various Flow Levels and Impurities

	Case A		**Case B**		**Case C**		**Case D**	
Fueling Location	Top		Divertor		Top		Divertor	
Fueling Rate(D°/s)	1.05×10^{22}		1.05×10^{22}		5.6×10^{21}		5.6×10^{21}	
	η_{exh}	C_{exh}	η_{exh}	C_{exh}	η_{exh}	C_{exh}	η_{exh}	C_{exh}
Helium	1.1	6.1	0.9	4.3	n/a	n/a	n/a	n/a
Neon	2.3	14.2	1.2	7.8	1.6	6.2	1.0	4.6
Argon	17.0	85.0	6.0	28.5	3.7	11.4	2.1	6.2

[12]Wade, M.R., J.T. Hogan, R.C. Isler, et al., "Impurity Control Studies Using SOL Flow in DIII–D," presented at 17th IAEA Fusion Energy Conference, October 19–24, 1998, Yokohama, Japan, to be published in a special issue of Nucl. Fusion; General Atomics Report GA–A23001 (1998).

of impurities. The results in Table 2.4.3–1 also suggest that enrichment is sensitive to the choice of impurity. For example, in the $1.0{\times}10^{22}$ D^0/s case, helium enrichment increases ~20%, neon enrichment increases ~90%, and argon enrichment increases ~200% when comparing the top D_2 fueling case (Case A) with the divertor fueling case (Case B). Similar trends in argon enrichment have been inferred from SPRED UV measurements of argon line radiation in the core and divertor plasmas [3]. Analysis shows that this strong Z dependence is consistent with a combined neutral ionization and ion transport picture of impurity entrainment in the divertor [3].

Besides the improvement in impurity enrichment, several advantageous by-products arise from the application of this technique. First, at a high flow level, the SOL broadens and its density increases to $1.5{\times}10^{19}$ m^{-3} while the electron temperature remains approximately 10 eV. Such profiles provide excellent screening of impurities emanating from the vessel wall and an excellent environment for impurity radiation. Secondly, the ELM amplitude is reduced by approximately a factor of two relative to standard ELMing H–mode conditions [7]. This reduction is accompanied by a proportional increase in the ELM frequency such that the time-integrated energy carried out by the ELMs is approximately the same, but the instantaneous perturbation on the edge and divertor plasma induced by each ELM is much smaller. Modeling has also shown that the ELM dynamics are important in the obtainable impurity enrichment with higher frequency ELMs leading to improved enrichment [8]. Note that these changes are accomplished without significant impact on the core energy confinement.

Radiative Divertor Plasmas. Motivated by the favorable argon enrichment values obtained in the induced SOL flow case, recent experiments on DIII–D have focused on producing radiative divertors using the "puff and pump" technique with argon as the seeded impurity. These experiments have produced radiative plasmas that possess many of the aspects required by the radiative "solution" embodied in the ITER design criteria. In particular, radiative fractions up to 75% with concomitant heat flux reduction (a factor of 4) have been achieved simultaneous with good core energy confinement ($\tau_E = 1.1\ \tau_{E,ITER93H}$) and minimal core contamination ($Z_{eff} < 1.9$). This represents the first successful demonstration of radiative divertor operation using a seeded impurity in which all of these requirements are achieved simultaneously. An overview of the results are presented here.

The best radiative discharges to date have been produced in a lower-single-null configuration with $I_p = 1.3$ MA, $B_T = -2.1$ T, $q_{95} = 4.1$, $\kappa = 1.75$, $<\delta> = 0.28$, and $P_{NBI} = 11.9$ MW. In these discharges, a SOL flow is applied through deuterium injection near the symmetry point at the top of DIII–D at rate of $2.45{\times}10^{22}$ D^0/s throughout the current flattop phase (Fig. 2.4.3–1). This strong D_2 flow alone produces plasma conditions in which radiation levels are higher than normally found in ELMing H–mode plasmas. During this phase, the total radiated power [Fig. 2.4.3–1(c)] represents approximately 50% (~6.0 MW) of the total input power ($P_{NBI} + P_{OH} \approx 11.9$ MW) with $P_{rad,div}$:$P_{rad,SOL}$:$P_{rad,core}$ (MW) = 3.5:2.0:0.5. The peak heat flux incident on the outer divertor target, inferred from IRTV measurements, is ~2.0 MW/m^2 [Fig. 2.4.3–1(d)], which is a factor of two lower than the value normally obtained at these power levels with no external gas injection [9]. Carbon is observed to be the dominant impurity in the core plasma with $Z_{eff} \approx 1.4$, [Fig. 2.4.3–1(e)] and carbon and deuterium are the main radiating constituents in the divertor.

A further increase in radiation is then observed upon the introduction of argon at 2.0 s, which is injected from the private flux region of the divertor at a rate of $1.26{\times}10^{21}Ar^0$/s. During the argon injection phase, the total radiation increases up to ~75% (8.5 MW) with $P_{rad,div}$:$P_{rad,SOL}$:$P_{rad,core}$ (MW) = 4.3:2.4:1.8. The radiation from the core plasma is localized in the last 10% of the plasma volume while

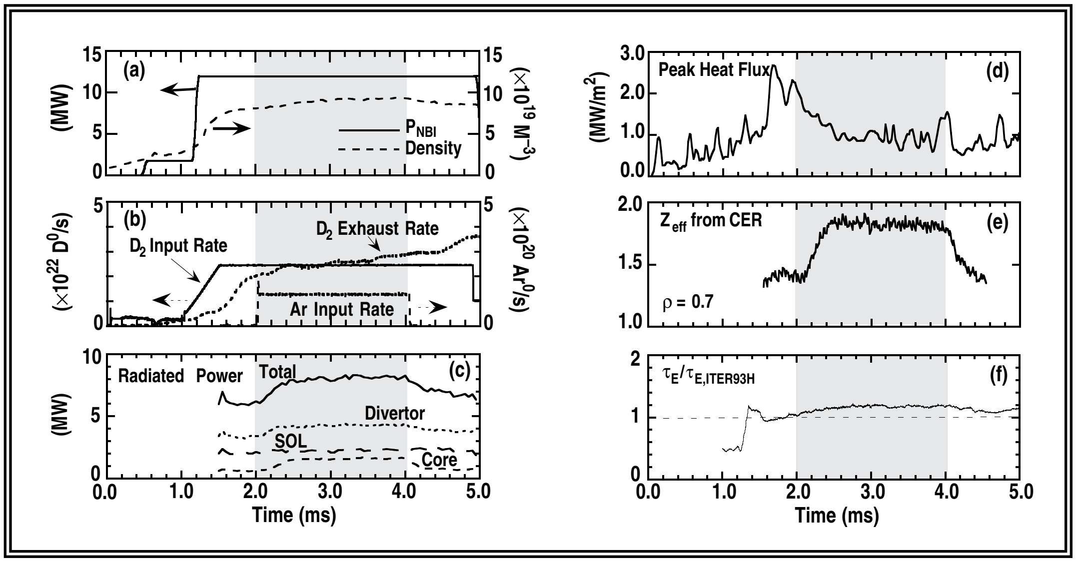

Fig. 2.4.3–1. Temporal evolution of the (a) input power and line-averaged electron density; (b) deuterium input and exhaust rates and argon input rate, (c) radiation power balance; (d) peak heat flux at the outer strike point; (e) Z_{eff} and (f) $\tau_E/\tau_{E,ITER93H}$ in a radiative discharge produced via argon injection simultaneous with strong D_2 injection and divertor exhaust.

the radiation in the divertor plasma is distributed fairly evenly over the entire divertor volume (Fig. 2.4.3-2). The increase in core radiation consists almost entirely of line radiation from argon and is consistent with the measured argon concentration (~0.20%) in the core plasma. Meanwhile, the increase in the divertor radiation includes roughly equal increases in both carbon and argon radiation; thus, carbon remains the dominant radiator in the divertor. Even though the biggest increase in radiation is observed to come from the core plasma, it is important to note that the divertor radiative efficiency (defined as the ratio of the power radiated in the divertor to the power conducted to the divertor) actually increases substantially during the argon injection phase from 40% up to 55%. The increase in radiation is accompanied by a factor of 2 decrease (relative to the pre-argon injection phase) in the total and peak heat flux incident on the outer divertor target plates [Fig. 2.4.3–1(d)]. The outward shift of the heat flux profile observed during the argon injection phase is indicative of partial detachment of the outer strike point [Fig. 2.4.3–2(b)]. However, Langmuir probes measurements at the outer strike point suggest that the outer divertor leg remains attached as the particle flux profile remains peaked at the separatrix location. Furthermore,

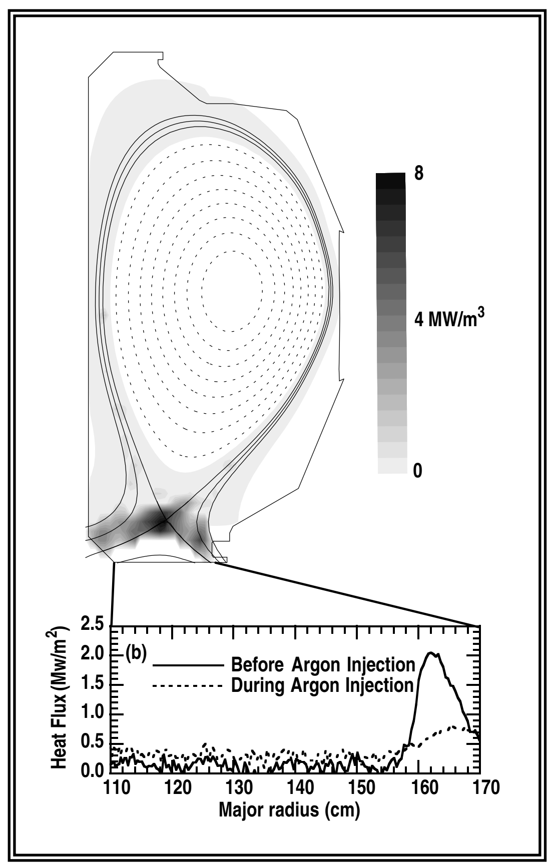

Fig. 2.4.3–2. Two-dimensional radiation profile inferred from bolometry data and the heat flux profile at the divertor target taken at 3.5 s of the discharge described in Fig. 2.4.3–1.

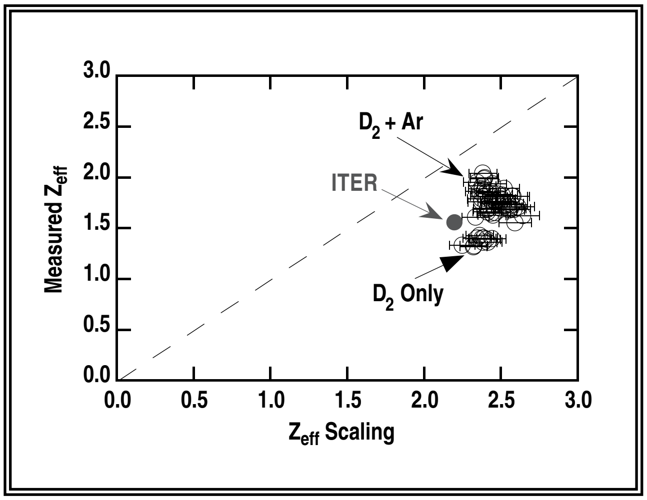

Fig. 2.4.3–3. Measured Z_{eff} versus the predicted scaling $Z_{eff} = 1 + 4.5\ P_{rad}\ Z^{0.12}/(S^{0.94}\ \bar{n}_e^{1.89})$ where P_{rad} is the total radiation power, Z is the charge of the primary impurity, S is the plasma surface area, and $\bar{n}_e$ is the line-averaged density. The horizontal error bars depicted here represent the uncertainty related to the impurity charge to use in the scaling expression [ranging from Z = 6 (carbon) to Z = 18 (argon)]. The ITER data point is calculated using P_{rad} = 150 MW, S = 1250 m^2, $n_e = 0.96\times10^{20}\ m^{-3}$, and Z = 18 (argon).

the measured $D_\alpha/D_\beta > 20$ in the vicinity of the OSP, indicating that significant recombination is not occurring [10].

The core carbon fraction does not change during the argon injection phase while the argon fraction increases to ~0.20%, resulting in $Z_{eff} \approx 1.85$. The incremental Z_{eff} (i.e., $Z_{eff} - 1 = 0.85$) in this case is a factor of 2 smaller than that predicted by the scaling for radiative plasmas given by Matthews et al. [11]. In fact, the measured incremental Z_{eff} in these "puff and pump" discharges is found to be consistently lower than the predicted scaling value, regardless of the level of argon (Fig. 2.4.3–3). This is consistent with the fact that the radiation in the DIII–D radiative discharges comes primarily from the divertor plasma; therefore, one might expect that P_{rad} is not a linear function of Z_{eff} as suggested by the Matthews scaling law. Global energy confinement is not affected by the introduction of argon with $\tau_E = 1.7\ \tau_{E,ITER89P} = 1.1\ \tau_{E,ITER93H}$. In fact, there is little variation of energy confinement in these types of discharges over a wide range of radiative fraction (P_{rad}/P_{NBI}). Furthermore, local transport analysis reveals little difference in χ_{eff} between the D_2 only cases and the D_2+Ar cases. These observations are consistent with the observation that there is little deterioration in the edge plasma during the argon injection phase as the pedestal pressure remains approximately the same as in the pre-argon phase. ELMs remain Type I in character with the ELM frequency increasing to 200 Hz during the argon injection phase. One final observation is that energy confinement in this regime is insensitive to plasma density, in contrast to results in RI–mode studies on TEXTOR [12].

Conclusions. These studies have demonstrated that SOL flow has many inherent benefits for ELMing H–mode operation including: (1) improved impurity enrichment, especially of high Z impurities; (2) thicker SOL profiles at the midplane; and (3) a factor of two reduction in the heat loss per ELM without any deleterious effects on plasma confinement. Furthermore, a radiative plasma which meets all of the relevant criteria embodied in the ITER design has been produced on DIII–D using this technique in combination with argon injection. The core argon fraction in these plasmas (~0.20%) is consistent with the maximum fraction allowed in ITER as estimated by computational simulations [13]. Previous analytic estimates show that the maximum heat flux reduction in ITER that could be expected given this argon fraction would be ≈50% [14]. However, assuming that an argon enrichment consistent with this experiment (i.e., η_{exh} = 3.0) can be achieved on ITER, then this percentage would increase to over 67%, consistent with the ITER criteria that P_{target}/P_{loss} = 33%. The results described in Section 2 indicate that even higher divertor enrichment of argon ($\eta_{exh} \approx 20$) can be obtained using induced SOL flow and that a more closed, baffled divertor geometry may increase the argon enrichment further. Finally, it is worth noting that the "solution" described here embodies many of the favorable aspects of a "hybrid" radiative solution in which low-Z impurity radiation predominates in the divertor and radiation from a high-Z impurity embellishes this divertor radiation while providing additional radiation in the core plasma.

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2.4.4. PLASMA FLOW IN THE DIII–D DIVERTOR[13]

Indications that flows in the divertor can exhibit complex behavior have been obtained from 2-D modeling [1,2] but so far remain mostly unconfirmed by experiment. An important feature of flow physics is that of flow reversal. Flow reversal has been predicted analytically [3] and it is expected when the ionization source arising from neutral or impurity ionization in the divertor region is large, creating a high pressure zone. Plasma flows arise to equilibrate the pressure.

A radiative divertor regime has been proposed in order to reduce the heat and particle fluxes to the divertor target plates. In this regime, the energy and momentum of the plasma are dissipated into neutral gas introduced in the divertor region, cooling the plasma by collisional, radiative and other atomic processes so that the plasma becomes detached from the target plates. These regimes have been the subject of extensive studies in DIII–D [4] to evaluate their energy and particle transport properties, but only recently it has been proposed that the energy transport over large regions of the divertor must be dominated by convection [5] instead of conduction. It is therefore important to understand the role of the plasma conditions and geometry on determining the region of convection-dominated plasma in order to properly control the heat and particle fluxes to the target plates and hence, divertor performance.

Owing to increased awareness of the important role of flows in the divertor, efforts are being made to characterize plasma and impurity flows in the divertor region. Divertor spectroscopy has been used to study impurity flows in ASDEX-Upgrade [6] and DIII–D [7] and probes for background plasma flow in DIII–D [8], Alcator C-Mod [9], TdV [10], and ASDEX-Upgrade [11], yet results are still partial and preliminary within a growing body [12,13] of well documented divertor physics.

Results and Discussion. We have measured the Mach number of the background plasma ion (D^+), in the DIII-D tokamak divertor, by using a fast scanning probe which is introduced vertically from the floor as shown in Fig. 2.4.4–1(a). The experiments were performed in lower single null divertor configuration discharges with plasma current I_p=1.4 MA, toroidal field B_T=2 T (∇B drift towards lower divertor), flattop duration of 3.8 s. and chord-averaged density of 0.5–1.0×10^{20} m^{-3}. The discharges are heated primarily by neutral beam injection at power levels of 4–5 MW. If a strong gas puff is introduced during the discharge, the divertor plasma temperature drops and the density increases as the plasma detaches from the target plate. We have studied attached and detached discharges in order to compare the flow patterns and their role in particle and energy transport. Since the fast probe is fixed in space, the divertor plasma has been scanned horizontally (in R), to allow exploration of various regions of the divertor, as shown in the two vertical cuts in Fig. 2.4.4–1(a).

For attached divertor conditions, in H–mode, we observe plasma flow accelerating towards the plate in the lower divertor, in agreement with classical expectations [14], as shown in Fig. 2.4.4–1(b,c) (diamonds). As the neutral density in the divertor increases, and the temperature is reduced, a narrow region of flow reversal at ~1×10^6 cm s^{-1} develops at the separatrix, as shown in Fig. 2.4.4–1(d,e) and extends further to the upper divertor and private region for very high neutral density conditions. The development of flow reversal [11] is strongly dependent on the particle source [3] and of great relevance for impurity transport since the impurities can then easily escape the lower divertor [15], defeating its purpose of impurity control. Impurity flow reversal (CIII) has been also observed [7] in the DIII–D divertor near the

[13]Boedo, J.A., G.D. Porter, M.J. Schaffer, et al., "Plasma Flow in the DIII–D Divertor," Proc. 25th European Conf. on Controlled Fusion and Plasma Physics, June 29–July 3, 1998, Zofin, Praha, Czech Republic, Vol. 22C, p. 822 (European Physical Society, 1998); General Atomics Report GA–A22915 (1998).

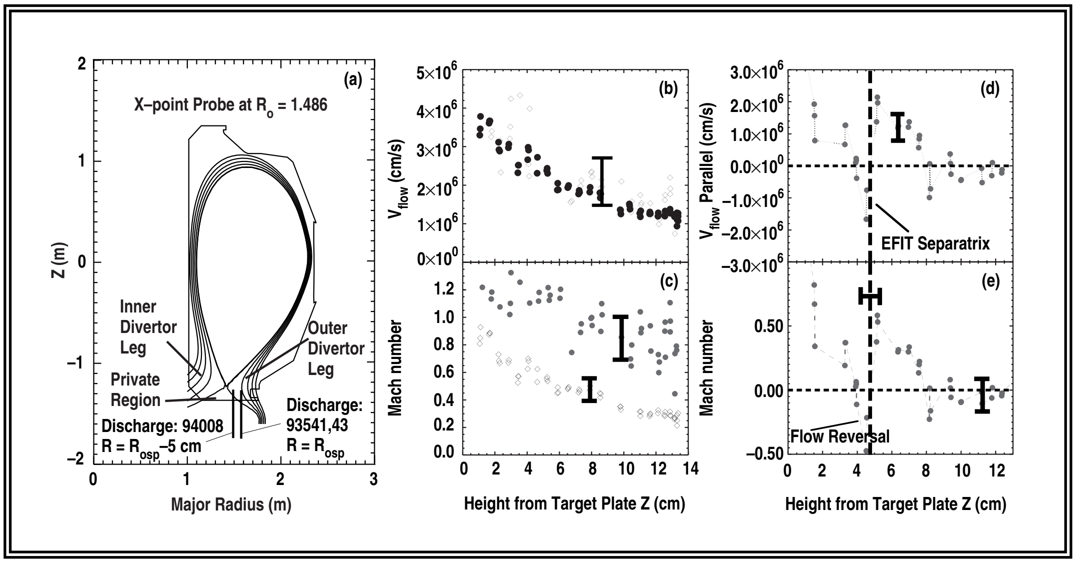

Fig. 2.4.4–1. The plasma and divertor geometry and the probe paths are shown. The path labeled R_{osp}–5 cm crosses the separatrix and the path labeled R_{osp} enters the SOL directly (a) Measurements of ion velocity (b) and Mach number (c) are obtained by the probe for attached (diamonds) and detached (circles) discharges. We also show the measurements of flow velocity (d) and Mach number (e) obtained at the path crossing the separatrix, and showing flow reversal.

separatrix and in the upper divertor at speed comparable to that of the background plasma. This measurement confirms previous observations of impurity transport in ASDEX-U [6].

As the neutral density in the divertor is increased, the temperature in the divertor decreases further to a regime where recombination starts playing a role and the plasma is detached from the divertor plates. For detached divertor conditions, in H–mode, the plasma flows towards the divertor plate at sound speed over an extended region comprising much of the SOL as shown in Fig. 2.4.4–1(b,c) (circles). Heat and particle transport under these conditions are then dominated by convection [5]. By comparing the parallel convected heat flux inferred from probe data [16–18] to the total heat flux at the plate measured by an IR camera [18], we find that 80% of the heat flux can be accounted for in semi-detached plasmas and 20%–30% in attached [19] plasmas.

We have modeled all the aforementioned discharge conditions with the code UEDGE [20] in 2 dimensions as shown in Fig. 2.4.4–2(a). We can reproduce the main features observed [19]: 1) flow reversal, as shown in Fig. 2.4.4–2(b), and to be compared to Fig. 2.4.4–1(d,e), and 2) accelerated flow towards the plate as shown in Fig. 2.4.4–2(c), and to be compared to Fig. 2.4.4–1(b,c) (diamonds). We can also reproduce results of convective flow over large volumes of the divertor as shown in Fig. 2.4.4–1(b,c) (circles). The latter figures are cuts on a 2-D plot, typical UEDGE output, such as the one shown in Fig. 2.4.4–2(a) and meant to reproduce the probe trajectories shown in Fig. 2.4.4–1(a). Work is in progress to improve the accuracy of the simulations by tuning the physics of the carbon source and calculating self-consistently the electric fields (and thus drifts) in the divertor region.

The relevance of the electric fields in the divertor region is under scrutiny since electric fields as large as 100 V/cm have been observed by the scanning probe across the separatrix as shown in Fig. 2.4.4–3. These fields are located at the boundary between the private region and the SOL and can produce

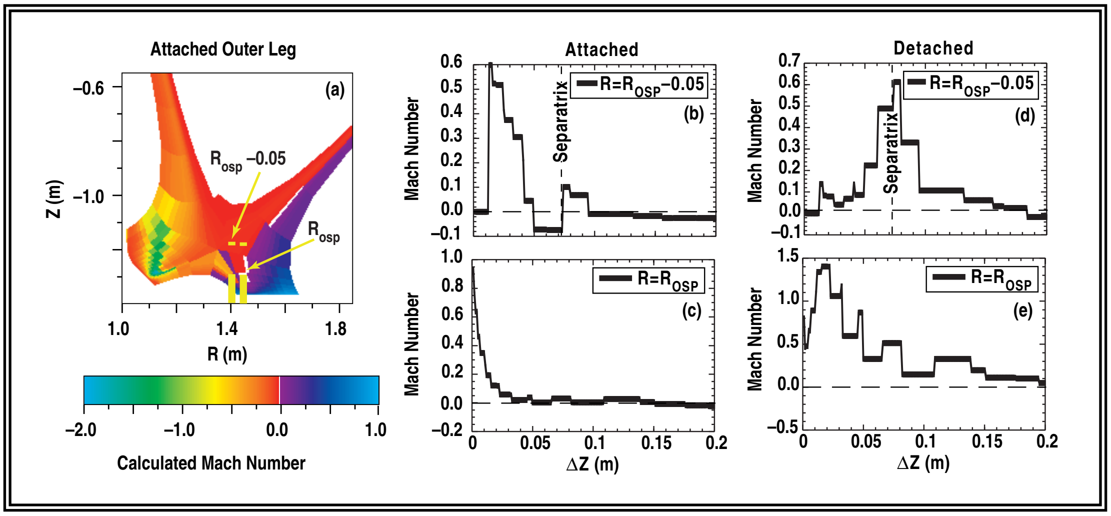

Fig. 2.4.4–2. The 2-D calculations of Mach number obtained by UEDGE and two paths meant to simulate those taken by the probe are shown (a). Calculated Mach number for cuts along those paths crossing the separatrix (b) and entering the SOL directly (c) are shown for attached discharges. Calculated Mach number for the same paths (d–e) are also shown.

$\vec{E}_\psi \times \vec{B}_\phi$ flows of the order of 0.3–1×10^5 cm/s away or towards the target plate, introducing a significant amount of poloidal velocity shear (~0.5–1×10^5 s^{-1}). The poloidal flow speed is between 10% and 40% of the parallel flow and thus can affect its direction appreciably. The particle flux induced by the poloidal flows is of the order of 2–4×10^{22} m^{-2}s^{-1} which is larger than the estimated radial flux (5×10^{20} m^{-2} s^{-1}) inferred from UEDGE or from turbulence measurements at the midplane. The $\vec{E}_\psi \times \vec{B}_\phi$ particle flows thus can potentially affect particle balance in the divertor considerably.

Conclusions. We have observed complex structures in the deuterium ion flows in the DIII–D divertor. Features observed include reverse flow, convective flow over a large volume of the divertor and stagnant flow. We have measured large gradients in the plasma potential across the separatrix in the divertor and determined that these gradients induce poloidal flows that can potentially affect the particle balance in the divertor. Introduction of self-consistent electric fields in UEDGE is in progress.

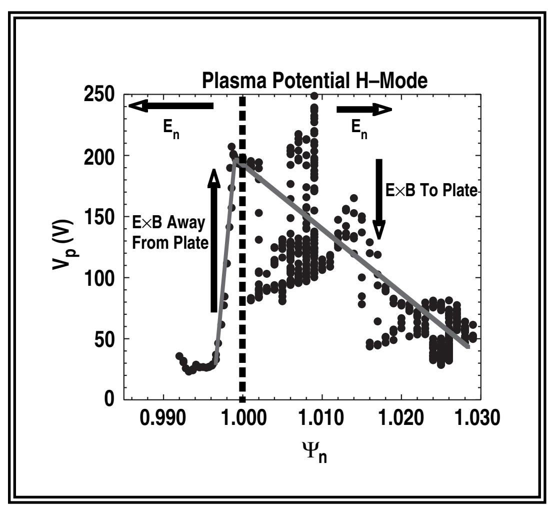

Fig. 2.4.4–3. Plasma potential profile (V) plotted versus normalized flux ψ_n. The separatrix is at ψ_n=1. The electric field is of the order of 200 V/cm.

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2.4.5. INVESTIGATION OF DENSITY LIMIT PROCESSES IN DIII–D[14]

Density limit studies and projections are crucial in the design of a fusion reactor's basic operational regime. The operating plasma temperature is set by reactivity considerations. Thus the plasma density, in particular the density in the high reactivity region, determines the fusion power production for an optimized plasma temperature. In present day tokamaks in the high confinement mode (H–mode), the density profiles are relatively flat and it is difficult to create a peaked density profile with H–mode level energy confinement. This flatness of the profile has an undesirable consequence: achievement of a high density core requires a high density edge plasma. However, it is precisely the edge region which can impose the lowest absolute density limit. Thus achievement of a high central density can be restricted by edge density limits.

Historically density limits have been measured and cited as limits on the line-average density ($\bar{n}_e$) measured with an interferometer chord, preventing conclusive identification of the underlying mechanisms. Studies [1,2] of multi-machine databases have identified a commonly observed density limit scaling: $\bar{n}_e \propto I_p/a^2$, where I_p is the plasma current and a is the minor radius. However extrapolation of this scaling to reactors can be misleading because the underlying processes have not been definitively determined. The scaling has consequences for fusion reactors: many D-T reactor designs must operate above this limit for economic competitiveness. Theories indicate that several distinct processes exist which can limit density in either the core, edge, or divertor plasma. Motivated by the International Thermonuclear Experimental Reactor's (ITER) need [3] to operate with density above the Greenwald limit (n_{GW}) [2] with H–mode energy confinement, a multi-year experimental campaign has been carried out in DIII–D to identify density-limiting processes and determine techniques to avoid them [4]. Here density "limit" is used loosely since we include processes which prevent attainment of high density operation *with* high energy confinement, as opposed to exclusively disruptive processes. One of our primary goals was to separate edge and core density limit mechanisms.

A density limit in H–mode discharges is most easily observed in a density ramp with external gas fueling. In DIII–D, the following time sequence is usually observed as $\bar{n}_e$ is increased:

1. The divertor plasma partially detaches [5] from the outer target plate upon crossing a private flux region neutral pressure limit [6] and a divertor "MARFE" forms on the low-field side of the X–point region
2. The divertor plasma completely detaches, i.e. the divertor D_α and target particle flux are reduced to ~ 0
3. The divertor "MARFE" begins to migrate to closed field lines
4. An H–mode-to-L–mode confinement transition is observed ELM activity ceases, and the MARFE encroaches onto closed field lines in the X–point area

This entire sequence was eliminated [4] by active pumping of the divertor with the in-vessel cryopump. Pumping maintained the private region neutral pressure below the critical value [6] for partial detachment onset. Pellet fueling replaced gas fueling to raise the plasma density while maintaining low divertor neutral pressure.

[14]Maingi, R., M.A. Mahdavi, T.W. Petrie, et al., "Investigation of Density Limit Processes in DIII–D," presented at 17th IAEA Fusion Energy Conference, October 19–24, 1998, Yokohama, Japan, to be published as General Atomics Report GA–A23000.

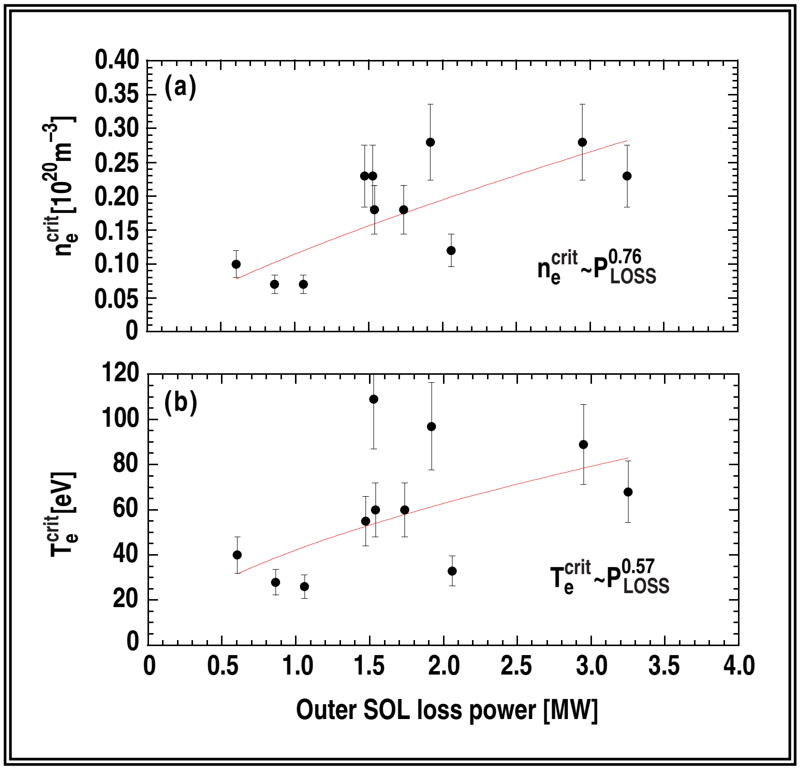

Fig. 2.4.5–1. Scaling of (a) critical separatrix density and (b) temperature at detachment onset as a function of the loss power to the outer divertor.

Partial detachment [5] is in fact an attractive reactor operating scenario because of reduced heat flux without deleterious confinement effects. Borass' 2-point SOL and divertor plasma model has been benchmarked [7] with experimental data from existing tokamaks and used to predict conditions required for detachment onset for ITER. This model was applied [8] to DIII–D conditions and predicted a power scaling of the critical upstream separatrix density and temperature (n_{upst}^{crit} and T_{upst}^{crit}) for partial detachment onset: $n_{upst}^{crit} \propto P_{div,outer}^{0.7}$, $T_{upst}^{crit} \propto P_{div,outer}^{0.3}$, where $P_{div,outer}$ is the power flow into the outer divertor. The upstream parameters when the divertor temperature, T_{div}, falls below 5 eV were taken as the partial detachment onset point in the modeling. Figure 2.4.5–1 shows that experimentally $n_{upst}^{crit} \propto P_{div,outer}^{0.76}$, $T_{upst}^{crit} \propto P_{div,outer}^{0.57}$, from Thomson Scattering measurements of electron parameters just above the outer midplane at partial detachment onset. Thus the predicted density scaling appears to fit our data but the temperature dependence is stronger than predicted by the model. Also, the absolute upstream separatrix density predicted by the model is about 2× higher than our measured values — more work is required to understand this particular discrepancy. This study demonstrates the importance of local parameter analysis for these processes: we have previously reported [9] that the critical $\bar{n}_e$ for detachment onset is almost insensitive to the global heating power, i.e., $\bar{n}_e^{crit} \propto P_{heat}^{0.15}$. In fact that same dependence is present in these data but the global analysis masks the important dependencies. The reason is quite simple: gas puffing preferentially raises the SOL density relative to $\bar{n}_e$. Thus while the changes in the $\bar{n}_e$ are small during these scans, the SOL density increases much more rapidly.

The high density H–L back transition is often accompanied [4,10,11] by MARFEs on closed field lines just inside of the X–point region. In DIII–D this transition is usually observed at between 0.7–1.0*Greenwald scaling. We have formulated the MARFE onset criterion for DIII–D discharge parameters and shown [12] that a discharge with a MARFE was predicted to be unstable to MARFE formation, whereas a discharge without a MARFE was predicted to be stable. In the low edge temperature region near the separatrix for T_e<75 eV, the MARFE onset critical density increases as the edge T_e^4, primarily because of the $T_e^{7/2}$ dependence of the parallel heat flux which can stabilize the radiative cooling mechanism responsible for the MARFE. When combined with the ITER-89P scaling law for L–mode energy confinement, it can be shown* [12] that the MARFE onset critical upstream electron density (n_e^{crit}) has the following scaling:

$$n_e^{crit} \propto \frac{I_p^{0.96}}{a^{1.9}} \xi^{-0.11} P_{heat}^{0.43} R_m^{0.17} B_t^{0.04} \left[\kappa^2\left(1+\kappa^2\right)\right]^{-0.22} \quad , \tag{1}$$

where ξ is impurity concentration, P_{heat} is the heating power, R_m is the major radius, B_t is the toroidal field, and κ is the elongation. This scaling is remarkably similar to Greenwald, both in the strong I_p and *a* dependencies but also in the weak dependencies on other quantities (except P_{heat}). Thus the scaling is clearly applicable to L–mode plasmas which usually have T_e<100 eV at the edge. H–mode confinement scaling laws have similar dependencies on engineering parameters, leading to a similar scaling for high density ELMY H–mode plasmas in DIII–D. Note that the scaling above predicts a stronger dependence on P_{heat} than observed on present day machines — we speculate that this discrepancy is related to global parameter analysis presented in most density limit studies and global confinement scaling used to arrive at Eq. (1). This scaling yields an edge plasma density limit which should be of no concern for reactors such as ITER which are expected to operate at edge temperatures well above 100 eV.

After eliminating the detachment and MARFE sequence with divertor pumping, pellet injection was used to fuel the core. However, pellet fueling characteristics lead to other restrictions [4] in the high-density operational window. We observed a stronger than linear plasma current dependence of the density decay time following pellet injection, which suggested operation at high I_p was favorable for increasing density. In contrast to Greenwald's original analysis [2], we found no correlation between the density decay time and $\bar{n}_e/n_{GW}$. In addition, pellet fueling efficiency was found to decrease with heating power, suggesting low heating power was favorable. At B_t =2.15 T the heating power was ≤2 × L-H confinement transition power threshold. In this regime pellets produced H-L transitions which rapidly ejected the pellet density in <10 ms. Access to high density was achieved by operating at low B_t, giving more margin over the L-H threshold. We found that MHD modes could be de-stabilized at densities as low as $\bar{n}_e/n_{GW} \sim 0.8$ during pellet fueling. The n=2 modes resulted in tolerable particle and energy confinement degradation (10%–15%), but the n=1 modes were catastrophic. The cause for the onset of these MHD modes is unclear, but the n=1 modes were avoided by operating at $P_{heat} < 3$ MW ($\beta_N \leq 1.7$).

By studying each of the aforementioned physical processes and selecting conditions to avoid them, we have achieved discharges (e.g. Fig. 2.4.5–2) for the first time at $\bar{n}_e/n_{GW} \geq 1.5$ for up to 600 ms, with a peak energy confinement time of ~1.2*ITER93H scaling. Due to the operating conditions required [4], these discharges were ELM-free and suffered from core impurity accumulation and a central power balance limit. Our effectiveness in heating the center was limited by the neutral beam technique; the heating deposition became peaked well off-axis during the high density phase, leading to a central radiative loss rate 4–5 times larger than the neutral beam heating rate. The central power balance density limit has been shown to be very high for reactors in which the heating profile will always be peaked on-axis.

In summary, the various edge density limits we have studied all extrapolate favorably for ITER. By avoiding the pellet fueling, confinement, and MHD limits, we were able to successfully achieve the central core radiative collapse density limit. This limit has been shown to be very high for ITER. The focus of present studies is on understanding the various pellet-fueling related limits. In addition, injection from the high-field side has been shown [13] to increase fueling efficiency at high heating power which should allow us to raise the maximum $\beta_N > 2$, ITER's high value. This capability has been installed into DIII–D and will be used in upcoming experiments.

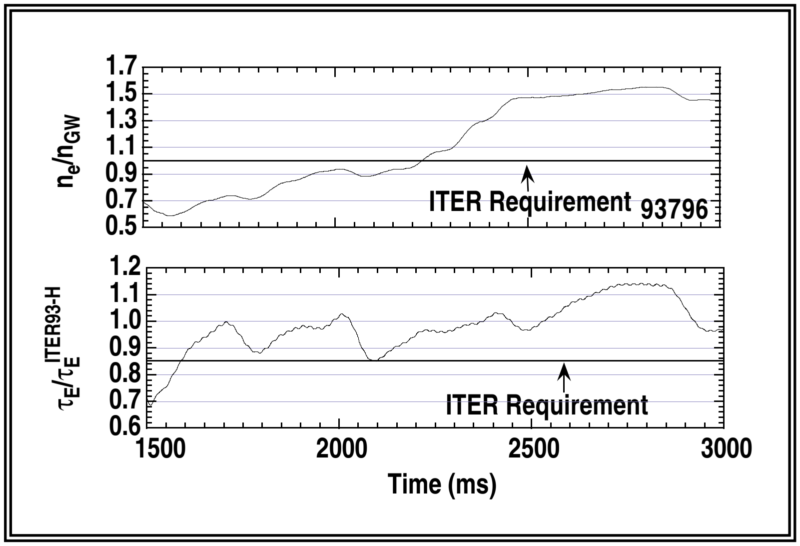

Fig. 2.4.5–2. Demonstration discharge with density ≥1.5× Greenwald scaling and H–mode confinement.

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*Note that Section 2.4.5 erroneously reported $n_e^{crit} \propto P_{heat}^{0.17}$; the correct relation is $n_e^{crit} \propto P_{heat}^{0.43}$.

2.5. WAVE PARTICLE INTERACTION PHYSICS

2.5.1. CURRENT PROFILE MODIFICATION WITH ELECTRON CYCLOTRON CURRENT DRIVE IN THE DIII–D TOKAMAK[15]

Control of the plasma current profile is necessary to extend the high performance discharges observed on the DIII–D tokamak and other tokamaks to steady state. Beyond the obvious need to maintain the total plasma current non-inductively, both the stability of the plasma and the transport of energy across the magnetic field depend on the current profile. Electron cyclotron current drive (ECCD) is a leading candidate to fulfill the role of plasma current profile control due to the straightforward ability to control the location and the magnitude of the non-inductive current under a wide variety of conditions, and the absence of the technical complication of plasma-antenna interactions.

The present system for ECCD on the DIII–D tokamak consists of two gyrotrons operating at 110 GHz. The system and verification of its proper operation have been described in detail elsewhere [1,2], so only a brief description will be given here. The two gyrotrons are rated for 0.9 MW for 2 s and 0.8 MW for 1 s, respectively. The pulse lengths are currently limited by heating of the gyrotron output window, but are adequate for the present proof-of-principle experiments. The power is transmitted via evacuated corrugated waveguide (31.75 mm diam) to the tokamak. Each transmission line contains a pair of miter bends which use grooved mirrors to set almost any desired polarization. Two separate launcher assemblies, neither of which have a vacuum window, have copper mirrors which can steer the beam poloidally. The toroidal angle is fixed in each launcher — either for co-current drive (ϕ = 24°–31° depending on the poloidal angle) or for nearly radial launch, which allows heating without current drive. All of the results reported here use the co-current drive launcher. A vacuum opening of the transmission line is required to switch between launchers, so comparison of co-current drive with pure heating is not possible in a single day. The experiments reported here all employ second harmonic absorption of the extraordinary-mode polarization. Polarization purity and deposition location experiments have been successfully carried out [1,2]. As a whole, the ECCD system has a demonstrated reliability comparable to the neutral beam systems on DIII–D.

The results reported here represent the proof-of-principle phase of a program to implement an active current profile control system on the DIII–D tokamak. Three key elements of the proof-of-principle are presented here. First, the ability to modify the current profile by varying the deposition location is demonstrated by changes to global quantities related to the current profile such as the internal inductance (ℓ_i) and the appearance of MHD instabilities identified with the q=1 surface such as sawteeth or m=1/n=1 modes. Second, localized current drive is measured by means of an analysis technique which makes use of the unique diagnostic capabilities of the DIII–D tokamak. While central current drive has been previously measured on DIII–D, the first quantitative measurement of localized off-axis ECCD in any toroidal device is reported here. Third, these current drive measurements are compared with various theoretical calculations in order to validate a predictive model of ECCD.

The discharges for the study of ECCD utilize early neutral beam injection (NBI) to delay the onset of sawteeth by raising the electron temperature (T_e) and to allow continuous measurement of the internal magnetic fields by means of motional Stark effect (MSE) spectroscopy [3]. This enables detailed reconstruction of the magnetic equilibrium with the EFIT code [4]. An example of the effect of central ECCD is shown in Fig. 2.5.1–1. A single gyrotron is pulsed on at 1.5 s. The time sequence of equilibria show

[15]Luce, T.C., Y.R. Lin-Liu, J.M. Lohr, et al., "Current Profile Modification with Electron Cyclotron Current Drive in the DIII–D Tokamak," presented at 17th IAEA Fusion Energy Conference, October 19–24, 1998, Yokohama, Japan, to be published in a special issue of Nucl. Fusion; General Atomics Report GA–A23002 (1998).

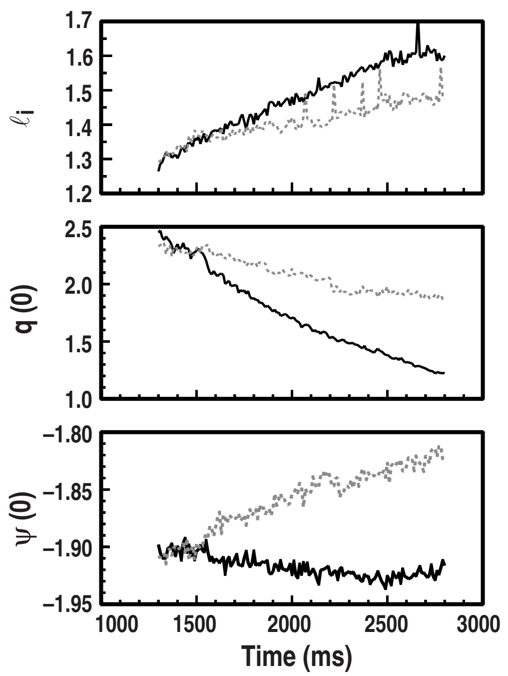

Fig. 2.5.1–1. Time histories of the internal inductance ℓ_i, the central safety factor q(0), and the poloidal flux at the magnetic axis ψ(0) for discharges with ECCD (solid line) and without (dashed line). A single gyrotron delivers ~0.5 MW starting at 1.5 s.

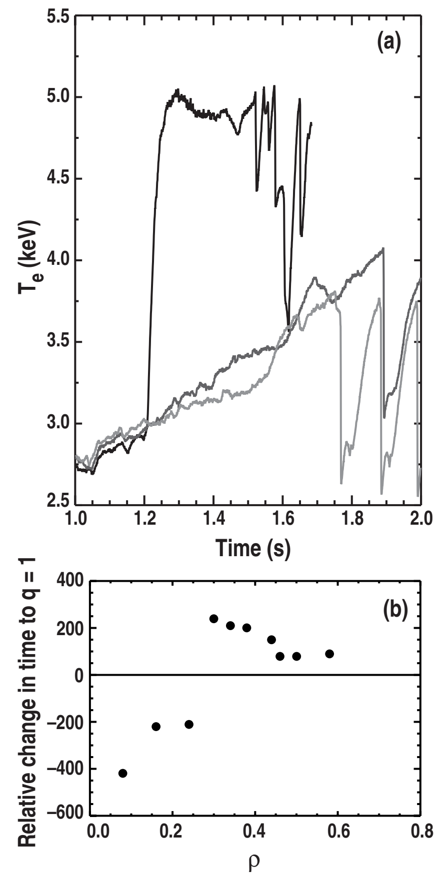

Fig. 2.5.1–2. (a) Time histories of central electron temperature. The first sharp drop in temperature in each trace signifies the time of the first sawtooth. The central ECCD case has the first sawtooth at 1.52 s while the off-axis case has the first sawtooth at 1.89 s. The NBI-only fiducial case has the first sawtooth at 1.74 s. Note that the sawteeth are also much more rapid in the central ECCD case. (b) A dataset of similar discharges (same q, density, and ECH power) showing the effect illustrated in (a) is systematic. Current drive inside of ρ = 0.3 hastens the onset of sawteeth while current drive for ρ > 0.3 delays the onset of sawteeth as discussed in the text.

that the ℓ_i and central safety factor q(0) deviate significantly from the NBI-only fiducial. While central heating should eventually have a similar result, the time scale for current penetration should be longer, not shorter, if this were a pure heating effect. The time history of the central poloidal flux ψ(0) also directly indicates that non-inductive current is the cause of the peaking. The time derivative of the poloidal flux is the local loop voltage, and the case with ECCD has a negative voltage indicating a non-inductive current source greater than the total existing current density on axis.

Another indication of current profile modification is the timing of the appearance of a q=1 surface in the plasma as evidenced by the onset of sawteeth or m=1/n=1 modes. For current drive on-axis, the q=1 surface should appear more quickly than in the fiducial case since current is being supplied more rapidly than is possible by diffusion. In the case of weak off-axis current drive, the current profile is broadened and less inductive flux is required which together delay the appearance of the q=1 surface. Evidence of these effects is shown in Fig. 2.5.1–2. The top part of the figure shows time histories of central T_e for three cases: NBI-only, ECCD at ρ = 0.2, and ECCD at ρ = 0.45. (The coordinate ρ is the square root of the normalized toroidal flux normalized to the edge value which acts as a relative radial coordinate.) In the off-axis case, the first sawtooth crash is delayed by more than 0.1 s, while central case induces the first sawteeth over 0.2 s earlier, despite the significant rise in T_e. The lower part of the figure shows the change in the time at which a q=1 surface appears, relative to a NBI-only fiducial with equal plasma current (I) and toroidal field (B), as a function of deposition radius. The dataset consists of ten discharges with equal injection power (P_{EC}) and electron density (n). The systematic trend discussed above is apparent and the magnitude of the effect appears consistent with the magnitude of the driven current, which drops with radius due to the lower local T_e. The conclusion is that the ECCD is capable of making measurable modifications of the current profile over the range of radius where experiments were carried out (ρ = 0.1–0.6). This sets the stage for the next step, which is to quantify the location and magnitude of the ECCD.

Measurement of the non-inductive current profile in the absence of resistive equilibrium requires simultaneous knowledge of the current density profile and the internal electric potential or loop voltage (V). These quantities are inferred on DIII–D by calculating the poloidal flux ψ on a spatial grid as a function of time [5]. The high spatial resolution measurements of the internal magnetic fields by MSE are necessary to provide the required accuracy and resolution. Two spatial derivatives of ψ give the total current density while the time derivative of ψ at constant ρ gives the loop voltage at that surface. Using a neoclassical conductivity [6], the non-inductive current density (J_{NI}) is given by the difference of the total current density ($J_{||}$) and the inductive current density ($J_{OH} = \sigma E_{||}$). The remaining JNI is a combination of NB, bootstrap, and EC current. To isolate the ECCD, an NBI-only fiducial is prepared identically to the ECCD shot, and the difference is formed. This difference is corrected for the change in kinetic parameters between the two shots, but that correction is usually small. The assumptions of neoclassical resistivity and bootstrap current have been validated experimentally [5].

Two examples of off-axis ECCD analyzed by this technique are shown in Fig. 2.5.1–3. The left-hand column shows a case with deposition at ρ = 0.2 and the right-hand column shows a case with ρ = 0.45. Starting with the left-hand case, the top figure shows $J_{||}$ from the equilibrium reconstruction for the ECCD and NBI-only cases. Notice that in the 0.5 s since the turn-on of the ECCD, the current profile has been substantially modified in agreement with the discussion above of Fig. 2.5.1–2. This magnitude of change is consistent with resistive simulations. The next box down shows the inferred loop voltage as a function of radius for both cases. The error bars are the random errors arising from fitting the time series of equilibria. This is estimated to be the dominant source of random error in this calculation and is propagated throughout the remaining calculations. As explained above, the neoclassical conductivity is calculated from the measured n, T_e, and impurity concentration (Z_{eff}) and combined with V and $J_{||}$ to give J_{NI} (third box). (The graph ends at ρ = 0.7 because no Z_{eff} measurements are available outside of this.) Finally, the difference in the non-inductive current between the ECCD and fiducial shot is shown in the bottom box. This difference is ascribed to ECCD. The integrated difference current out to ρ = 0.4 is 48 kA. The apparent current for $\rho > 0.4$ is ~10 kA and is likely due to the accumulation of the systematic errors of this technique. While the accumulated random error in the driven current is large (~34 kA), making definitive comparisons at that level difficult, the peak current density is >2 standard deviations (2 σ) from 0, and the peak is clearly resolved to better than 1 σ.

The right-hand column represents a case with the beam steered to ρ = 0.45. In this case, no significant change in $J_{||}$ is observed after 0.5 s (top box), but the ECCD is revealed by the reduction in V required

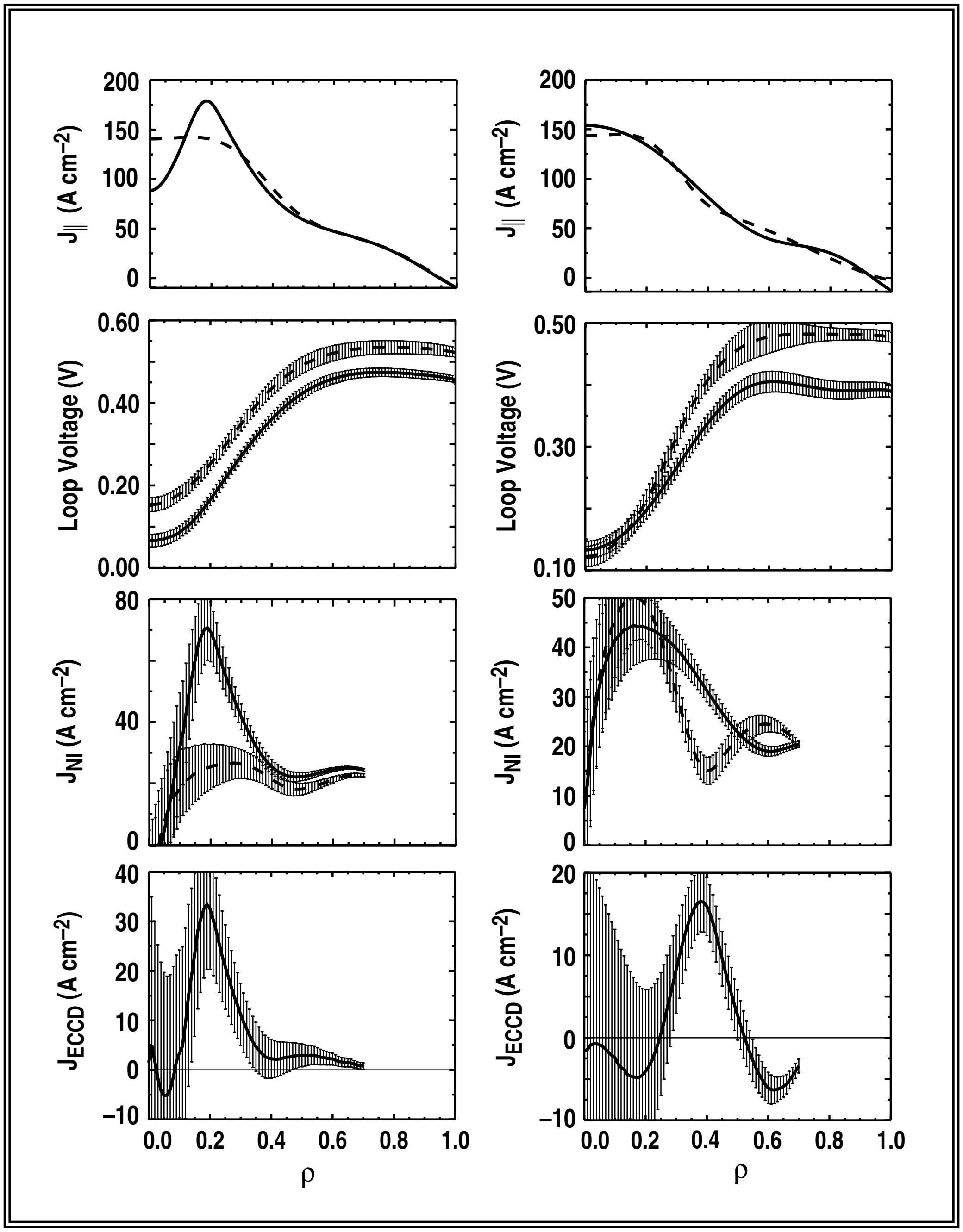

Fig. 2.5.1–3. Measurements of the current density due to the ECCD. The left column is analysis of a case with the beam aimed at ρ = 0.2. The plasma parameters in the ECCD discharge are B = 1.97 T, I = 0.98 MA, $\bar{n}$ = 1.7×10^{13} cm^{-3}, P_{EC} = 1.03 MW. The right column is analysis of a case with the beam aimed at ρ = 0.4. The plasma parameters in the ECCD discharge are B = 1.76 T, I = 0.89 MA, $\bar{n}$ = 1.8×10^{13} cm^{-3}, P_{EC} = 1.14 MW. The solid lines are the ECCD discharge in each case and the dashed lines are the NBI-only fiducial discharge. The top box is the total current density, the next box is the loop voltage, the third box is the non-inductive current density, and the bottom box is the ECCD current density. All traces are plotted versus the radial coordinate ρ.

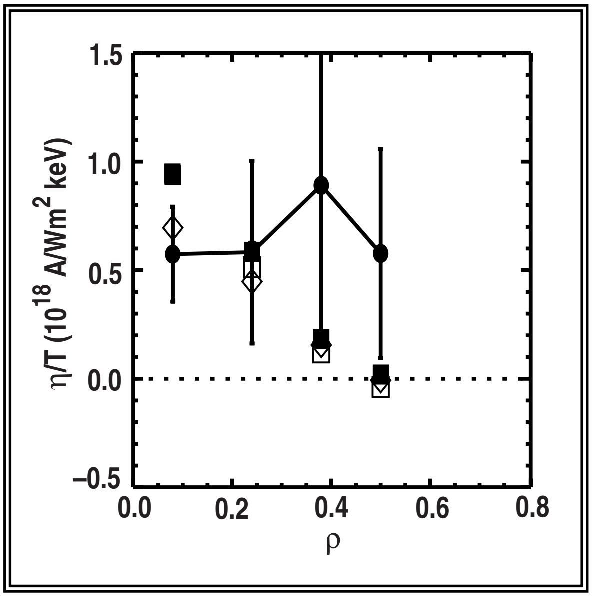

Fig. 2.5.1–4. Comparison of experimental and theoretical current drive efficiency normalized to temperature. The experimental efficiency (circles) is roughly independent of radius while the theoretical efficiencies drop sharply with increasing radius. Three theoretical calculations are shown: linear theory (diamond), quasi-linear Fokker-Planck calculation (open square), and Fokker-Planck with parallel electric field (filled square).

locally to drive the same current (second box). The difference in non-inductive current appears at the expected location and is reduced in magnitude from the $\rho = 0.2$ case (third and fourth boxes). The peak is resolved to 1 σ and is >2 σ from 0. The driven current in the positive peak is 31 kA. These cases are typical of the presently analyzed dataset in that the inferred peak current density appears at the expected location within the systematic errors of the aiming calibration, and the peak current density is significantly above any systematic or random errors apparent in the data.

The combination of poloidal beam steering and variation in the toroidal field allows assessment of the effects of trapped electrons on the ECCD. Two types of scans have been analyzed using the current drive analysis technique described above. The first is a scan of poloidal position at fixed toroidal field such that the resonance intersects the magnetic axis. The second type is a correlated variation of B and poloidal aiming to scan the poloidal deposition location at fixed ρ. In varying B, the plasma current is varied proportionately to keep the q profile similar, in order to avoid any difficulties with MHD instabilities. This dataset was obtained with roughly constant line-averaged electron density ($1.7–1.8\times10^{13}$ cm^{-3}) and P_{EC} (0.95–1.14 MW). The figure of merit chosen to evaluate these scans is the local current drive efficiency η ($\equiv nI_{EC}R/P_{EC}$, with n the density at the deposition location and the major radius of the center of the flux surface where the current is driven) normalized by the theoretically expected linear temperature dependence. The radial scan shown in Fig. 2.5.1–4 indicates that the normalized efficiency η/T is independent of ρ in the region where experiments were carried out ($\rho = 0.1–0.5$). Note that the driven current does drop over this range; it is only the normalized efficiency which is constant. This lack of dependence on ρ is in contrast to the theoretical results also illustrated in Fig. 2.5.1–4. Three types of calculations are displayed in the figure — a linear calculation [7], a quasi-linear Fokker-Planck calculation [8], and a quasi-linear Fokker-Planck calculation with the effects of $E_{||}$. The Fokker-Planck calculations have been verified with an independent code [9]. With the exception of the centermost case where the high power density results in a significant quasi-linear effect, the three calculations obtain roughly the same answers. Since the effective trapped particle fraction rises by approximately a factor of 2 over this range of ρ, it appears that the normalized efficiency does not depend as strongly on this quantity as predicted by theory.

This same conclusion is consistent with the results of the poloidal location scans shown in Fig. 2.5.1–5. At both $\rho = 0.35$ and $\rho = 0.45$, the normalized efficiencies are well above the theoretical predictions, indicating that the effect of trapped electrons is significantly less than predicted. The effect of trapped electrons is not completely absent as shown by points in the figure which represent calculations of the ECCD in the absence of trapped particles. The poloidal variation in both scans is due in part to the local trapped particle fraction change and in part due to the upshift of the toroidal index of refraction due to damping at small major radius. The relative importance of these two effects is being investigated.

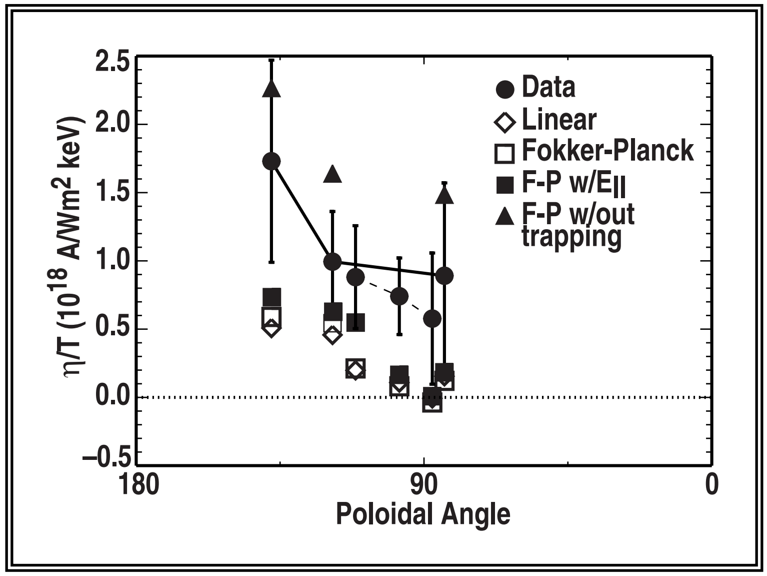

Fig. 2.5.1–5. Variation of normalized efficiency with poloidal angle at fixed ρ. The points connected by the solid line are ρ = 0.30–0.38. The points connected by the dashed line are for ρ = 0.44–0.50. Poloidal angle is defined as the angle with respect to the major radius with the magnetic axis as the origin. The outboard midplane is the 0° reference, vertically above the axis is 90°, and the inboard midplane is 180°.

One possible explanation for the weaker trapped electron effect is modification of the trapped particle boundary by finite collisionality. The theoretical calculations applied to the scans in Figs. 2.5.1–4 and 2.5.1–5 all impose a trapped particle boundary assuming zero collisionality, i.e., a boundary which continues down to zero velocity. In the trapped particle region of velocity space, the characteristic time is the bounce time, which is assumed in these calculations to be much shorter than the pitch-angle scattering time characteristic of the passing region. Therefore, electrons which diffuse into the trapped particle region from the co-current side emerge rapidly (compared to the pitch-angle scattering time) on the counter-current side. This is the Ohkawa effect [10]. Finite collisionality reduces the size of the trapped region roughly in proportion to $\sqrt{\nu_*}$, and reduces it preferentially at low velocity. This has the somewhat surprising result that finite collisionality increases the net current, because electrons spend more time on the co-current side of the distribution. The data in Fig. 2.5.1–5 lie between the calculations with zero collisionality and the calculations with no trapped particles, which indicates that a reduction of the trapped particle region can explain the data. A predictive model based on finite collisionality theory has not yet been developed.

A qualitative assessment of the implications of this effect for ECCD on DIII–D and future devices can be made. The discharges reported here have the same collisionality as envisioned for high fusion power devices such as ITER [11]. However, the plasma β is significantly lower than planned for these discharges. The effect of higher β is to move the wave-particle interaction to higher velocity where the finite collisionality effects are small and to higher parallel velocity (due to relativistic effects) where the distance to the trapped particle region is larger. Therefore, the observed enhancements in the DIII–D experiments should be less significant in next step devices, The same argument is true for the advanced tokamak discharge scenarios for DIII–D which have higher β and lower collisionality than these proof-of-principle discharges. Using the same zero collisionality calculations as referenced above, substantial off-axis current is predicted at the half radius in these scenarios, due to the higher β.

With the present 1 MW ECCD system on DIII–D, it has been possible to modify the current profile evolution and make clear measurements of localized off-axis ECCD. The measured off-axis current drive efficiencies are higher than predicted, indicating that some refinement of the theory is necessary. This work supports the development of the 6 MW ECCD system planned for completion in 2000 as an active current profile control tool for the DIII–D tokamak.

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2.5.2. FAST WAVE HEATING AND CURRENT DRIVE IN ELMING H–MODE PLASMAS IN DIII–D[16]

The DIII–D Fast Wave Current Drive (FWCD) program has concentrated on studying the basic physics of direct electron heating and current drive in low density L–mode discharges from its inception in 1991 through 1996. Among the reasons for this are the controllable low densities that can be achieved in L–mode, thus maximizing the power per particle available with a fixed transmitter power, and the high and nearly steady antenna loading that is obtained, maximizing the coupled rf power levels with a given maximum antenna voltage. Among the principal results of the L–mode experiments [1,2] was the observation of nearly full absorption of the coupled FW power, despite calculated first-pass absorption in the range of 8%–15%. However, the heating and current drive efficiencies dropped off sharply when the calculated first-pass absorption was less than about 8% in the condition studied. This was explained by an unspecified edge loss mechanism at the level of 4% per bounce. A survey of possible edge loss mechanisms in these experiments [3] showed that the most plausible candidate for the important edge loss mechanism was dissipation in far field rectified rf sheaths [4].

The more recent DIII–D FWCD work has been aimed at extension of the operating regimes of the experiments, including higher total power by combining FW with NBI [5] and 110 GHz ECH [6]. Generally, these higher power levels result in H–mode confinement, and in a quasi-steady-state condition, Edge Localized Modes (ELMs). Over the past two years, the DIII–D FWCD systems have been modified to improve their capabilities under a wider range of dynamic antenna loading conditions, such as those characteristic of ELMing H–modes. In this paper, the first results of extending the FWCD studies to ELMing H–mode discharges are presented.

Technical Improvements to DIII–D FWCD Systems. The DIII–D FWCD system consists of three four-element antenna arrays, of two different designs, and three transmitters, also of two different designs. The original system (referred to as 285, from the toroidal angle in the DIII–D vessel at which the antenna array is located) was operated at frequencies close to 60 MHz in these experiments, and the two newer systems (referred to as 0 and 180) were used at approximately 83 MHz. A considerable simplification of the 285 system's transmission line that was carried out in 1997 is described in detail in Ref. [7]. The resulting system has only one adjustable tuning element (the "decoupler" stub), yet the standing wave ratio seen by the transmitter is less than 1.25 at all times during the discharge despite rapid fluctuations in the antenna loading resistance of more than a factor of four in ELMing H–mode. One further refinement in the operation of 285 that was used in the 1998 experiments was to adjust the operating frequency, within the instantaneous bandwidth of the transmitter, to compensate for changes in the reactive component of the antenna loading caused by different antenna/plasma gaps. Since this system has no adjustable tuning elements to compensate for these changes, the frequency adjustment (in the range of 59.8–60.1 MHz) is necessary to minimize the power diverted to the dummy load.

The modifications to the two newer systems to enhance their H–mode capabilities were to the arc protection circuits, which were reconfigured to be functionally identical to the older system's. The resulting arc protection system distinguishes rapid load transients that appear symmetrically on all four elements of an array from localized changes in impedance that affect only one element at a time. Impedance changes of the symmetric type occur at L–H transitions and ELMs, or are caused by any other nearly axisymmetric fluctuation, while localized arcs in a single antenna in an array cause impedance changes of

[16]Pinsker, R.I., F.W. Baity, W.P. Cary, et al., "Fast Wave Heating and Current Drive in ELMing H–mode Plasmas in DIII–D," Proc. 25th European Conf. on Controlled Fusion and Plasma Physics, June 29–July 3, 1998, Zofin, Praha, Czech Republic, Vol. 22C, p. 1406 (European Physical Society, 1998); General Atomics Report GA–A22917 (1998).

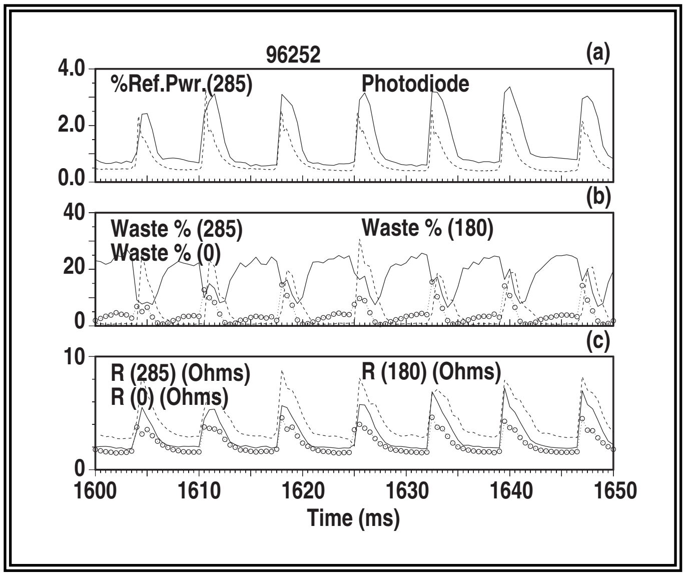

Fig. 2.5.2–1. (a) Time histories of the power reflection coefficient seen by 285 system's transmitter (solid line) and photodiode signal (dashed) in ELMing H–mode discharge. (b) Percentage of power diverted to dummy load for the 285 (solid), 180 (dashed), and 0 (symbols) systems. (c) Coupling resistance for the 285, 180 and 0 (similar labeling) antenna arrays.

the latter type. Furthermore, all three systems are equipped with "balanced fault detectors," which remove the rf drive when the antenna loading is anomalously high. The principle of this detector was based on the observation of a type of fault which appeared symmetric to all four array elements, but corresponded to a very high resistive loading, much higher than observed even at the peak of an ELM. This fault is interpreted as an rf discharge filling the antenna housing. Balanced faults were sometimes triggered by an ELM, particularly when the plasma surface was close to the antennas, and thereby constituted one limitation on the power that could be reliably coupled to ELMing H–mode discharges.

A 50 ms-long time history showing a typical example of operation of these systems in an ELMing H–mode is shown in Fig. 2.5.2–1. The coupling resistance of all three antenna arrays increases by a factor of between 2.5 and 4 at each ELM (indicate by the increase in the photodiode signal which measures D_α radiation at the plasma edge). Despite this substantial change in the antenna impedance, the power reflection coefficient seen at the transmitter output increases only to about 3%. The fixed pretuning arrangement used on the 285 array (no variable tuning elements other than the decoupler stub) causes the fraction of the transmitter power that is diverted to the dummy load (the waste percentage) to *decrease* at each ELM from about 25% to less than 10%. The tuners that remain in the 0 and 180 systems are adjusted so that the fraction of power diverted to their dummy loads is virtually 0 between ELMs, *increasing* to about 25% at the peak of each ELM. Which of these two configurations is more desirable is not obvious at present; comparison of these configurations is among the technical goals of these experiments. In 1998, the 285 system was reliably operated with a peak rf voltage limit of about 25 kV, while the 0 antenna operated successfully at peak voltages over 30 kV. Operation of the 180 system was limited not by peak antenna voltage, but by transmitter power, as a result of the higher ohmic losses in that system's relatively long transmission lines.

Electron Heating and Current Drive Experiments in ELMing H–mode Discharges. The first set of systematic experiments on FWCD and electron heating in ELMing H–mode plasmas were performed in the sawtooth-free portion (early neutral beam injection) of 1.4 MA discharges at a toroidal field of 2.05 T in a lower single null divertor configuration. 5 MW of NBI and up to 1.2 MW of 110 GHz ECH [6] were used to produce a target plasma with $T_e(0)$~4.5 keV at a line-averaged electron density of ~3.5×10^{19} m^{-3} with an ELM frequency of ~100 Hz. Discharges were studied with different "outer gaps" (the outer gap is the distance from the separatrix to the limiter at the outboard midplane; the face of the antenna Faraday shields is recessed about 3 cm behind the limiter) between 2.5 and 8 cm, with all other parameters held as constant as possible. The total fast wave power that could be coupled to these discharges was less than

2 MW, owing to the very light antenna loading obtained between ELMs — comparable to only twice the loading obtained in the absence of plasma (half of the net power is coupled to the plasma) at the larger outer gaps. The resistive loading decayed exponentially with the outer gap for all three arrays, as shown in Fig. 2.5.2–2(a). The rate of decay of the loading as the gap was increased was ~60% larger than had been observed in L–mode, which further exacerbated the difficulty of high power coupling at large gaps.

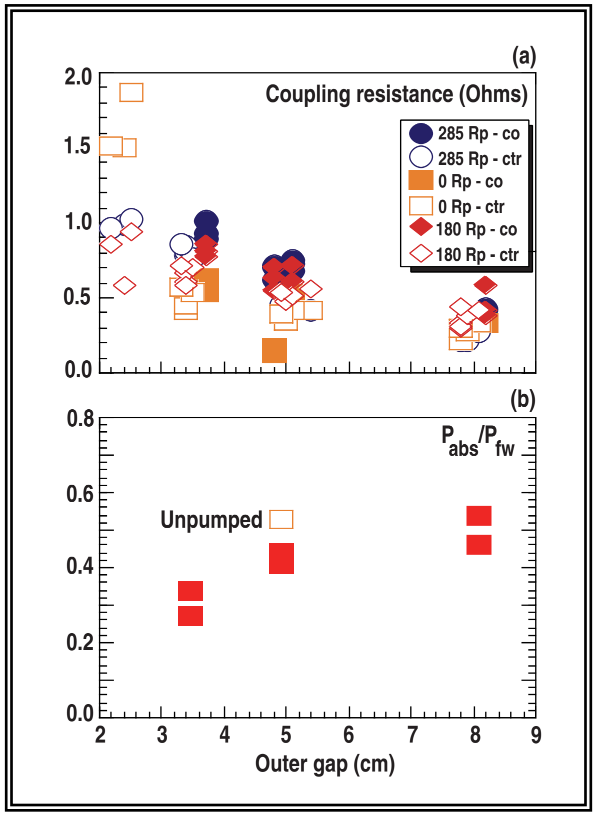

Fig. 2.5.2–2. (a) Coupling resistance for all three antenna arrays (between ELMs) in the outer gap scan. (b) Fraction of coupled fast wave power deposited within $\rho \leq 0.6$ from the ECE modulation analysis.

The FW power directly deposited on electrons was measured using standard Fourier techniques by modulating the FW power at 40 Hz (80% modulation depth, square wave envelope) and measuring the correlated response on the electron cyclotron emission (ECE) radiometer channels covering the core of the plasma ($\rho \leq 0.6$). In the case of the 8 cm outer gap, about half of the coupled FW power could be accounted for in this way, which is a fraction comparable to that obtained in previous studies under good absorption conditions. At smaller outer gaps, however, the fraction of the power appearing in the plasma core dropped substantially [Fig. 2.5.2–2(b)]. At an outer gap of 5 cm, the power that could be found in the modulation analysis was somewhat higher in a discharge in which the divertor cryopump was not used compared with a pumped case at the same outer gap. This gap dependence might be expected from a model in which the edge loss due to far field sheaths is the dominant mechanism competing with the weak central absorption due to TTMP and Landau damping: the rf electric field available at the wall to produce the $E_{||}$ that in turn excites the sheaths [4] is exponentially larger at smaller gaps, just as the antenna loading increases at smaller gaps. Furthermore, the electron density at the wall would be expected to be higher at smaller gaps, which also tends to increase the power dissipated in the sheaths.

This trend continued in the FW current drive results. The loop voltage profile as a function of time and the fast wave current drive were deduced by comparing sawtooth-free discharges with co- and counter-current drive antenna phasings, as has been described previously [5]. Up to 80 kA of central FWCD were measured in the 8 cm gap case, as shown in Fig. 2.5.2–3(a); again, the measured driven current declined at smaller gaps, despite the higher coupled FW power. The measured FWCD was compared with code calculations based on the ergodic multipass limit [8], with no assumed edge loss. The results are shown in Fig. 2.5.2–3(b); a substantial edge loss [2] would be required to explain the current drive and heating deficit in the 3.5 cm case. The key point is that the central current drive efficiency, a local quantity, cannot itself depend on the edge plasma conditions. Hence, the measured sensitivity of the driven current to the edge plasma parameters must result from the dependence of the edge loss mechanism that competes with central damping in the multipass limit on edge parameters.

Future experimental work will aim to measure the far-field sheaths directly with probes distant from the fast wave arrays and to investigate the correlation between the sheaths and the edge loss required to account for the measured central electron heating and current drive efficiencies as a function of outer gap. Also, previous observations have shown a relationship between the ELM characteristics and the central heating efficiency. Direct measurement of the sheaths and their dependence on the ELM characteristics should facilitate finding ELMing H–mode conditions that are compatible with efficient high-power central fast wave heating and current drive.

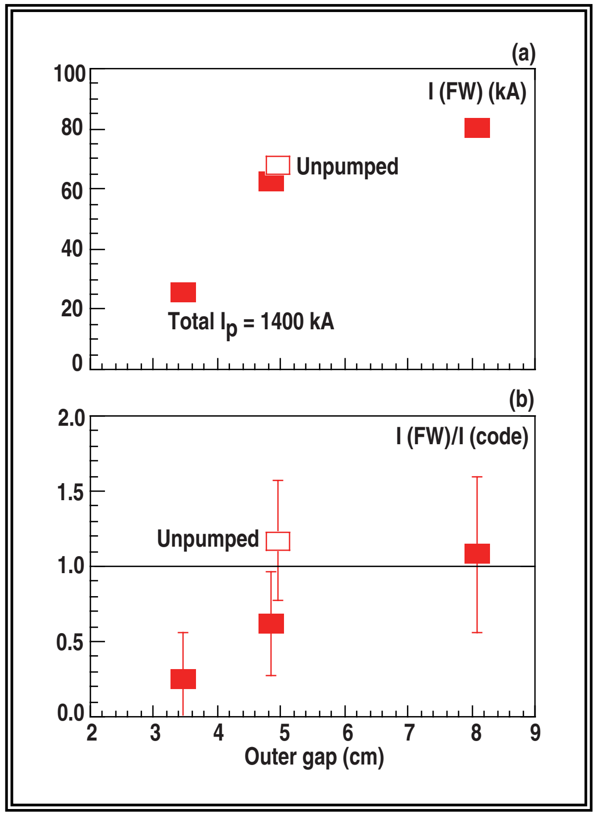

Fig. 2.5.2–3. (a) FW-driven current (co-current phasing) in the outer gap scan. (b) Ratio of measured FW-driven current to that calculated from the multipass code, assuming no edge loss.

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3. THE DIII–D NATIONAL FUSION FACILITY — STATUS AND UPGRADES

3.1. OVERVIEW OF CURRENT CAPABILITIES

DIII–D National Fusion Facility provides the capability to carry out a wide range of state-of-the-art tokamak experiments. At the heart of the facility is the DIII–D tokamak, which is capable of operating at plasma currents up to 3.0 MA with a magnetic field of 2.2 T. The DIII–D tokamak is renowned for its research in highly noncircular limiter and divertor plasma configurations. Substantial plasma heating and current drive capability is available from 20 MW (delivered) of neutral beam heating, 6 MW (source) of ICRF power and 2 MW (source) of ECRF power. The DIII–D diagnostics set provides over 50 diagnostic systems capable of providing definitive measurements of plasma parameters in the core, edge, and boundary regions of the plasma. Control of the tokamak, heating systems, and auxiliaries is managed through a set of interconnected computers.

Operation of the DIII–D facility is the responsibility of GA, who provides the core operational engineering and technical staff, along with the appropriate infrastructure to organize the effort. Many collaborators in the DIII–D Program participate in operational activities including, in particular, the design, installation, and operation of diagnostics and plasma systems such as rf heating, pellet injection, etc. GA is responsible for coordinating and focusing these efforts, ensuring safety, and maintaining appropriate levels of quality.

The DIII–D facility provides over 100,000 sq. ft. of floor space on a ten acre site dedicated to support the activities of the DIII–D Program and its collaborators (Fig. 3–1). The DIII–D tokamak is located at the heart of the facility (Fig. 3–2) with the many support systems, utilities, and diagnostics arrayed around it.

Excellent progress has been made over the last ten years in keeping the facility vital by continually upgrading and refurbishing the tokamak systems in order to address the challenging issues of tokamak research. The major changes are summarized in Table 3–1 along with some of the important steps in physics progress. Not shown here are the numerous improvements to the diagnostic systems that have made the DIII–D facility the best diagnosed tokamak in the world. The current status of the diagnostics is presented in Section 3.6. Also not shown are the many ongoing upgrades to the computer systems for control data acquisition and analysis that have been carried out to meet the needs of the program. The current status of the system is given in Section 3.5.

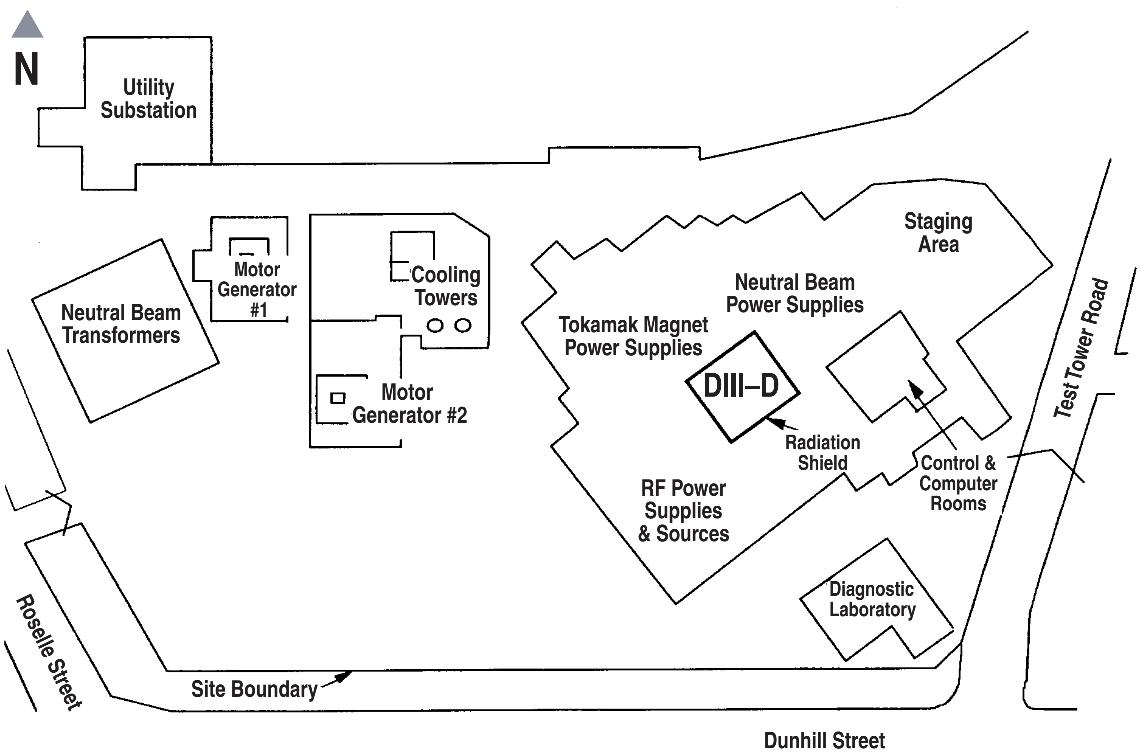


Fig. 3–1. The DIII–D tokamak facility spans a half city block.

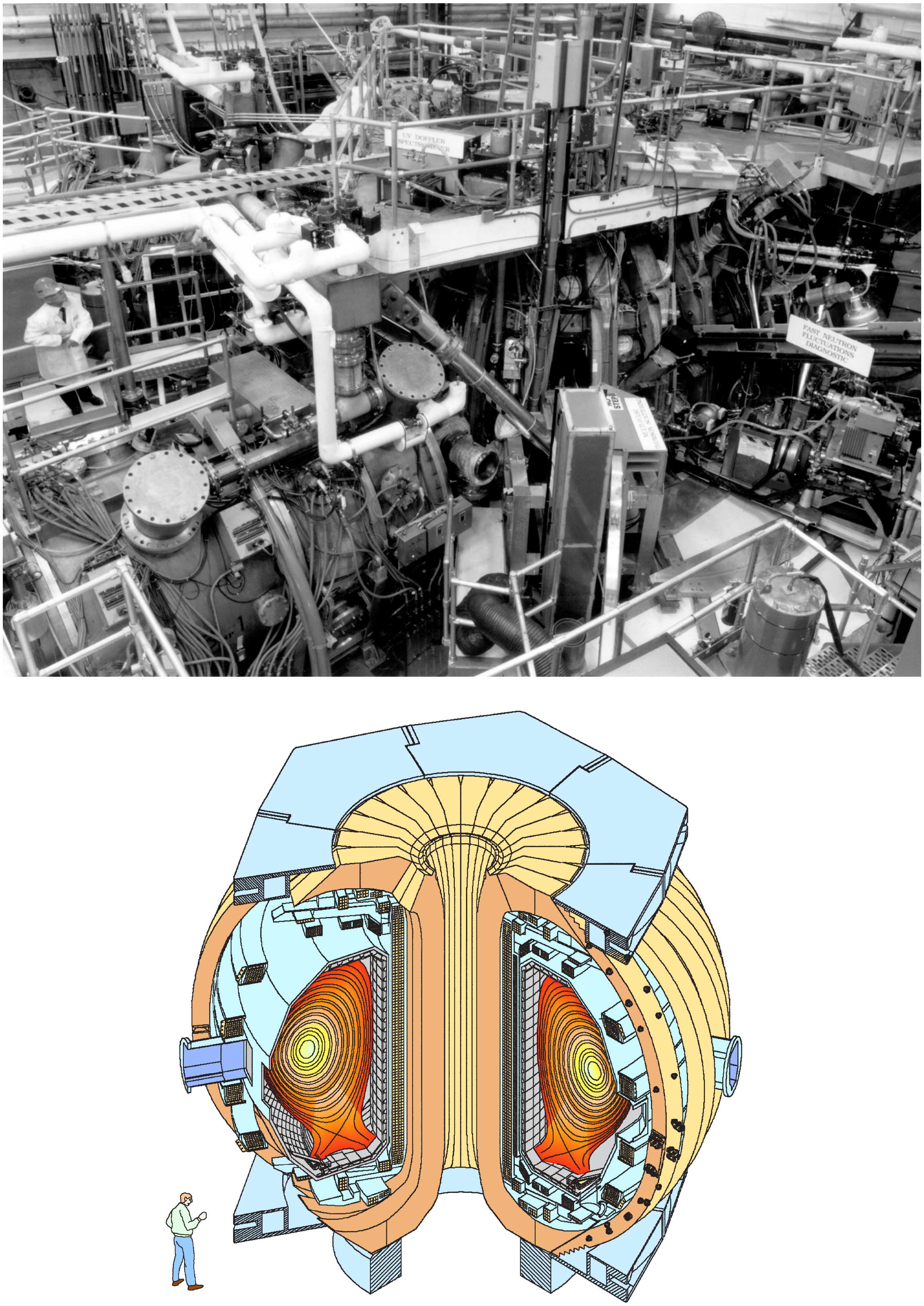


Fig. 3–2. The heart of the facility is the DIII–D tokamak with its many support systems, utilities and diagnostics.

TABLE 3–1
MAJOR EVENTS IN THE HISTORY OF THE DIII–D FACILITY — 1989 THROUGH 1998

Month	Year	Event
Apr	89	Radiation shield enclosure complete
Aug	89	Achieved 19.6 MW neutral beam injection
Oct	89	Achieved 3 MA double-null divertor
Jan	90	Install CER and horizontal ECE diagnostics
Feb	90	Achieved 10 s H–mode plasmas
Feb	90	Motional Stark effect diagnostic operational
June	90	Install ADP ring electrode
June	90	Install first FWCD antenna
June	90	Install first 110 GHz ECH launcher
Nov	90	Multipulse Thomson scattering complete
Dec	90	1 MW FWCD power into tokamak
Dec	90	Fast stroking Langmuir probe installed
Jan	91	First biased internal ring experiments
May	91	First boronization
May	91	VH–mode discovered
June	91	Beta of 11.2% achieved
Mar	92	110 GHz ECH into plasma
June	92	Between shot boronization
Nov	92	Install ADP in-vessel cryopump
Jan	93	Cover all vessel walls with graphite tiles
Feb	93	Switched to exclusive of glow discharge cleaning
June	93	Repaired TF coil prestress structure
Oct	93	Measured He exhaust from plasma
Oct	93	Achieved beta of 12.6%
Jan	94	Install deuterium pellet injector
Apr	94	Complete installation of second and third ECH antenna
Sep	94	Magnetic field correction coil installed
Oct	94	2 MW ICH with new antennas
Feb	95	Install divertor Thomson scattering
Feb	95	Install divertor reflectometer
Feb	95	Install beam emission spectroscopy diagnostic
Apr	95	Ohmic heating coil lead failure identified
Jul	95	Ohmic heating coil returned to service at half capability
Dec	95	Record DIII–D performance: 4.4 MJ stored energy, Q_{DD} = 0.0015
Feb	96	Remote experimental connectivity begun
May	96	Plasma control using real-time equilibrium calculations
June	96	First 110 GHz ECH physics experiments; T_e = 10 keV
Feb	97	Upper divertor plenum and cryopump installed
Feb	97	Radial MSE diagnostic installed
May	97	Isoflux plasma feedback control implemented
June	97	Second 110 GHz gyrotron generates 850 kW, 2-s pulse
June	97	Remote operation from LLNL
Nov	97	Repair of ohmic heating coil lead completed
Apr	98	Off-axis ECCD experiments
May	98	First resistive wall mode control experiments
June	98	Hampton University VUV diagnostic operational
Sep	98	Install central Thomson scattering
Sep	98	Install high field side pellet injector

3.2. TOKAMAK OPERATIONS

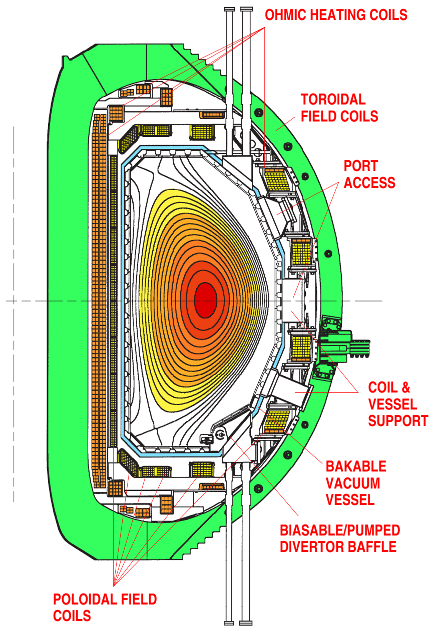
In the years 1998 and preceding, the DIII–D facility operated for research from 8:30 a.m. to 5:00 p.m. with technical staff arriving up to two hours earlier to prepare and staying afterward for shutdown. Research operations have been carried out on a five-day-a-week basis for three weeks of operation followed by two weeks of maintenance, calibration, and testing. In addition, longer shutdowns are sometimes needed for major maintenance tasks. Typically one longer period is set aside each year for new installations and major refurbishments. In recent years, the number of operating weeks has been limited by funding (8 weeks in FY97 and 13 weeks in FY98 compared to up to 27 weeks earlier). The number of operating hours per year could readily be increased by a factor of two or more with appropriate funds.

The tokamak is housed within the machine hall, which provides access control during operations and provides radiation shielding to allow deuterium to be used as fuel in the tokamak. Within the machine hall, the tokamak is surrounded by heating systems, most notably the large neutral beam lines, diagnostic systems, and other auxiliary systems.

The DIII–D tokamak uses conventional water-cooled 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. They routinely operate at full 2.2 T toroidal field and at 2 MA plasma current for a discharge flat-top duration of 5 s (Fig. 3–3). This can be readily extended to 10 s with modest upgrades of the coil system connections, feeds, and power supplies. Operation for longer duration at lower field and plasma current is also possible. The DIII–D coil configuration is noteworthy for its 18 independently controlled poloidal field shaping coils, each powered by an independent current regulator. These coils shape the highly noncircular plasma cross sections which are typical of the DIII–D Research Program. Lastly, there is a set of six 5 m^2 picture frame coils mounted in a belt around the midplane which correct the residual error fields due to anomalies in the magnetic field configuration.

DIII–D presently has a comprehensive carbon first wall and divertor targets to protect the vacuum vessel in areas of high heat fluxes and to limit high Z impurities in the plasma (Fig. 3–4). Graphite is an effective choice because it has low atomic mass so that sputtered graphite entering the plasma has little impact, it has good thermo-mechanical properties in contact with the hot plasma, and it has good thermal conductivity. The first wall is a robust system, operating for ten years without failures, providing maximum experimental time. The wall consists of inertially cooled graphite tiles, absorbing energy during a discharge then releasing it to the water cooled vessel wall through a compliant heat transfer interface in the ten minutes between discharges. The first wall is conditioned for operation by first baking and outgassing under vacuum at 350°C. The wall is then coated with a fine layer of boron (boronization) which serves largely to getter oxygen in the vessel. Finally, helium glow discharge cleaning is used in the interval between discharges to clean and degas the wall surfaces before the next discharge.

A wide range of plasma configurations can be run including single-null divertors with either upper or lower null, highly shaped double-null divertors, and limiter discharges on any of the surfaces. The tile design provides cavities between the vacuum vessel wall and the tiles for diagnostics, protecting them and their signal cables from large heat fluxes.



	PRESENT	PROPOSED
Major radius	1.67 m	
Minor radius	0.67 m	
Maximum toroidal field	2.2 T	
Available OH flux	7.5 V–s	
Maximum plasma current*	3.0 MA	
Neutral beam power (80 keV)	16 MW	8 MW
RF power (110 GHz)	2.0 MW	10 MW
RF power (30–120 MHz)	6 MW	8 MW
Current flattop (divertor at 2 MA)	5 s	10 s

*Divertor operation; Limiter operation of 3 MA achieved.

Fig. 3–3. DIII–D capabilities allow a wide range of research and technology issues to be addressed.

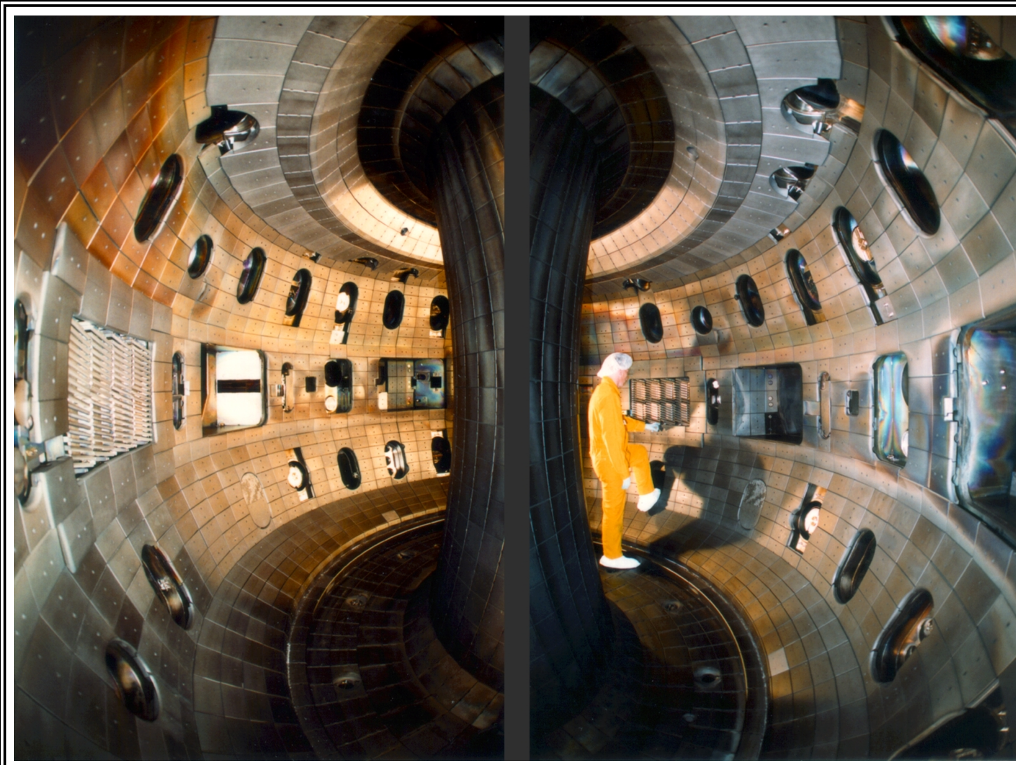


Fig. 3–4. The entire DIII–D first wall is graphite.

A toroidally continuous baffle with cryopump in the lower divertor region has been in use for several years (Fig. 3–5). Divertor characterization experiments have been successful using the extensive lower diagnostic set to benchmark many computer plasma models. Low triangularity plasmas are routinely pumped by placing the separatrix at the aperture of the divertor plenum to provide density and particle control with 40,000 l/s pumping speed. These pumps operate at liquid helium temperatures and actively pump both the plasma fuel gas and all volatile impurities during the discharge. They also have the capability of substantially lowering the plasma density. The cryopump at the bottom of the plasma chamber is optimized to pump the edge of single-null divertor discharges with low triangularity.

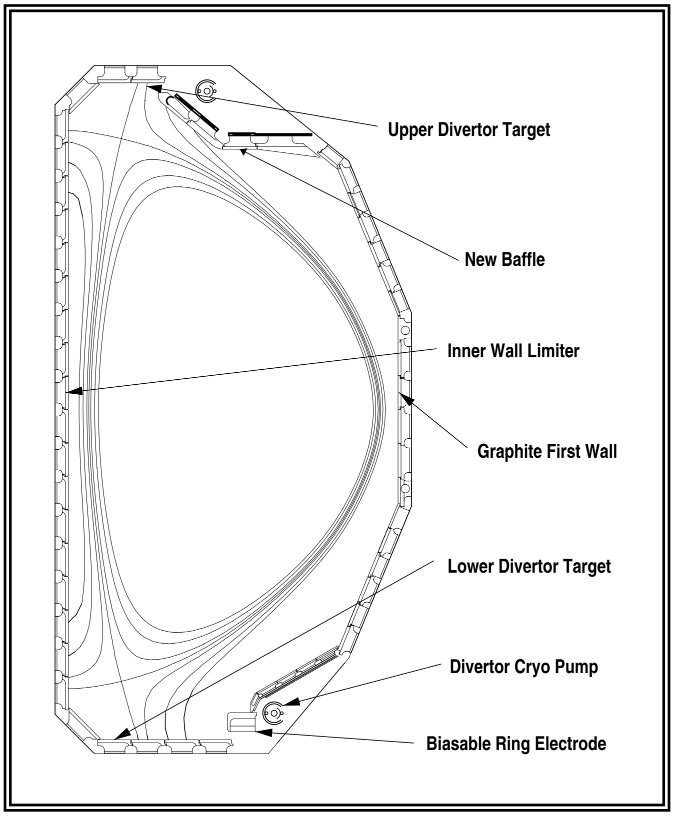


Fig. 3–5. The carbon first wall and divertor targets protect the vacuum vessel and limit high-Z impurities.

An integrated biasable ring electrode has allowed the study of the effects of electric fields on the neutral pressure in the baffle, as well as evaluated novel noninductive startup techniques. The upper divertor target area was modified in the beginning of FY97 with the installation of the first phase of the Radiative Divertor. This hardware included a toroidally continuous baffle and cryopump similar to the existing lower system in the bottom except the aperture to the pumping plenum is at a smaller minor radius, allowing pumping and particle control high performance, high triangularity discharges. The structure is comprised of water-cooled Inconel 625 panels with graphite tiles mounted to the surface. The cryopump is of a design similar to the proven lower pump. The pump provides pumping speeds of nearly 40,000 l/s for the high performance discharges. The D_2 is defrosted from the pumps during the helium glow between discharges and the pumps are fully defrosted and outgassed during nonoperational periods.

The gas puff system and pellet injector provide for control of the plasma fueling. The gas system provides a completely programmable source of a diverse range of gases to initiate the discharge and fuel it from the edge during the pulse. The pellet injector provides fueling deeper into the plasma by injecting high velocity pellets of frozen fuel gas from the plasma edge. The injector is capable of delivering a continuous stream of pellets during a discharge.

A substantial number of more utilitarian systems are necessary to operate the facility. Prime power for the auxiliary heating systems is taken from the local utility power mains. The power for the coil systems is supplied by one of two flywheel energy storage motor generators (525 and 260 MVA). These generators are spun up to full energy between discharges and then the energy is drawn out during the 10 s of the discharge. The coils are powered by a set of phase controlled power supplies. In the case of the plasma shaping coils, there is a series switching current regulator in series with each. The auxiliary heating systems are powered by 12 high voltage power supplies each typically capable of 6 MW of power.

A 150 l/h helium liquifier provides the cryogenic helium needed to support operation of the neutral beamlines, ECH magnets, pellet injector, and divertor cryopumps. A substantial water conditioning system supplies the high pressure, high purity water needed to cool the coils and other systems.

Operation of the tokamak with deuterium fuel results in significant neutron production. These neutrons create a need for radiation monitoring and control. The radiation shield forming the wall and roof of the machine hall reduces the radiation levels sufficiently to allow the facility to be operated with acceptable exposure to the public and workers. Radiation levels at the site boundary are limited to 40 mRem/yr by agreement with the DOE. Radiation levels for staff are limited to 5000 mRem/yr by the NRC and internally to 400 mRem/qtr. The facility is operated within the ALARA principles in order to keep radiation doses as low as reasonably achievable. The quarterly site radiation levels are summarized in Fig. 3–6.

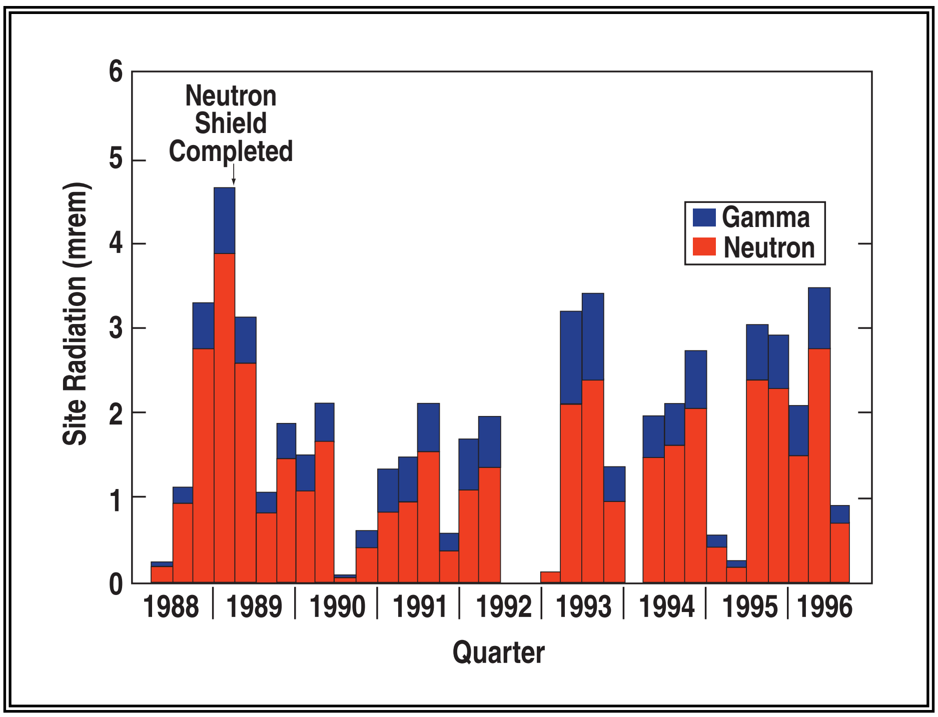


Fig. 3–6. Quarterly boundary radiation levels show the site is maintained well below the 40 mrem operating limit.

Substantial auxiliary heating is provided to heat the tokamak discharges to the temperatures needed to achieve the conditions appropriate for efficient fusion reactions and to facilitate driving currents in the plasma. The capabilities of these heating systems are summarized in Table 3–2. The neutral beam systems are the workhorse of day-to-day operation. They are routinely available on demand to provide heating at their design levels. They have also become an important source for a number of diagnostics including ion temperature, current profile, and turbulence. The ICRF system is fully operational and experiments are underway to refine techniques to couple the power to the plasma. The 110 GHz ECH system has been commissioned and has been used in plasma experiments.

TABLE 3–2
POWER TO PLASMA OF AUXILIARY HEATING SYSTEMS (JUNE 1998)

System		P_{max} (MW)	Duration (s)	P (5 s) (MW)	P (10 s) (MW)
Neutral beams	LBL 80 kV	20	3	16	8
ICH	ABB 30–55 MHz*	2.8	20	2.8	2.8
	FMIT 30–60 MHz	1.4	≥10	1.4	1.4
ECH (110 GHz)	Gycom gyrotron	0.75	2	0.30	0.15
	CPI gyrotron	0.80	0.8	0.28	0.24

*The rf power must be decreased above this frequency (to 20% at 120 MHz).

3.3. NEUTRAL BEAM HEATING SYSTEMS

The DIII–D Neutral Beam Systems consist of four beamlines, and each beamline has two positive ion sources in parallel, focused through a common drift duct. These neutral beam systems were designed for 5 s deuterium beam operation at beam energy of 80 keV with 16 MW of total injected neutral beam power from eight sources. They routinely operate at this level. Improvements in operational technique and in system hardware have led to the routine operation in deuterium at beam power level of 20 MW for 3.5 s. Successful testing and operation of three ion sources at 93 keV deuterium beam energy also leads to the possibility of enhancing system capability to 28 MW. Control and data acquisition computers have recently been upgraded, along with several instrumentation and control systems to improve system functionality, availability, and reliability.

3.4. RF SYSTEMS

At present, we have two 110 GHz gyrotrons operating at a nominal 1 MW (source) power level. The first gyrotron is made by Gycom in Russia. It has an edge-cooled window of boron nitride which limits the pulse length to 2.0 s at a power level of 1 MW. It has achieved power levels of 960 kW for 2.0 s pulses in tests in Russia. The other gyrotron is made by CPI (formerly Varian). It has a face-cooled window of sapphire which limits the power to 1 MW for 0.8 s or 0.5 MW for 2 s. Both vendors indicate their designs are cw compatible except the window. These gyrotrons have injected power into DIII–D through the transmission system, and the beam patterns and locations generated in the vacuum vessel correspond approximately to those expected from the theory of Gaussian beam propagation and from vacuum ray tracing using a 3D computer model. A third 1 MW gyrotron is expected from CPI in late 1998.

The transmission system for these gyrotrons is evacuated corrugated waveguide of diameter 31.75 mm propagating the HE_{11} hybrid mode. Presently there are four launchers on the DIII–D tokamak, each capable of launching 1 MW of ECH power. Each launcher comprises a 60.3 mm diameter corrugated waveguide launcher, a fixed focusing mirror located about 30 mm from the termination of the waveguide, and a steerable flat mirror that can be pivoted poloidally so the rf beam can be aimed at any elevation between the plasma center and the upper edge. They are tilted off-normal by 19 deg in the toroidal direction so that the rf beam is launched in a manner which will drive toroidal plasma co-current near the cyclotron resonance. The steering in the toroidal direction can be changed, but only during a vent of the vacuum vessel. The present mirrors are tilted 19 deg in the toroidal direction in order to generate co-current drive.

The three ICH heating systems consist of three power sources all capable of a nominal 2 MW of power at the source and three launchers located on the midplane of the machine. The capabilities of these systems are summarized in Table 3–3. The antennas are each protected with a Faraday shield set at the nominal angle of a field line during normal operation and bumper limiters to protect them from excessive direct contact with the plasma.

TABLE 3–3
ICH HEATING SYSTEM CAPABILITY

POWERSYS	PWR (MW)	f (MHz)	Duration (s)	ANT	Duration (s)	Functionality
FMIT	2	30–60	10	4 strap	2	FWCD. FWH
ABB	2	30–120	20	4 strap	10	FWCD, FWH
ABB	2	30–120	20	4 strap	10	FWCD, FWH

3.5. COMPUTER SYSTEM

An extensive array of computer systems is used to operate the tokamak and auxiliary systems, collect the data, and carry out the analysis (Fig. 3–7). These computers are interlinked in a network that effectively applies these resources to the needs of the program. The tokamak control computer provides for control and monitoring of the entire operating cycle. Critical safety limitations are applied with hardwired systems. The heating systems are separately controlled. The acquisition and archiving of data is controlled by another computer that serves as the hub of a large network of computers, both at GA and off-site, used to provide storage and analysis of data. In addition, an array of computers is used to operate, manage, and analyze the data for the diagnostics.

The plasma control system provides state-of-the-art high speed digital control of the plasma cross-sectional shape (magnetic configuration) and key plasma profile parameters. This system uses multiple input multiple output control technology that allows the wide range of plasma shapes studied in the DIII–D Program to be routinely operated on a shot to shot basis. Recently the implementation of isoflux control has provided realtime control of the plasma boundary using realtime calculation of the MHD equilibrium to evaluate the plasma configuration. The system has the capability for integrated control of the plasma profile parameters using diagnostic measurements as inputs. The system also serves as a platform for the seamless addition of control functions for other parameters such as the plasma density, total energy, or coupling to the ICRF antenna.

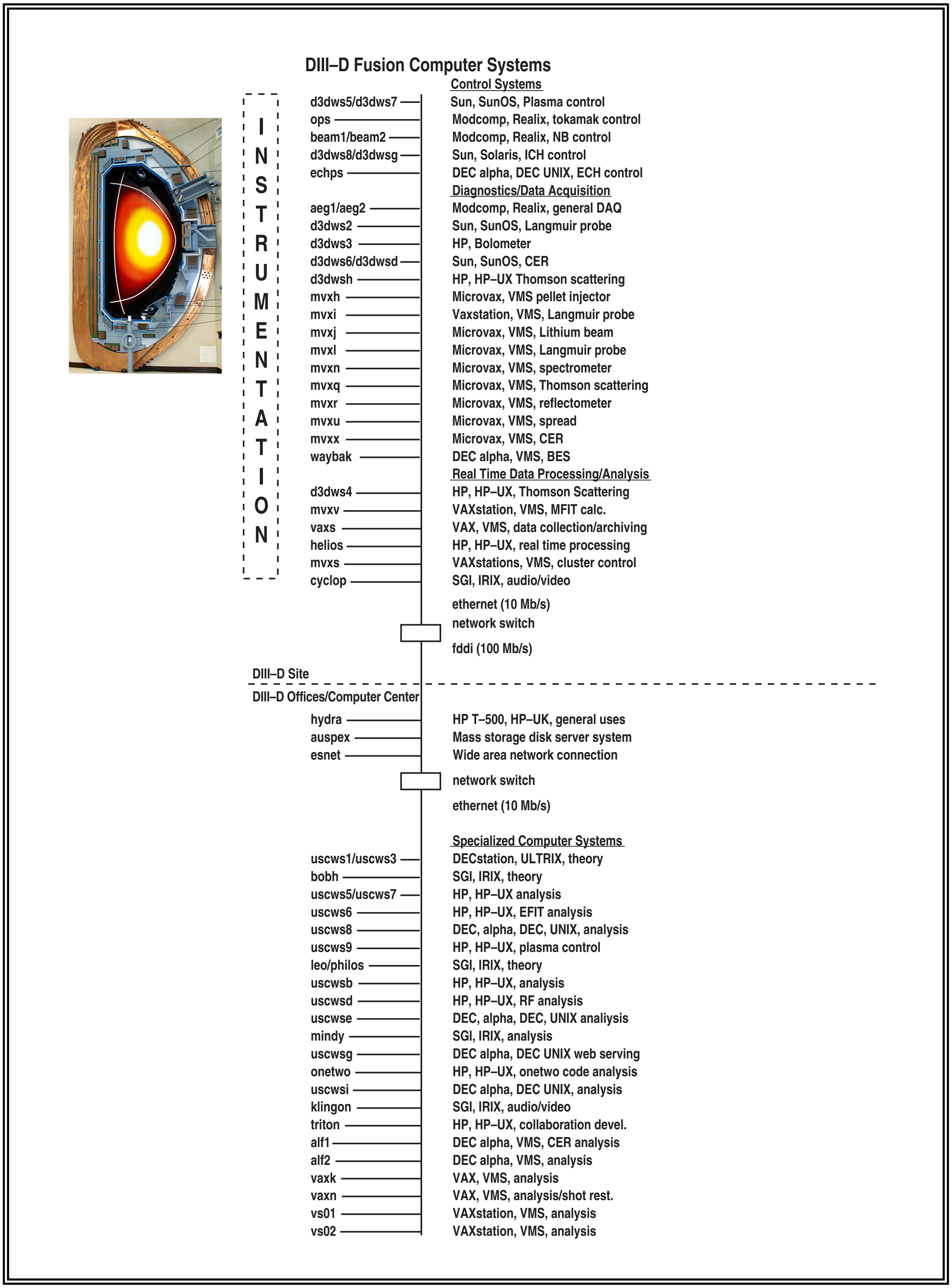


Fig. 3–7. An extensive array of computer systems operates the tokamak and collects and analyzes the data.

3.6. DIAGNOSTIC SYSTEMS

The DIII–D plasma diagnostic set is made up of more than 50 instruments built and operated by the DIII–D National Program. This ensemble of instruments is the most complete of any tokamak in the world and routinely produces the high quality data required to fuel the DIII–D Scientific Research Program. The DIII–D diagnostics set includes extensive divertor and edge measurement capability, plasma core profile measurements of density, temperature and plasma current and a large suite of fluctuation diagnostics. A complete list of the diagnostic systems installed on DIII–D and the measurements that they make is shown in Table 3–4.

3.7. COMMUNITY OUTREACH

The fusion group maintains a vigorous program of community outreach. Tours are provided for students and a wide variety of professional and community organizations.

The Fusion Group operates an Education Outreach Program for middle and high school students throughout San Diego County. The program enables teachers and scientists to work together closely to produce effective educational materials on fusion science and technology for classroom use, and allows students unique opportunities to discuss science, engineering, and math topics with professional scientists and engineers. Key deliverables from previous work include workbooks, a curricular chapter on the electromagnetic spectrum, a poster for classroom display, a videotape on nuclear fusion energy production and DIII–D facility tours and tour stations. Workshops which cover the curricular materials are provided to teachers and educators to enhance their fusion knowledge base. Over the previous two years, more than 100 educators have attended our workshops, while others have used the materials as a basis for teaching a unit on nuclear fusion. Future work will include the production of an interactive CD–ROM on plasma science and fusion technology, expansion of the curriculum notebook to include activities on radiation, ICF, plasma, and fusion related engineering, and significant contributions to professional journals and classroom textbooks on plasma science and fusion technology. Classroom visits by scientists and engineers will also become part of the program.

A unique aspect of the Education Outreach Program is a three-hour DIII–D Tokamak Facility tour given to student groups. The tour is the culminating activity in the Educational Outreach Program. Students are given a brief overview of the on-going, worldwide efforts in harnessing nuclear fusion as an energy source and are then given a multistation tour of the facility. At the different stations, small groups of students participate in demonstrations and hands-on activities that cover general areas of science and technology. The stations are titled “Plasma — The 4th State of Matter,” “The Electromagnetic Spectrum,” “Engineering Analysis and CAD,” “Data Acquisition and Computers,” “Radiation, Radioactivity, and Risk Assessment,” “Inertial Confinement Fusion,” and “DIII–D Model and Experimental Hall.” At each station, scientists and engineers discuss topics and present demonstrations to reinforce the concepts presented. During the previous two years, more than 2000 students have toured the DIII–D facility.

TABLE 3–4
DIAGNOSTIC SYSTEMS INSTALLED ON DIII–D

Electron Temperature and Density	
Multipulse Thomson scattering	8 lasers, 40 radial points
ECE Fourier transform spectrometer	Horizontal midplane profiles
ECE radiometer	Horizontal midplane
Multichannel vibration compensated (infrared) interferometer	3 vertical chords, 1 radial chord
Microwave reflectometer	Midplane edge profiles
Ion Temperature and Velocity	
Charge exchange recombination spectroscopy	16 vertical channels; 16 horizontal channels; 3 mm edge resolution
Core Impurity Concentration	
VUV survey spectrometer (SPRED dual range)	Radial midplane view
Visible Bremsstrahlung array	Radial profile at midplane, 16 channels
Radiated Power	
Bolometer arrays	2 poloidal arrays, 48 channels each
Divertor Diagnostics	
Visible spectrometer	7 channels
VUV survey spectrometer (SPRED)	Vertical view along outer divertor leg
Tangential TV (visible)	2–D image of lower divertor
Tangential TV (VUV)	2–D image of lower divertor
Infrared cameras	5 cameras
Graphite foil bolometers	12 locations
Fast neutral pressure gauges	4 locations in divertors
Penning gauges	Under divertor baffle
Baratron gauge	Under divertor baffle
Langmuir probes	18 radially across lower floor, 2 upper divertor throat
Moveable Langmuir probe	Scannable through lower divertor outer leg
Tile current monitors	Radial and toroidal arrays
Reflectometer	Vertical view through X-point
Magnetic Properties	
Rogowski loops	3 toroidal locations
Voltage loops	41 poloidal locations and 30 saddle loops
B_θ loops	2 × 29 in poloidal arrays
Diamagnetic loops	9 toroidal locations
Plasma Edge/Wall	
Plasma TV	4 cameras, radial view, rf antennae
IR camera	Inside wall and coiling views
Visible filter scopes	16 locations
Moveable Langmuir probe	Scannable across outer midplane

TABLE 3–4 (CONTINUED)

Fluctuations/Wave Activities	
Microwave reflectometers	2 radial systems
Far infrared scattering	Radial view
Infrared scattering	Vertical view
Mirnov coils	Toroidal, poloidal, and radial arrays
Li beam injector	Radial beam with 16 channel tangential viewing channels
X-ray imaging system	100 channels, 5 arrays
RF probes	10 probes in poloidal array, 10 probes in toroidal array, 1 launch antenna
Fast Ion Diagnostics	
Neutral particle analyzer	Scannable horizontal view, 3 vertical views
Fast neutron scintillation counters	2 radial channels
Fusion products probe	1 new midplane probe
Plasma Current Profiles	
Motional Stark polarimeter	35 channels, 2 radial arrays
Nonthermal Electron Distribution	
Soft x-ray pulse height spectrometer	1 scannable radial view
ECE Michelson spectrometer	1 vertical view
Miscellaneous	
Neutron detectors	3 toroidal locations
Hard x-ray monitors	2 toroidal locations
Synchrotron (IR) radiation detector	2 tangential chords on midplane
Torus pressure gauges	
Residual gas analyzer	

4. MAJOR UPGRADE PROJECTS IN THE PAST 10 YEARS

4.1. NEUTRON SHIELDING

4.1.1. INTRODUCTION

A neutron shielding system was constructed for the DIII–D facility to allow a full program of experiments in deuterium while limiting the radiation at the site boundary to 20 mRem per year. The goal of the shielding effort for the DIII–D facility is to provide a shielding enclosure (walls and roof) which reduces the radiation at the site boundary by a factor of 300.

It is desirable to minimize the interference of the shielding system with the existing diagnostic and other devices. It was also necessary to maintain access to the torus hall using the existing overhead 20-ton bridge cranes in order to handle heavy equipment and machinery. The weight of the added shielding system was kept to a minimum since it will increase soil bearing pressures that can produce differential settlement in the DIII–D facility.

4.1.2. DESIGN DESCRIPTION

Several concepts were evaluated to provide the required supplemental shielding to the DIII–D facility. The conclusion of the engineering study is that the Translating Overhead Protective System concept best met the design requirements since it provides minimal experimental interference, maintains access to the machine pit area, and minimizes additional structural and foundation work, if any, that may have to be performed on the building.

Figure 4–1 shows the chosen concept, which consists of a fixed roof, translating roof, and side walls. Figure 4–2 shows a CAD isometric projection of the roof.

Table 4–1 presents the basic specifications for the shielding system. Water in the form of gel was used as the shielding material for the roof because of its availability, low cost, and ease of installation. The gel is packed in fiberglass boxes which are placed over the roof supporting structure. The gel, though more expensive than water, has the following advantages: (1) it has little water-sloshing effect during possible seismic activities, (2) it keeps boron uniformly distribution, and (3) it is leak resistant against small holes or punctures.

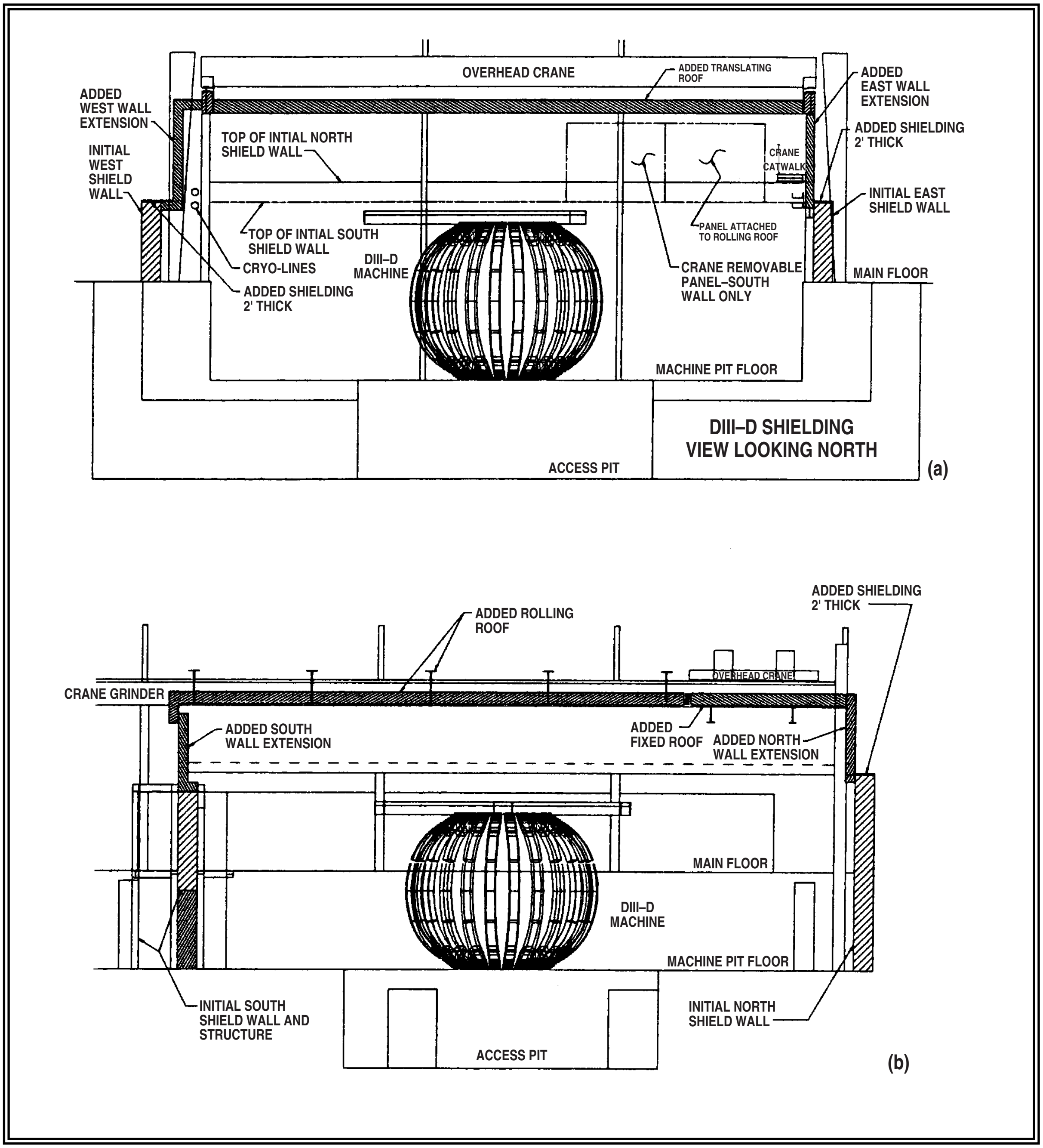


Fig. 4–1. Overall shielding geometry of DIII–D. (a) View looking north. (b) View looking west.

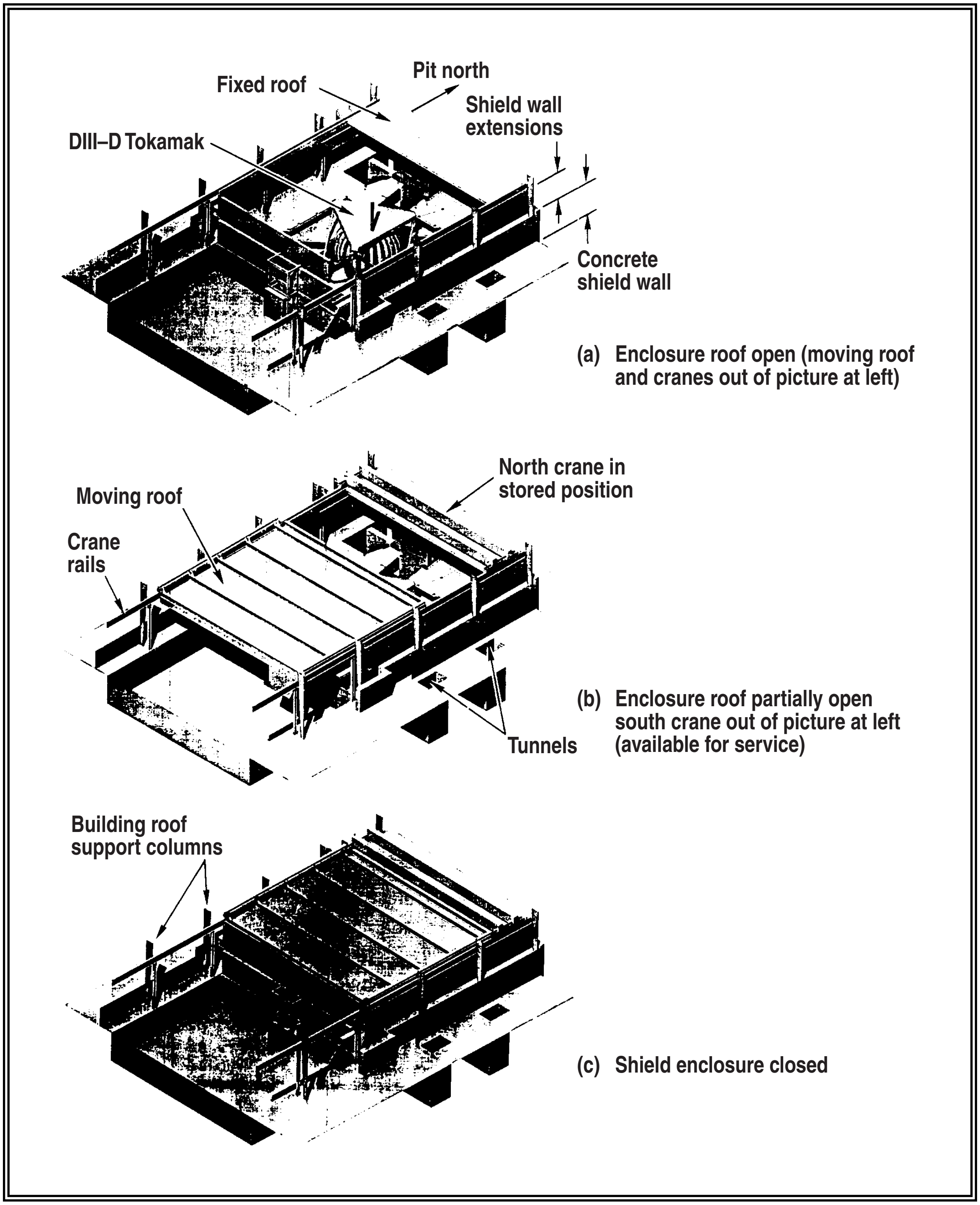


Fig. 4–2. Roof design. DIII–D radiation shield concept. The shielding enclosure is completely within the DIII–D building. The enclosure roof opens to allow crane access to the tokamak and surrounding equipment.

TABLE 4–1
BASIC SPECIFICATIONS OF THE NEUTRON SHIELDING

Shielding Materials	
Roof	13 in. of borated water in gel form packed in fiberglass boxes.
Walls	Pourable shielding materials or poly/boron sheets equivalent to 12-in. B-poly. Flexible hydrogenous foam is considered in penetrations and embedments where conduits and pipes have to be moved.
Dimensions	
Fixed roof	14 × 61 ft
Translating roof	54.5 × 61 ft
Fill-in walls	
North	61 × 7 ft
South	61 × 9 ft[a]
East	68.5 × 10.5 ft
West	68.5 × 12.5 ft[b]
Environmental Control Systems	
Heat removal	A new ventilation system installed in the torus hall area in order to regulate the temperature inside.
Fire protection	A fire-suppression sprinkler system along with smoke and fire alarm is provided.
Lighting	A lighting system is attached to the translating roof and sufficient lighting underneath the fixed roof.

[a]Includes 2-ft overhang attached to translating roof.

[b]Includes 3-ft, 10-in. horizontal slab wall.

4.1.3. FIXED ROOF

The fixed roof located at the northern end of the DIII–D facility is supported by the existing crane structure. The roof structure is made of supporting steel structure framing with built-up plate girders and 1–3/8-in. corrugated metal decking for support of the selected shielding material. In order to provide access to the 20-ton cranes, the existing 20-ton north crane will be parked over the fixed roof when the translating roof is closed over the pit. Also, to access the machinery underneath the fixed roof, such as the neutral beam source housings, etc., a 4-ton underhung bridge crane will ride on the east-west spanning girders and will be able to move objects at least 6 ft from the southern end of the fixed roof.

4.1.4. TRANSLATING ROOF

The translating roof structure rides on the existing crane support structure and can be moved in or out of the machine pit area. This permits easy access to the torus hall area and allows installation and maintenance of the moving portion of the roof to be performed in the high bay area away from the DIII–D machine, thus minimizing interference with the experimental work. The translating roof structure consists of a similar support structure framing with tapered built-up steel girders with preset camber spanning the east-west direction across the machine pit. A 1–3/8-in. corrugated metal decking supports the shielding materials between the girders.

4.1.5. SIDE WALLS

The side walls fill the space between the existing concrete shield walls and the roof shielding. The shielding material is a castable material in a hydrogenous binder placed in fireproof plywood forms.

4.2. DIVERTOR UPGRADES

4.2.1. INTRODUCTION

The objective of the divertor and first wall program for DIII–D is to protect the vacuum vessel in areas of high heat fluxes and to limit high Z impurities in the plasma. The first wall is a robust system consisting of inertially cooled graphite tiles, absorbing energy during a discharge then releasing it to the water cooled vessel through a compliant heat transfer interface in the ten minutes between discharges. A wide range of plasma configurations can be run including single-null divertors with either upper or lower null, highly shaped double-null divertors, and limiter discharges on any of the surfaces. The tile design provides cavities between the vacuum vessel wall and the tiles for diagnostics, protecting them and their signal cables from large heat fluxes.

The divertor improvements performed on DIII–D were staged as two programs. The first upgrade program was the Advanced Divertor Program (ADP) which consisted of a toroidally continuous ring electrode biasable to 600 V with 20 kA and a toroidally continuous cryocondensation pump situated under a toroidal gas baffle attached to the ring electrode (Fig. 4–3). The pumping is achieved by cryocondensation of particles on a 4.6 K liquid helium cooled surface. Particles condensed on the helium-cooled surface are prevented from recycling back into the plasma during a discharge. The second stage of divertor improvements is the Radiative Divertor Program (RDP), which is to provide particle control for high performance Advanced Tokamak (AT) operations and to develop methods of reducing the heat flux at the divertor target without impacting the core confinement. The RDP includes the installation of divertor structures and cryopumps to permit this new research to be carried out (Fig. 4–4).

4.2.2. ADVANCED DIVERTOR PROGRAM DESIGN

A cross section of the pump and ring electrode and their location in the pumping plenum is shown in Fig. 4–3. The pump is comprised of a series of concentric Inconel 625 tubes cut and assembled together. The 1 m^2 pumping surface consists of a 10-m long, 25-mm diam Inconel 625 tue with liquid helium flowing inside. Surrounding the pumping surface are liquid nitrogen cooled shields limiting the steady state heat load the liquid helium system to less than 10 W. Surrounding the nitrogen-cooled surfaces is a radiation/particle shield to prevent energetic divertor particles from releasing water previously condensed on liquid nitrogen surfaces. The aperture to the pump is created by cutting windows in the radiation/particle and outer nitrogen shields. The inner nitrogen shell shields the helium tube from incoming energetic particles. All particles entering the pump must bounce off a nitrogen-cooled surface at least twice before striking the helium surface. The nitrogen-cooled surfaces have a high emissivity to absorb a large fraction of the incoming thermal radiation. The helium and nitrogen systems are electrically connected only at the feed through flange to prevent eddy current heating and additional loads on the pump.

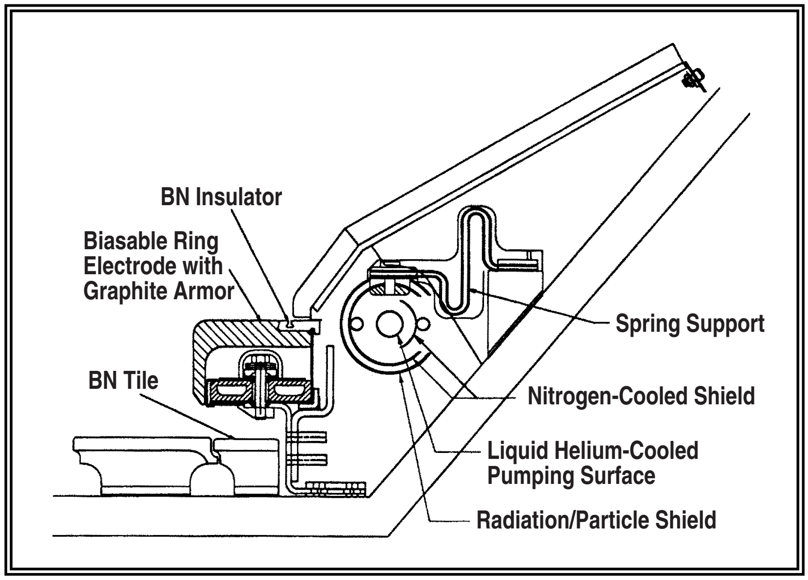


Fig. 4–3. Advanced divertor hardware in outer lower corner of DIII–D vacuum vessel.

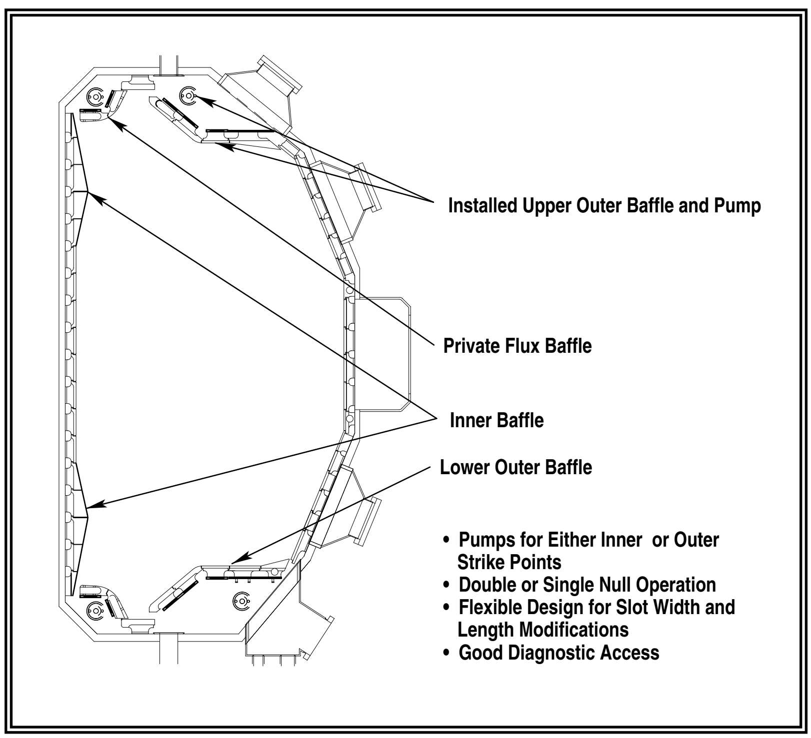


Fig. 4–4. The planned completion of the radiative divertor installation includes the lower baffle and the private flux baffles together with new tiles making up the inner baffles.

The pump is supported to the vessel via flexible supports attached to the nitrogen shell. The support is designed to allow for the thermal contraction of the system at cryogenic temperatures and for low heat leak while the vessel is at 25°C. The natural frequency of the support system in the vertical direction is designed to have no resonance near the 25 Hz natural frequency of the vessel wall. The supports are also designed to attenuate the impulse loads induced on the pump components during disruptions.

Flow of the liquid helium inside the 25 mm Inconel tube is annular flow around a 19 mm-diam core made of a thin walled tube (0.25 mm). The flow channel was designed to achieve a flow velocity of 0.5 m/s and maintain a large thermal capacity. The inner tube has slots to allow liquid helium to fill the center of the core. Flow restrictors are placed every 75 mm in the inner tube to minimize the amount of flow through the center to less than 10%. Extensive testing of different flow configurations led to the choice of this annular flow design based on its heat load capability.

During plasma experiments, the divertor strike point was varied with respect to the pump entrance aperture. A four-fold increase in the particle exhaust rate has been observed as the strike point is positioned closer to the pump. A maximum particle exhaust rate of 5×10^4 Pa-*l*/s has been achieved during experiments.

4.2.3. RADIATIVE DIVERTOR PROGRAM

The radiative divertor is a major element of the DIII–D Program and includes the installation of divertor structures and cryopumps as shown in Fig. 4–4. The first phase installation, the upper outer cryopump and baffle, was completed in February 1997 and is currently being used in the experimental campaign. The second phase is in progress, which is the detailed design and fabrication of the upper inner cryopump. When completed, the RDP will allow for pumping of all four strike points of a double-null high triangularity plasma.

The radiative divertor baffles have been designed to be very flexible as the height and width of the slots can easily be varied. This will allow DIII–D to both optimize the configuration based on experimental results and benchmark computer models with various configurations. The initial installation was designed for a slot width of 1.5 cm and a length of 23 cm based on the values from the combined UEDGE and DEGAS models for optimum reduction of the core ionization. The modeling indicates that if the slot is made narrower, the core ionization increases because the slot becomes a recycling source that is close to the plasma core. If the slot is made wider, neutrals can leak around the plasma and enter the core at the midplane. In the present design, the nominal slot can be changed by about 3.5 cm by adding thicker or thinner graphite tiles. The length of the slot can be increased from 23 to 43 cm by lengthening the supports for the tiles and adding a vertical baffle structure. We can also make a “gasbox” type of divertor by leaving the structure below the baffle open in the 43 cm slot case. It is envisioned that a height change could be done each year during the major DIII–D maintenance period. We plan to change slot widths and slot lengths guided by the data so that the important quantities can be determined and results evaluated.

4.3. 6 MW ICRF

4.3.1. INTRODUCTION

The FWCD program on the DIII–D tokamak is a collaborative effort. Oak Ridge National Laboratory (ORNL) led the design and fabrication of the FWCD antennas. GA was responsible for the generation, transmission, and coupling of the high power rf to the antenna. The experimental activities are carried out with multi-institutional participation.

The 6 MW upgrade consists of three systems each having its own 2 MW transmitter connected to a four-strap antenna by an insulated coaxial transmission line. The transmission line is configured to provide the flexibility of adjusting the phasing of the straps for electron heating (0,π,0,π) and for current drive phasing of (0,π/2,π,3π/2), while providing matching between the antenna-plasma impedance (≈1–4 Ω) and the 50 Ω impedance that the transmitter requires for optimum power delivery. The matching system must also compensate for the mutual inductance between the straps, which has been achieved by using a decoupler concept developed earlier for the original 2 MW DIII–D FWCD system.

An overall schematic of one of the two new transmitter/antenna systems is shown in Fig. 4–5. Both systems are topologically the same, although the routing of the transmission line is different in order to comply with the DIII–D building layout. The transmitters, transmission lines, and the four-strap antennas are described in more detail in the following sections.

4.3.2. TRANSMITTER

The high power rf transmitters are being supplied by THOMSCAST AG, formerly Asea Brown Boveri Infocom (ABB). These transmitters are of the same type as those in service on the ASDEX Upgrade experiment at the Max-Plank Institut für Plasma Physics. The overall design of a 2 MW rf generator is shown in the block diagram (Fig. 4–6).

The transmitter consists of four stages of rf amplification: pre-amplifier, predriver, 100 kW high power driver, and the 2 MW high power final. The input signal is fed to the 50 W rf preamplifier by way of an attenuator and the PIN regulator. The 5 kW predriver stage is of a straightforward design and uses a water-cooled Siemens-type RS1054 transmitting tetrode. The input and output tuning circuits of this grounded-cathode stage are motor driven. The 100 kW high-power driver stage is a grounded-grid configuration using a type 4 CW 150000 tetrode made by Eimac. The output circuit is designed as a coaxial l/4 circuit with a 50 W output impedance adjusted by a motor-driven variable coupling. The 2 MW high power final stage is entirely a coaxial design. It is fitted with an ABB type CQK 650-2 tube, operated in a grounded-grid configuration. The output circuit is made up of coaxial line sections with tuning achieved by motor-driven sliding elements. The control system can store the position of the tuning elements for up to 12 different frequencies, so that when the operator wants to change frequencies, a recall of the stored tuning element locations is all that is required. The specifications for the transmitter are given in Table 4–2.

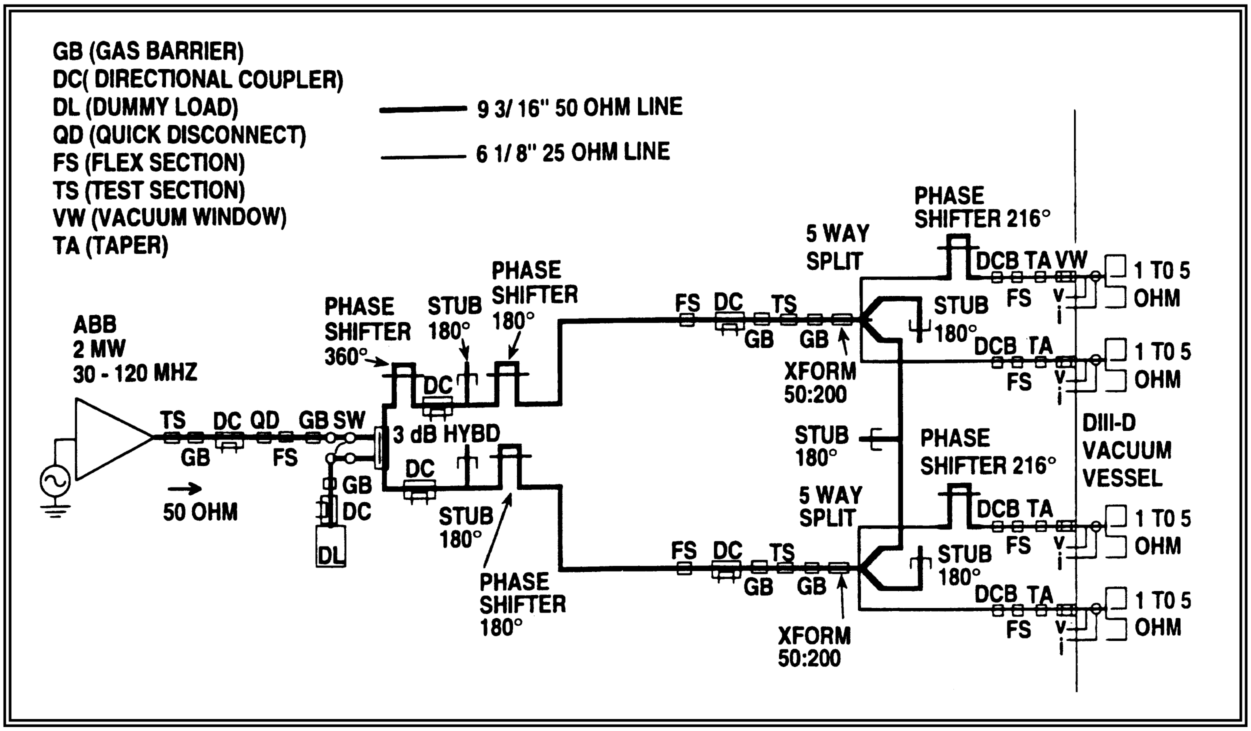


Fig. 4–5. Schematic for one of the 2 MW 30–120 MHz fast wave current drive systems.

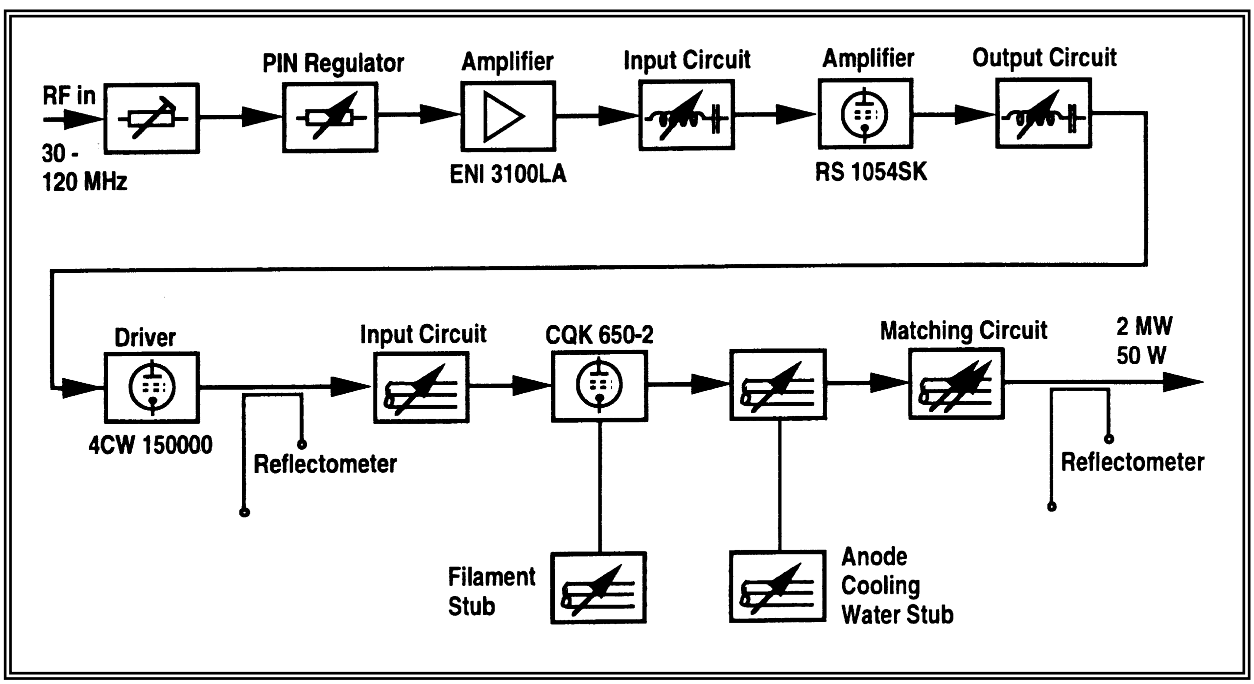


Fig. 4–6. Block diagram of the rf transmitter amplifier stages.

TABLE 4–2
SPECIFICATION OF THE ABB TYPE VU 62 B TRANSMITTER

Frequency range	30 to 120 MHz
Bandwidth	±0.25% – 1 dB; ±75 to ±300 kHz – 1 dB
Output power	From 30 to 80 MHz, 2.0 MW At 100 MHz, 1.5 MW At 120 MHz, 1.0 MW Into 50 Ohms, VSWR 1:15
Pulse duration	20 s maximum
Duty cycle	10% maximum

4.3.3. TRANSMISSION LINE

To achieve the phasing control, the topology shown in Fig. 4–5 was chosen. Alternate straps of the antenna are maintained at 180˚ phasing by use of two resonant loops of 6-1/8-in. transmission line, which are fed at a high impedance point. The resonant loop arrangement is used to lock the phase difference between two of the four straps of the antenna. The phase shifter in this loop has sufficient travel to equalize both sides of the loop. This allows the antenna straps to launch symmetrical spectra. To obtain directed spectra for current drive, the phase shifter of each of the loops is adjusted to create a half wavelength difference in the two sides of the loop.

In the situation of the "1-3, 2-4" feed scheme with resonant loops, the voltage magnitudes at the two resonant loops feed points must be equal to obtain equal antenna currents. By connecting a decoupler between the two resonant loop feed points, phase-independent decoupling can be obtained. The connection of a decoupler at each feed point creates a 5-way connection point.

The final elements that control the loop phasing are a 3 dB hybrid for power splitting the feed from the transmitter and a 360˚ phase shifter which sets the overall phase difference between the two 5-way crosses. The 3 dB hybrid is optimized to split the power from the transmitter over the 60 to 120 MHz band.

In addition to the main elements described above for phase control, other transmission line components were included for personnel safety, ease of testing, and improved reliability. Some of these components are (1) test sections which allow the high power center conductor to be quickly removed and linked to two type N connectors; (2) a four-port coaxial switch to allow the transmitter to be switched from the 3 dB hybrid to the dummy load; (3) gas barriers which allow the system to be pressurized to 3 atm (each transmission line has five separate zones) with insulating gas such as dry air, N_2, or SF_6; (4) dc breaks which electrically isolate the transmitter and impedance matching equipment from the DIII–D torus and torus hall (the dc break's standoff is 30 kV continuous and the rf insertion loss is less that 0.05 dB); and (5) flex sections to allow for thermal growth and installation mismatch.

4.3.4. ANTENNAS

The outer midplane locations of the three fast wave antennas are indicated in Fig. 4-7. The first antenna (F1) has been used in many experimental campaigns, with a variety of Faraday shield configurations, including two tests with no shield. The other two antennas (F2A and F2B) are identical and were installed in 1994. Spectral calculations have been performed with the RANT3D code. Figure 4–7 also indicates the location of the rf pick up probes and the microwave reflectometers.

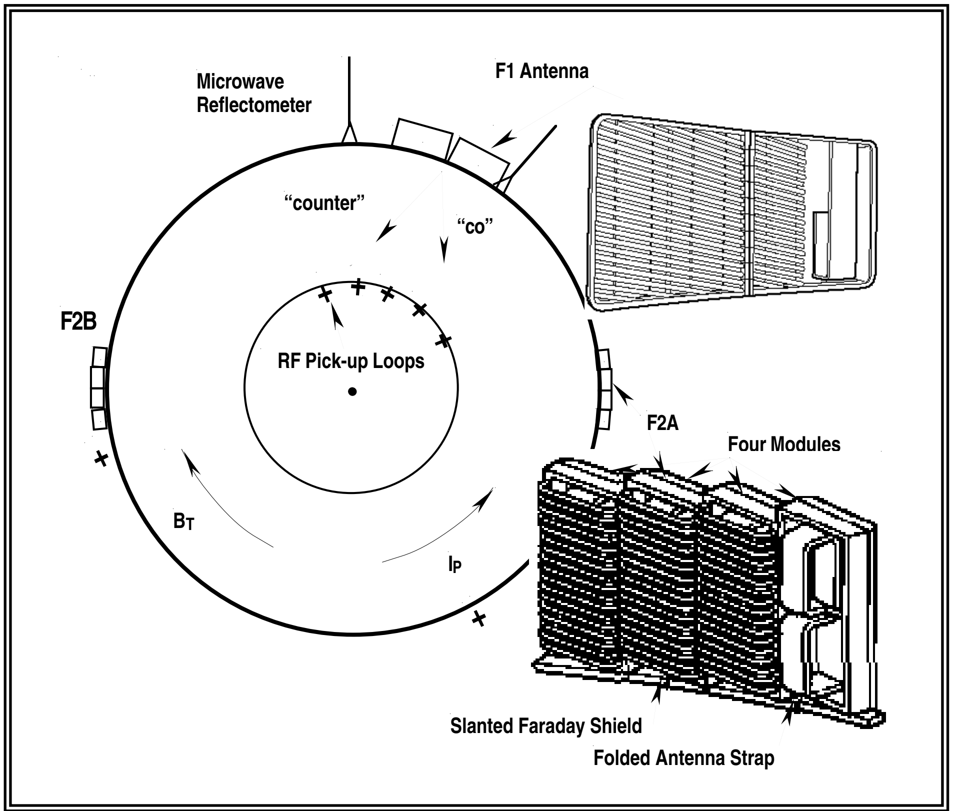


Fig. 4–7. Layout of DIII–D showing the location of the F1 and two F2 antennas, the rf pick up probes and the reflectometer horns.

The Faraday shield rods for all antennas are slanted at 12 deg relative to the midplane as shown in Fig. 4–7, to better match the magnetic field line pitch at the shields. All plasma facing wall surfaces on DIII–D are armored with graphite tiles which provide protection for the Faraday shields which are situated 1 to 2 cm behind the surface of the tiles at the outer midplane. All shields have the plasma facing sides coated with boron carbide, although the F1 antenna has been operated for some experiments with no shield at all, and also with no boron carbide coating on the shield, that is, the inconel substrate facing the plasma. The boron carbide coatings have performed satisfactorily provided that the coating is thin enough (< ~100 μm). Thicker coatings cracked and eroded during plasma operation, apparently due to lack of thermal contact to the base shield rod material. DIII–D routinely uses boronization for wall conditioning.

4.4. 3 MW, 110 GHz ECH System

4.4.1. INTRODUCTION

To support the AT operating regimes in the DIII–D tokamak, methods need to be developed to control the current and pressure profiles across the plasma discharge. In particular, AT plasmas require substantial off-axis current in contrast to normal tokamak discharges where the current peaks on-axis. On DIII–D electron cyclotron current drive (ECCD) is being employed to drive the off-axis current in AT plasmas. The first phase of the ECH is 3 MW of electron cyclotron heating power. This involves the installation of three gyrotron systems operating at 110 GHz, the second harmonic resonance frequency on DIII–D, with each system generating nominally 1 MW. The three systems will use one GYCOM (Russian) gyrotron and two CPI (formerly Varian) gyrotrons, all with windowless evacuated corrugated low loss transmission lines. The second phase of the ECH upgrade, initiated in FY99, is to add three more gyrotrons to increase the ECH system power to 6 MW. The final phase would increase the power to 10 MW.

4.4.2. RF SYSTEM OVERVIEW

Three gyrotrons have been installed and operated on the DIII–D tokamak. One, a GYCOM gyrotron has been in operation since 1996, the second a Communications and Power Industries (CPI) gyrotron model VGT-8011A has been in service since May 1997. Both gyrotrons are nominally 1 MW at a central frequency of 110 GHz. Although each gyrotron is designed for long pulse capability (>10 s), their present pulse capability is limited to 2 s and 0.8 s, respectively, owing to the output windows currently installed upon the tubes. The GYCOM gyrotron uses a BN edge-cooled window, and CPI uses a double-disk sapphire window design with an inert Chloro-fluorocarbon (FC-75) coolant flowing between the two disks. The third gyrotron is also a CPI gyrotron with basically the same internal configuration as the tube with the double disc window but has been built with a CVD diamond window. This gyrotron offers the first practical design of 1 MW cw operation. The gyrotron performance parameters are shown in Table 4–3.

TABLE 4–3
GYROTRON PERFORMANCE PARAMETERS

	GYCOM	CPI No. 1	CPI No. 2
Frequency, GHz	109.8 – 110.15	110	110
Output window design	Single disk	Double disk	Single disk
Material	BN	Sapphire	CVD Diamond
Cooling method	Edge cooled	Face cooled	Edge cooled
RF power (kW) and pulse duration (s)	(960)/2.0	(350)/10 (530)/2 (1000)/0.8	(1000)/10 Effectively cw
Efficiency, %	38.0	32.0	32
Beam voltage, kV	72.0	80.0	80
Beam current, A	33.8	35	35

The transmission line for all three systems uses 31.75 mm diam aluminum circularly corrugated waveguide carrying the HE_{11} mode. The waveguide diameter represents a compromise between power handling capability and the desirability that the transmission line be insensitive to misalignment, thermal growth, and motion. For the first two gyrotrons, the rf beam exiting the side of the gyrotron is a modified gaussian beam with a flattened profile to minimize the peak temperature and thus the stresses in the window. Since the window and rf beam are made as large as practical, ≈100 mm, to reduce the thermal load on the window, direct coupling from the gyrotron into the waveguide is impractical and an interface device is required. Using two mirrors, the rf beam exiting the gyrotron is phase corrected to restore it to a free-space Gaussian and is then focused to couple into the 31.75 mm waveguide diameter. These mirrors are housed in a mirror optics unit (MOU) which also contains a water-cooled resistive load, which absorbs any stray rf power that exits the gyrotron at an angle that cannot be focused into the waveguide. The output beam of the diamond window gyrotron is a free-space Gaussian, but is larger in diameter, ≈ 51 mm, than the 31.75 mm diam of the waveguide, so a MOU will also be needed to focus the beam into the waveguide.

An entire single transmission line system, shown in Fig. 4–8, consists of 6 mitre bends and is ≈40-m long with an estimated 2% loss in the waveguide and 0.6% loss per mitre bend. The mitre bend losses are from mode conversion 0.5% and ohmic losses 0.1%. The waveguide is evacuated to a pressure of $\approx 1 \times 10^{-5}$ Torr by a turbomolecular pump at the MOU and a similar pump on a special section of waveguide near the tokamak, where the waveguide has been slotted to allow pumping between the corrugations. This waveguide pumping section is placed as close to the DIII–D vacuum vessel as practical so that any impurities evolving from the waveguide upstream of the tokamak can be pumped out before they reach the plasma and possibly contaminate it. For the FC–75 cooled CPI gyrotron, a fast shutter located just upstream of the pumping section has been installed. This shutter can close faster than the pressure wave can travel down the waveguide and, in conjunction with the pumping section, maintains the vacuum pres-

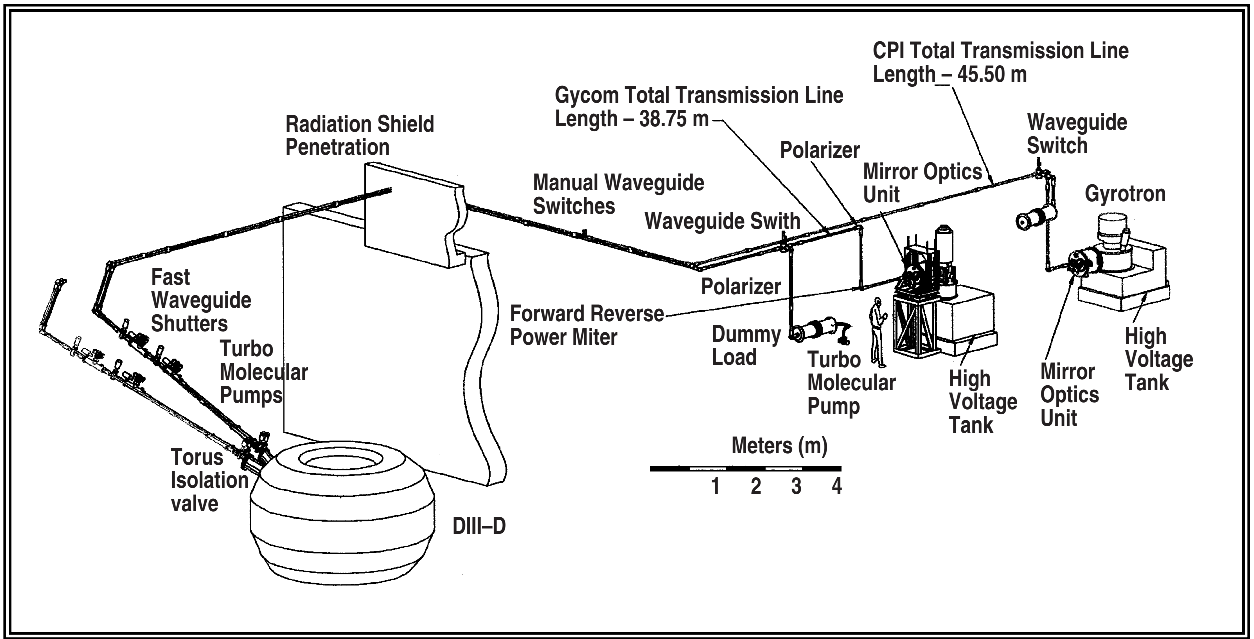


Fig. 4–8. ECH system layout showing the routing of the transmission line and the location of the major transmission line components.

sure at the tokamak entrance waveguide. To aid in gyrotron optimization, a dummy load is connected to the system via a waveguide switch located near the gyrotron. Polarization control is achieved by a set of grooved motorized polarizing mirrors mounted in two of the mitre bends. By appropriate rotation of these two mirrors, any elliptical polarization desired can be obtained.

There are two sets of dual launchers attached to DIII–D. Each launcher is composed of two mirrors; a focusing mirror and a flat tilting mirror. In the first set, the flat mirrors are permanently angled at 19 deg off normal to provide the appropriate current drive injection angle. The flat tilting mirror rotates vertically so the injected beam can be steered poloidally from slightly below the midplane to the outermost top edge of the plasma. On the second set of launchers, the tilting mirrors are aimed normal to the plasma. The gyrotrons can be interchanged between these two launcher sets by the simple task of breaking the connection at the second-to-last mitre bend and adding or removing a 2 m length of waveguide.

5. SUPPORT SERVICES

5.1. QUALITY ASSURANCE

Fusion Quality Assurance (QA) engineers, inspectors, and support personnel maintained a high level of activity during 1998. Significant projects supported were the repair of the E–coil solenoid leak, Radiative Divertor tooling and Ring panels, optical alignment of the ORNL High Field Pellet Injector, Tile and Bumper Limiter replacement at 225 degrees, and the installation and alignment of the ECH Launcher waveguides at the 255 degree, R+1 port.

The Fusion Quality Assurance group performed the following specific jobs:

1. Reviewed and approved all DIII–D design drawings, specifications, procedures, and procurement requisitions. Participated in design reviews and chaired the Material Review Board (MRB).
2. Performed receiving inspections, source inspections, and measurements of purchased and fabricated material, parts, subassemblies and assemblies.
3. Revised and released for use 14 DIII–D Work Procedures. The Work Procedures describe how key tasks in the DIII–D program are carried out. The Work Procedures were revised to make them better describe actual work practices and to be consistent with higher level procedures and the Fusion Group QA Manual and Fusion Group Procedures. In addition, Chapter 1 of the Fusion Group QA Manual and the Quality Assurance Program Document for the DIII–D program were revised to reflect recent organizational and procedural changes.
4. Completed the semi-annual building concrete footing and building column settlement surveys; no unexpected subsidence was detected. An inspection of the cracks in the concrete walls adjacent to the machine was also performed with no noticeable changes noted.
5. The Continuous Improvement Committee continued following-up on previously completed Continuous Improvement Opportunity (CIO) forms to verify that the actions taken were effective.

During the period 1988 through 1998, the Fusion Quality Assurance group performed all relevant Quality Assurance/Quality Control tasks required to support the DIII–D Program. During this period, several events and associated Quality Assurance activities were particularly noteworthy.

- Routinely assisted the project with precise optical alignments of equipment and diagnostic experiments to ensure optimum performance of the devices.
- Occasionally assist project personnel including collaborators in obtaining as-built measurements in and around the machine. In addition, they periodically perform reverse engineering of modified or experimental parts.

- Developed an instruction to provide guidance for the assembly of a data package for each diagnostic installed on the DIII–D machine. The data package serves as a reference document for any individual interested in a specific diagnostic and is made up of documents generated during the design, fabrication, and installation of the diagnostic.
- Implemented the Continuous Improvement Committee established by the Senior VIce President. Continuous Improvement is a means by which every member of an organization continually examines all of the processes, procedures, methods and activities by which work is accomplished and suggest constructive changes.
- Provided training to over 200 Fusion Group personnel including collaborators in selected requirements of the Fusion Group QA Manual, Fusion Group Procedures, and the DIII–D Work Procedures.
- Issued 24 Fusion Group Procedures. The Fusion Group Procedures implement the requirements specified in the Fusion Group QA Manual.
- Developed and released a new Fusion Group QA Manual complying with DOE Order 5700.6C. Project personnel were major contributors to the manual.
- Fusion QA initiated trend reports on supplier performance in an effort to increase awareness of good and poor performance.
- Since the majority of Fusion manufactured components are produced locally by a limited number of smaller machine shops, Fusion QA has adopted a policy of conducting periodic in-process inspections during fabrication. This practice has resulted in discovering machine setup problems, design drawing misinterpretations, machine programming errors, and machinist errors in sufficient time to prevent scrapping the part.
- Fusion QA inputs inspection and acceptance information into the Dun & Bradstreet Millennium Accounts Payable/Purchase Order System. In addition to providing a database of inspection activities, the system requires QA acceptance of articles to be inspected before the supplier is paid, which has decreased the number of submitted parts not meeting design requirements.

5.2. PLANNING

The Planning group supported operation and maintenance of the DIII–D facility. Planning and Control provided long-term program planning, as well as day-to-day scheduling (cost control, preparation of Field Work Proposals, and Cost and Fee Proposals), processing of purchase requests, expediting and reporting of status. These support activities are essential to constraining the program within prescribed budgets and schedules. Our planning activities (budget, schedule, resource) enabled us to maximize the utilization of available resources for accomplishment of program goals and were important in planning and replanning of scope, budget, and schedule with fluctuating funding levels.

Major planning activities during 1998 included work on ECH and radiative divertor upgrades. Over the period of this contract, major planning activities included work on ECH, radiative divertor, diagnostics, E–coil leak repair, and ICRF.

5.3. ENVIRONMENT SAFETY AND HEALTH

5.3.1. FUSION AND DIII–D SAFETY

The fusion safety program provides for the safe operation of the DIII–D facility and for a safe working environment for employees and visitors. Special programs address high voltage and high current, high vacuum systems, ionizing radiation, microwave radiation, cryogenics and the use of power equipment and machine tools. Therefore, a special two volume Safety Manual set was written entitled "Doublet III Safety Procedures for Facility and Equipment Operation;" volume #1 contains policies and volume #2 contains procedures. These two manuals point out the specific safety rules and operation procedures to be adhered to while at the DIII–D site. Both manuals refer to the company safety manuals for the general safety and environmental rules and regulations. DIII–D is provided support by GA's Licensing, Safety and Nuclear Compliance organization and GA's Human Resources Safety organization in areas such as health physics, industrial hygiene, environmental permitting, hazard communication, hazardous waste, and industrial safety.

Fusion has established a Safety Committee in accordance with company policy as a means of focusing on and addressing both the numerous safety issues faced daily and longer range safety needs and goals. The Fusion Safety Committee is comprised of representatives from various departments within the Fusion Group, including management, supervisors, operators and technicians. Its chairman is the DIII–D Associate Program Director; the vise chairman is a manager from one of the organization in fusion and the secretary is the Fusion Safety Officer. The Safety Committee meets twice a month to address safety activities and concerns of the Fusion Group such as hazardous work requests, radiation work authorizations, accident/incident reports, near misses, equipment malfunctions, accident avoidance programs, supervisor involvement, training, inspections, access control procedures and high voltage hazards. The Safety Committee also solicits specialized help from any of the five fusion safety subcommittees during reviews of lasers, electrical systems, vacuum systems, the use of chemicals or cryogens.

In addition to the Fusion Safety Committee's oversight of activities at DIII–D, two individuals are dedicated full-time to on-site "preventive" safety involvement. Their activities include writing and reviewing procedures, developing and conducting special training classes, conducting inspections and follow-up, maintaining safety equipment and calibrating safety monitoring devices, MSDSs and hazardous waste collection sites, interfacing with GA's Licensing, Safety and Nuclear Compliance organization and HR Safety organization, and providing continuous oversight of fusion employees, collaborators, visitors and contractors to assure compliance with established safety policies, procedures and regulations.

The DIII–D Emergency Response Team (ERT) consists of individuals involved directly with maintenance and operation of the DIII–D equipment. They are trained in cardiopulmonary resuscitation (CPR), first aid, use of self-contained breathing apparatus (SCBA) and the use of fire extinguishers, evacuation and crowd control and facility familiarization. The team can respond within seconds to provide immediate assistance until outside emergency assistance arrives. Internal emergency drills are conducted annually and drills in conjunction with outside emergency responders and hospitals are conducted biannually. These drills have included extrication of a dummy from inside the DIII–D Vacuum vessel and a simulated diborane release. Local outside emergency responders are given a biannual tour of DIII–D with an explanation of our emergency response systems.

5.3.2. INSPECTIONS

Safety inspections are conducted throughout the year to promote an active Hazard Prevention Program. The inspections are conducted by a combination of Fusion, GA Licensing, Safety and Nuclear Compliance or GA/HR Safety personnel and outside consultants. A report is provided to the Fusion Safety Committee, where corrective action assignments are made. Table 5–1 lists inspections for the DIII–D tokamak.

TABLE 5–1
INSPECTIONS FOR THE DIII–D TOKAMAK

Inspection	Frequency
Site inspection (GA)	Monthly
Slings and lifting equipment (GA)	Quarterly
Fire extinguishers (GA)	Monthly
Stationary fire equipment	Yearly
Electrical consultant	Biannually
CAL/OSHA Consulting Service	Yearly
Insurance carrier inspection	Yearly (multiple)
S.D. City Fire Department (CEDMAT)	Yearly
GA Safety Committee Hazards Survey	Yearly
NRC	Yearly
DOE/OAK Safety Review Inspection	Yearly

The Fusion Safety Officer is responsible for tracking the progress of all inspection discrepancies and ensuring resolution.

All new employees, collaborators and long term visitors must go through a thorough and comprehensive one–on–one indoctrination by the Fusion Safety Officer and the Pit Coordinator. They are informed of the specific potential hazards that are present daily at DIII–D and the special safety precautions and rules that apply, with specific emphasis on the areas where they will be working. The necessary training classes are recommended due to the information gained at the indoctrination. Individual contractors and subcontractors assigned to work at the DIII–D facility also receive a similar indoctrination.

5.3.3. TRAINING

Training is all-important to the safety of both personnel and equipment. Due to the complexity of the DIII–D site and its operation, numerous safety-training classes are conducted. DIII–D treats all long-term visitors and collaborators just like regular GA Employees and includes them in all required safety training. Listed below are examples of the various classes given:

- Confined Space Entry
- LockOut/TagOut
- Fall Prevention and Protection
- Laser Safety
- Hazard Communication
- Hazardous Waste Disposal
- CPR
- Radiological Safety
- Back Injury Prevention
- Ergonomics
- Emergency Response Team Training
- Working Safely and Effectively in the DIII–D Pit
- Fork Lift Operator Training
- Crane and Rigging Training
- Electrical and High Voltage Safety
- High Potential Testing Safety
- Power Tool Safety
- Cryogenic Safety
- General Laboratory Safety

A stationary power tool training program is in effect which requires that any individual that uses a power tool attached to the floor must be trained in the safe use of that tool. This is accomplished by training videos, written tests and hands-on verification of proficiency by shop supervisors.

5.3.4. OTHER ACTIVITIES

As a result of an accident involving a Japanese scientist at Lawrence Livermore National Laboratory (LLNL) in February 1992, the U.S. DOE and Japan formed a Joint Working Group. The U.S./Japan Joint Working Group (JWG), led by Steve Rossi from the Office of Fusion Energy, met at the DOE headquarters in Germantown, Maryland in October 1992 and in Tokyo and Mito City, Japan in January 1993 to discuss the issues of safety of inter-institutional collaborations. Subsequent visits to the U.S. were in 1993, 1995 and 1997 and to Japan in 1994, 1996. The group consists of four U.S. and four Japanese safety representatives from national laboratories, private contractors and universities. The Fusion Safety Officer represents both GA and the U.S. industrial contractors. A Japan/English Safety manual for JAERI was generated with the help of individuals from both sides of the JWG.

The Fusion Safety Officer conducted collaborative safety visits for the University of Wisconsin-Madison in an effort to share safety knowledge within DOE laboratories and Universities.

The Fusion Safety Officer attended or participated in the following groups or training sessions/seminars:

- DOE Tiger Team Training
- PPPL Tiger Team
- ORNL Machine Guarding
- DOE Contractors Safety & Health Conference
- Inter-laboratory Safety Group
- Cal/OSHA Update
- National Safety Council Conference on Safety & Health (6 conferences)
- 14th Annual Symposium on Fusion Engineering (ES&H co-chair)
- Occupational Safety & Health Training Institute (six different courses)
- DOE Contractors Fire Safety Conference

A safety check-in form was instituted to ensure that all visitors and collaborators receive the correct indoctrination and information upon arrival at DIII–D.

A Hazardous Work Authorization (HWA) review and signature sheet was instituted to ensure that when an HWA is approved by the Fusion Safety Committee, each person on the personnel involved list is informed of the requirements and procedures listed in the HWA.

A Safety Library was established to enable fusion employees' better access to all relevant safety policies, procedures and requirements. A current file of all MSDSs and the chemical inventory is also available in the library.

6. DIII–D COLLABORATIVE PROGRAMS

6.1. DIII–D COLLABORATIVE PROGRAM OVERVIEW

DIII–D is a national fusion facility with approximately 50 participating institutions with approximately 300 scientific users spending more than one week at DIII–D each year. These collaborations of national and international organizations carries out the scientific research of the DIII–D Program Plan. This DIII–D collaborative research spans a wide range of activities with a wide range of facilities and institutions as described below.

6.2. LAWRENCE LIVERMORE NATIONAL LABORATORY

The Lawrence Livermore National Laboratory (LLNL) Collaboration with DIII–D started in 1986 with a focus on divertor physics, divertor modeling, current drive and diagnostic tasks. At least half of the group lived in San Diego and some commuted; this ratio has grown over the year to nearly two-thirds of the staff at DIII–D in 1998. The first LLNL diagnostic was an Infrared TV system to measure divertor heat flux, which resulted in some of the first direct measurements of heat flux on a diverted tokamak. A visible TV camera (with spectral filters to look at deuterium radiation) and fast ion gauges were added. These data were compared with the UEDGE model of the divertor plasma. Research was started on reducing the divertor heat flux by increasing divertor plasma radiation; this is the currently accepted divertor "solution" for the ITER design. Since that time, several new divertor diagnostics have been developed and installed by the collaboration, most notably the divertor Thomson scattering system and an imaging bolometer system which is the only divertor system in the world. Very low electron temperatures and high electron densities are measured during radiative divertor operation, which has verified the code predictions that other physical processes, such as volume recombination and convective heat flux, are important in the divertor. Unique tomographic reconstruction algorithms were developed for the bolometer and visible camera data so that the total and particular radiation constituents have been measured. An absolutely calibrated extreme ultraviolet spectrograph was recently installed and has determined that the major constituent of radiation is carbon radiation (at 1550 Å). A collaborative effort with Hampton University has obtained the first 2–D pictures of this radiation. The UEDGE code has been continuously developed and benchmarked against the data. The most recent development is the inclusion of particle drifts and the comparison of data from the UCSD group. LLNL also played a lead role in the development of the radiative divertor hardware which has been recently installed. This has allowed particle control in highly shaped plasmas.

The LLNL collaboration has been active in two other areas: tokamak improvements [advanced tokamak (AT)], and remote site collaborations. The focus of the AT research has been the measurement and control of the current profile with a 35-channel motional Stark effect (MSE) diagnostic. This is arguably the best system in the world and measures the plasma current density as a function of radius. This has been a key measurement in the improvement of tokamak confinement and in the study of negative or weakly sheared profile plasmas. The current profile is now routinely calculated at nearly 200 time points between discharges and changes are made in the neutral beam timing and other parameters to optimize the profile. In the future, the DIII–D team plans to actively control the current profiles and may use real-time feedback with the MSE diagnostic. Contributions have also been made in the area of transport, particularly in the area of electron transport.

The goal of the work on the remote experimental site is to facilitate experiments by personnel located both at and away from the DIII–D site. In many cases, these tools help both sides of the collaboration. One example is the use of off-site computers to process MSE and magnetics data which are available to all researchers between shots. Remote experiment demonstrations have been carried out linking LLNL to DIII–D and Alcator C–Mod.

6.3. OAK RIDGE NATIONAL LABORATORY

The ORNL collaboration in the past ten years has focused on MHD studies, particle transport and control, pellet injection, and radio frequency (rf) physics and technology.

ORNL staff took a leading physics role in the area of MHD equilibrium, stability, and plasma performance and lead to two significant milestones in tokamak research. The first was the high beta achievement (average beta of 11% and central beta of 43%) using dynamic plasma shaping, which showed the access to second stability regime in the plasma core. The second experimental campaign was the high performance D–D operation (high equivalent D–T fusion yield of $Q_{DT} = 0.3$) in DIII–D, demonstrating neoclassical ion heat transport over the entire profile.

ORNL's work in the particle transport and control area began in collaboration to design the advanced divertor structure. ORNL staff installed pressure gauges and helped design the divertor cryopump and played key roles in the first set of density control experiments in H–mode. ORNL staff led particle balance studies and showed that divertor pumping could replace He GDC to maintain low recycling conditions in ELMing H–modes. The ORNL pellet injector was installed in 1994 and has been used in particle transport and plasma optimization studies and for disruption mitigation studies using impurity pellets. These experiments have enabled DIII–D to operate density above the Greenwald density limit with good H–mode confinement. A novel approach to mitigate against disruptions using a massive gas puff has also been developed.

The ORNL group investigated helium transport in the core and in the edge plasma and demonstrated the exhaust of helium from ELMing H–mode plasmas with a pumped divertor in DIII–D, indicating adequate helium ash removal in ITER. It has also been demonstrated that strong plasma flows in the scrape-off layer can effectively entrain impurities in the divertor. Radiative divertor plasmas using puff-and-pump and radiating mantle plasmas (i.e., RI–mode) have been produced. Properties of the core transport and possible use of edge stability control were investigated. Spectroscopic studies of the divertor plasmas have investigated impurity and plasma flows as well as volume recombination. First studies on the effects of recycling neutrals on transport barriers and confinement have revealed a correlation between the H–mode threshold and the penetration of neutrals into the core plasma.

ORNL designed and built the three fast wave current drive antenna arrays currently on DIII–D. The first four-element array was installed in 1990 and has been in use ever since. In that time, there have been three different Faraday shields used with this antenna. The second and third arrays, installed in 1994, are identical and were designed for long-pulse operation, incorporating water cooling. The design of these antennas resulted from extensive modeling and experimental development at ORNL. These antennas have been used for the first tokamak experiments with fast wave current drive, demonstrating efficient on-axis current drive. ORNL maintains these antennas and leads the collaborative effort for rf operations on DIII–D.

6.4. SANDIA NATIONAL LABORATORY

The collaboration between Sandia National Laboratories (SNL) and GA on the DIII–D tokamak has centered on physics and technology of the tokamak boundary and plasma/wall interaction zone. This involved work in materials testing and thermo-mechanical engineering analysis, plasma wall interactions, design and implementation of advanced plasma diagnostics, scrapeoff layer (SOL) and divertor physics, and plasma edge modeling. Three SNL groups at both the Albuquerque and Livermore laboratory sites and full and part time SNL personnel at DIII–D performed this work. Measurements at DIII–D provide important information for the fusion program studies at SNL.

SNL expertise in high heat flux technology helped optimize DIII–D performance. The graphite walls in DIII–D are a mosaic of tiles engineered to handle high heat and particle flux including resistance to thermal shock. Tile testing for GA at SNL's High Heat Flux Facility in Albuquerque was instrumental in choosing the best design and material for use in DIII–D. The high heat flux engineering and diagnostic expertise at SNL developed boundary diagnostics for DIII–D including embedded target plate Langmuir probes and two fast reciprocating Langmuir probes designed to quickly plunge into the plasma edge.

Edge and divertor measurements helped SNL to explore key plasma physics issues. The array of 28 target plate probes provides density, temperature, and particle flux measurements across the divertor floor, at the two baffle entrances, and in the upper divertor. These measurements are especially important for detachment and particle balance studies. SNL implemented and operates this diagnostic.

SNL performed plasma/material interaction studies on the interior walls and divertor plates of the tokamak using specially developed techniques and instruments. Metal deposition measurements on the graphite tiles by in situ beta back-scattering showed net deposition at the inner strike point. Specially prepared, removable Divertor Materials Exposure System (DiMES) samples were exposed to well-diagnosed tokamak divertor plasmas and subjected to extensive surface analysis to determine erosion and redeposition. ITER erosion and redeposition models were corrected to incorporate the DIII–D erosion and redeposition measurements.

6.5. PRINCETON PLASMA PHYSICS LABORATORY

Physicists from Princeton Plasma Physics Laboratory (PPPL) have taken part in the DIII–D scientific program for a number of years, largely through interactions between individuals at PPPL and GA. This traditional collaboration typically involved analysis of magnetohydrodynamic (MHD) stability, micro-instability, and transport in specific DIII–D discharges. In FY97, after shutdown of TFTR, the PPPL/DIII–D collaboration underwent a major expansion and participation was further increased in FY98. PPPL is now one of the principal collaborating institutions in tokamak research at DIII–D.

PPPL has contributed to the DIII–D program in three principal areas. (1) PPPL scientists have directly participated in the DIII–D experimental campaign. (2) PPPL has provided physics support through scientific programming and code development, diagnostic engineering and technician effort, and design and implementation of new diagnostic capability. (3) PPPL has provided operations support through provision of full-time technical personnel and the development or procurement of hardware necessary for the DIII–D research program.

Beginning in FY97 and continuing in FY98, PPPL physicists led or participated in a number of important investigations. These included experiments which produced high performance plasmas with steady-state internal transport barriers, experiments in which Resistive Wall Modes were reproducibly observed, using saddle loop sensors installed by PPPL, and an experiment to investigate sawtooth stabilization by minority ICRF heating and by neutral beam injection.

The highest priority diagnostic task for the PPPL/DIII–D collaboration during FY98 was an upgrade of the Thomson scattering systems. The previous systems, with horizontal viewing of vertical laser beams, could not access the central core of DIII–D plasmas. As part of a joint effort with LLNL and GA, PPPL designed, fabricated, and installed critical in-vessel and ex-vessel components for the Tangential Central Thomson Scattering System and participated in system tests.

Active feedback control of MHD instabilities in DIII–D is a major goal of AT research and a principal area of concentration for the PPPL/DIII–D collaboration. Of particular interest is control of the Resistive Wall Mode, which limits the achievable pressure in high performance plasmas. In support of this important program, PPPL initiated the physics and engineering design of a nonaxisymmetric feedback coil project in FY97, installed a new six-coil set of saddle loop sensors in FY98, and is now procuring a power supply for active feedback stabilization experiments. The saddle loops easily detected the onset and evolution of Resistive Wall Modes during the 1998 experimental campaign. PPPL is supplying feedback power supplies in FY99.

PPPL technical support of the DIII–D facility has been especially effective. A PPPL operations engineer, an RF power engineer, and an RF technician played crucial roles in recent experimental campaigns and also conceived and implemented a number of improvements in the facility. An outstanding example of such improvements is a novel modification of the patch panel hardware for the field shaping coils. This allows the field shaping coils to be reconfigured much more quickly than before to greatly enhance experimental flexibility.

In support of the DIII–D ECH program, PPPL began development of an improved, steerable ECH/ECCD launcher in FY98. The first launcher to be completed in FY99 will be able to remotely control the toroidal and poloidal injection angles of two gyrotrons. This will allow DIII–D physicists, for the first time, to change the direction and radial location of electron cyclotron current drive on a shot-to-shot basis and will greatly improve the ability to distinguish the effects of heating and current drive.

6.6. UNIVERSITY OF CALIFORNIA, LOS ANGELES

The University of California, Los Angeles (UCLA) collaboration led by Dr. A. Peebles has been a member of the DIII–D research team since 1988 and has had a continuous on-site presence since then, typically comprised of both researchers and students. During this period, UCLA has contributed to the very successful DIII–D program in a number of areas, including the development of the paradigm of E×B flow shear suppression of turbulence in H–mode, VH–mode, and NCS regimes, general turbulence and transport studies, the physics of ICRF plasma heating, edge and core density profiles and divertor studies. In addition, UCLA brings to DIII–D an international reputation and expertise in the development and application of advanced millimeter-wave and far-infrared diagnostics. In FY98, the UCLA program was supported by a total of 5 FTE with 4 FTE on-site at DIII–D.

In general, the overall role of UCLA within the DIII–D Research program has been concentrated in four main strategic areas:

1. UCLA has made major scientific contributions to increasing understanding of the transport and turbulence properties of the advanced confinement operating regimes on DIII–D. For example, measurement of turbulent correlation lengths, and temporal evolution of fluctuating quantities, temperature have aided in revealing the major role that E×B shear suppression of turbulence plays in all of the AT confinement regimes.
2. UCLA has also made significant contributions to the wave particle and boundary physics areas on DIII–D.
   a. In the wave particle area, reflectometry has been utilized for the first time in a nonperturbing, local determination of the rf electric field profile associated with externally launched fast waves in a hot fusion plasma. This work was strongly collaborative with the ORNL and GA groups.
   b. In boundary physics, UCLA has made major contributions to improved understanding of the L–H transition, has developed reflectometry systems to measure both density profile and turbulence properties in the divertor, and has investigated in-out asymmetries in edge turbulence.
3. UCLA has developed, fully demonstrated, and integrated advanced millimeter-wave/far-infrared diagnostic systems into the DIII–D National Facility. The diagnostic systems developed for DIII–D have broad application and relevance to other devices within the restructured fusion energy sciences program.
4. Finally, UCLA has played the major role on DIII–D in the education of students in plasma physics, fusion science and millimeter/FIR technology. Over the past 10 years, four outstanding UCLA Ph.D. theses have been generated through participation in the DIII–D Program. Two of these directly led to Invited Talks for the students (a rare occurrence) at the APS Plasma Physics Division Meetings.

Finally, it should also be noted that interaction with the theoretical community has grown significantly over the years that UCLA has participated in the DIII–D Program. This close coupling between the theoretical, experimental and operational communities is one of the major reasons for the success of the UCLA participation in the DIII–D National Fusion Facility.

6.7. UNIVERSITY OF CALIFORNIA, SAN DIEGO

The University of California, San Diego (UCSD) collaboration in the DIII–D Program began in 1989 as part of the Institute of Plasma and Fusion Research and the Mechanical, Aerospace, and Nuclear Engineering (MANE) Department at UCLA. The program led by Prof. R. Conn relocated to the UCSD in 1994, becoming part of the newly formed Fusion Energy Research Program. The initial program, funded by the Transport Initiative, was a collaborative effort with SNL to construct and operate a fast reciprocating Langmuir probe array on the outboard midplane of the DIII–D tokamak for the study of tokamak boundary [edge and scrapeoff layer (SOL)] physics, edge turbulence, and L-to-H transition physics. Over the next ten years, the scope of the collaboration was broadened to include: (1) a second fast reciprocating Langmuir probe in the lower divertor to study the physics of main ion parallel and perpendicular flows, and detached and/or highly radiating divertor plasmas; (2) a disruption research program, including runaway electron generation and detection, and mitigation of disruption effects; and (3) a subcontract from GA to measure erosion rates of plasma facing materials in DIII–D using the DiMES actuator.

These research activities are organized into the following four subtasks:

1. **Edge Physics and Turbulence.** The principal objectives of this subtask are the investigation of plasma edge turbulence and transport, and the impact of turbulence and convective flows on the plasma edge and SOL; edge electron temperature fluctuations, and thermal transport; and detached and highly radiating divertor and boundary (RI–mode) plasmas.
2. **Disruption Physics and Mitigation.** The principal objectives of this subtask are the experimental investigation of DIII–D disruptions and mitigation techniques, runaway electron production, magnetic fluctuation effects on disruption runaways, two-color IR diode detector for runaway synchrotron radiation, and development of DISRAD, a fast disruption AXUV radiometer diagnostic.
3. **UCSD/GA/TEXTOR Collaboration, Radiating Boundary.** The principal objective of this subtask is to characterize the edge in RI–mode discharges in both limiter (TEXTOR) and elongated/divertor (DIII–D). Recent results include:
   a. Analysis of fluctuation and turbulent transport changes in RI–modes at TEXTOR.
   b. Evaluation of turbulence changes in the DIII–D boundary with radiative mantles.
4. **Divertor Materials Erosion Probe.** The principal objectives of this subtask are to perform, under subcontract to GA, DiMES materials experiments and operations, plasma analysis, measurement of materials erosion rates under conditions relevant to ITER and model validation in collaboration with GA, SNL, and Argonne National Laboratory.

6.8. UNIVERSITY OF CALIFORNIA, IRVINE

The focus of the UC Irvine research program led by Prof. W. Heidbrink has been the physics of non-thermal “fast” ions. These energetic ions are produced through neutral-beam injection, ion-cyclotron heating, and fusion reactions. UC Irvine has developed and operated a suite of fast-ion diagnostics and has led physics studies of the thermalization, confinement, and stability behavior of fast-ion populations.

Diagnostic efforts include fusion-product measurements of 2.5 and 14-MeV neutrons and 15 MeV protons, neutral-particle analysis of beam ions and ions accelerated by rf, and bolometric measurements of escaping beam ions.

Thermalization of deuterium beam ions was studied by injecting short neutral beam pulses into DIII–D and measuring the subsequent decay of the 2.5 MeV neutron emission. Excellent agreement with classical Coulomb scattering theory was obtained (to within ~15% accuracy). This “beam-blip” technique was subsequently employed on TFTR to measure the diffusion of beam ions and on JT–60U to measure the effect of toroidal-field ripple on beam-ion confinement.

The confinement of fusion products was consistent with classical expectations in the absence of MHD activity but was severely degraded in the presence of strong MHD modes. Losses of beam ions were observed during Alfven eigenmode activity and in plasmas with large tearing modes.

The toroidicity induced Alfven eigenmode (TAE) was studied extensively. The mode was first observed concurrently on TFTR and DIII–D in 1991. Large losses of up to 80% of the injected beam power were carefully documented. The stability threshold was an order of magnitude higher than initially predicted; the discrepancy was subsequently attributed to additional damping mechanisms. The instability saturated in a burst cycle (relaxation oscillation) caused by the expulsion of beam ions. Efforts to compare the eigenfunction with ideal MHD theory revealed significant discrepancies, indicating the importance of kinetic effects.

An equally virulent instability (dubbed the “BAE”) was observed at lower frequencies than the TAE; this mode is still poorly understood theoretically. Modes with frequencies that chirp down rapidly were also studied as were fishbones and sawteeth. A DIII–D team led by UC Irvine performed a TFTR D–T experiment that searched for alpha-driven BAEs.

6.9. UNIVERSITY OF TEXAS

The University of Texas (UT) Fusion Research Center (FRC) has been collaborating with DIII–D since October 1996. The FRC augments the existing capabilities of the DIII–D group by operating diagnostics for use in both the DIII–D core program and in specific FRC experiments. In particular, the group operates and maintains an electron cyclotron emission (ECE) radiometer, provides analyzed ECE data, and assists with the operation of a beam emission spectroscopy (BES) system and analysis of its data. Also, UT staff propose specific experiments to test models of turbulence and transport.

Over the last two years, progress has been made in several collaborative research areas. FRC members participated in experiments to test theory-based transport models and analyzed BES fluctuation data for same. In addition, data obtained during other experiments was used to evaluate transport changes associated with low-order rational q surfaces in the plasma. Progress was made as well in diagnostic development. Improvements have been made to the ECE radiometer resulting in greater reliability and better calibration stability. New software has been developed for ECE analysis including a program to use ECE data to constrain the equilibrium reconstruction code EFIT. New data analysis techniques have been applied to BES data to look for evidence of self-organized criticality. The FRC has plans for an upgrade of additional channels to the ECE diagnostic and increased participation in transport experiments.

6.10. COLUMBIA UNIVERSITY

Scientists from the Columbia University (Profs. Navratil, Mauel, and a postdoctorate student) have participated in experiments on DIII–D. The primary areas of collaboration have been in obtaining high performance with negative central shear and in stabilization of resistive wall modes. This collaboration helps connect the DIII–D research with the experiments on High Beta Tokamak — Extended Pulse (HBT–EP), which is a small tokamak at Columbia University presently studying the issue of wall stabilization. Part of the HBT–EP research program involves the use of nonaxisymmetric coils to force plasma rotation. The idea of forcing plasma rotation by rotating magnetic islands in the plasma originated at GA by T. Jensen. The collaboration presently consists of one full-time scientist permanently located at DIII–D in addition to active involvement on a part-time basis of other Columbia scientists in the planning and operation of the relevant experiments.

6.11. UNIVERSITY OF WISCONSIN

The University of Wisconsin (UW) collaboration on DIII–D has focussed on two major research topics central to magnetic fusion research: MHD studies, led by Prof. James Callen, and turbulence studies, under the direction of Prof. Raymond Fonck.

The MHD program began in 1995 and has involved five scientists, post doctoral researchers, and graduate students. Central thrusts of the MHD research have included neoclassical tearing modes, magnetic island physics, and disruption precursors. Neoclassical tearing mode studies included theory and simulations on the stabilization of these modes through the use of localized current drive and heating, dynamics of the MHD trigger and the effects of geometry and aspect ratio on pressure-driven magnetic islands. Also, UW has assisted in numerous GA led experimental studies on long-pulse beta limits in tokamaks. Magnetic island physics has involved the interpretation of internal fluctuation measurements during island growth; also, the delta-prime parameter has been obtained (both theoretically and experimentally) to determine the classical and neoclassical stability of the modes. Further, the distortion of magnetic islands resulting from the combination of flow shear and viscosity has been examined. Disruption studies have provided a model for the temporal evolution of an ideal MHD instability as it is driven slowly through its stability boundary. The spatial structure of observed precursors on electron temperature measurements via ECE are currently being compared with that predicted by the GATO ideal MHD stability code.

The turbulence program was initiated in 1995 and has resulted in the development of a multichannel density fluctuation diagnostic and the resulting analysis and interpretation of plasma turbulence measurements. This program has involved several scientists, graduate students and a technician as well as assistance from GA. A state-of-the-art BES system has been deployed on the DIII–D tokamak. Eight channels were operating near the end of 1995 with an additional eight added by mid-1996. In early 1997, the system was brought up to 32 channels. 2–D (radial, poloidal) fluctuation measurement capability has been implemented recently with initial measurements obtained.

6.12. UNIVERSITY OF MARYLAND

The collaboration between GA and the University of Maryland began in the mid-1980s. The primary focus of the General Atomics-University of Maryland collaboration led by Prof. R. Ellis and a postdoctorate student at DIII–D is the calibration, maintenance, operation, and improvement of the Michelson interferometer diagnostic system. This instrument, a baseline diagnostic, provides absolutely calibrated measurements of the electron temperature profile via ECE in DIII–D plasmas. These measurements are widely used by the DIII–D research team. Since the Michelson views the plasma along a horizontal chord, it is also known as the Horizontal ECE (HECE) system.

The original purpose of the Michelson interferometer was to measure nonthermal ECE emission from DIII–D along a vertical viewing chord. A result of this early work was the study of changes in the electron distribution function during ECH and current drive. The diagnostic was later converted to its present horizontal viewing chord for measuring the electron temperature profile. Since that time, several upgrades and improvements have been made. The lens-relay system which had been used to transmit the plasma light to the Michelson was replaced by a much more efficient and robust waveguide assembly. A technique to extract the temperature profile from the third-harmonic Michelson data was implemented, which has proven to be useful when the second harmonic data are “cutoff” due to high plasma densities. Measurements with the HECE system have determined the wall reflection coefficient for microwaves in a graphite-tiled tokamak to be 0.7.

Several projects are currently being pursued by the Maryland collaborators. The primary responsibility of Maryland scientists is the continued operation and improvement of the Michelson interferometer diagnostic system. Research is ongoing to find the cause of inconsistencies in ECE profile measurements in NCS plasmas — one of the results of this effort will be the use of Michelson data to constrain MHD equilibria (EFITs). The Maryland staff will add a new ECE diagnostic to DIII–D in 1999. This diagnostic will be a flexible fast channel that has many potential applications, such as determining the spectrum of synchrotron radiation produced by runaway electrons following plasma disruptions and measuring ECH temperature effects via third harmonic emission.

6.13. 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 many foreign collaborating activities. The DIII–D Program has played, and will continue to play, a lead role internationally in the AT research thrust. The flexibility of the DIII–D device allows early testing of new approaches that can, if successful, later be implemented on the larger tokamaks such as JET and JT–60U. DIII–D scientists have participated in such experiments on foreign machines transferring techniques developed on DIII–D. Working with foreign tokamaks of various sizes, DIII–D has played a key role in developing the dimensionless parameter approach to the scale size dependence of plasma confinement. The path of developing AT approaches on DIII–D and confirming those approaches on the larger foreign tokamaks will provide the scientific basis for use of AT operating modes on future international or domestic next step machines.

For example, in developing the reversed or NCS regime for use on JET, a strong interaction of the JET and DIII–D research staff took place. JET scientists came to DIII–D and JET-shaped plasmas were operated in DIII–D. In those plasmas, the techniques for timing and application of neutral beam heating were developed that allowed internal transport barrier regimes with negative central shear to be created. Later, about six DIII–D scientists went to JET and participated in the initial D–D experiments at JET in which NCS internal transport barrier plasmas were created in JET as preparation for the DT campaign. DIII–D personnel also participated in the DT campaign.

DIII–D and JAERI have had a long interactive history in AT studies, particularly around the issue of plasma shape, specifically triangularity. This interaction began early in the DIII–D Program with the discovery of the Type II or grassy ELM regime when high triangularity plasmas were operated in DIII–D. Stability analysis by JAERI scientists showed the edge of such plasmas was predicted to be in the second stability regime from which sprung today's intensive effort to understand the stability of the H–mode pedestal region. More recently, DIII–D scientists have first assisted in implementing on JT–60U the wall conditioning techniques developed on DIII–D for discharge optimization and, secondly, DIII–D scientists have assisted in developing on JT–60U plasma control approaches that allowed higher discharge triangularity. These higher triangularity discharges in JT–60U have better beta limit and ELM properties.

The EFIT equilibrium code has been exported from DIII–D to most tokamaks around the world, and has played a prominent role in the design and analysis of their AT experiments. The EFIT code was implemented on JT–60U by DIII–D scientists and used to analyze internal transport barrier discharges, as well as to deduce the radial profile of noninductively driven current in discharges that have not resistively relaxed to a new steady-state current profile. A similar use of EFIT has been made on Tore Supra as part of a larger collaboration on advanced methods of plasma control. Experts in DIII–D's digital plasma control system have participated in work on Tore Supra.

The shaping flexibility of DIII–D has enabled DIII–D to match the plasma shapes run in other tokamaks, e.g., JET, ASDEX Upgrade, Alcator C–Mod. This ability to run identical plasma shapes has enabled dimensionless parameter scaling studies of confinement to be carried out between DIII–D and these three tokamaks. The results have provided a more sound basis of projecting confinement to future devices, in particular, to ITER.

DIII–D began a detailed program of investigation of the effects of magnetic field errors on performance and, in particular, on plasma rotation. This work was expanded into collaborative work on the larger tokamak JET and the smaller tokamak COMPASS. The resulting three machine database has provided a scaling law that has been used to estimate error field problems on ITER. This collaborative work has continued onto the subject of neoclassical tearing modes which frequently appear in AT regimes.

The role of DIII–D in developing the principle of E×B shear suppression of turbulence as the reason for the confinement improvement in H–mode is well known. The detailed focus on the plasma edge made by DIII–D has motivated other tokamaks to mount new diagnostics focused on the plasma edge. The result has been a present intensive worldwide effort to understand the structure and physics properties of the H–mode shear layer. This work has high leverage on the overall performance of the tokamak because of the sensitive dependence of the overall confinement on the height of the H–mode pedestal; a dependence that results from "stiff" transport models for the core plasma predicted by theory.

Collaborations have been carried out with JET in England, ASDEX–Upgrade in Germany, Tore Supra in France, and JT–60U and JFT–2M in Japan through bi-lateral agreements. In addition to the benefits gained from DIII–D staff assignments in these and other laboratories, foreign scientists visiting DIII–D have made significant contributions to DIII–D Program goals. A summary of some of the recent major international collaborations by DIII–D staff members is given below.

JET (England) is the large European tokamak approximately twice the size and magnetic field strength of DIII–D. Our collaboration with JET is one of the largest of our international collaborations. Several DIII–D scientists from GA, LLNL, and ORNL collaborated at JET in a two-step exchange on NCS-type high performance tokamak discharges. First a series of experiments were performed on DIII–D with participation by JET scientists. Then DIII–D scientists participate in experiments at JET. In other experiments, DIII–D and JET scientists have carried out dimensionless scaling experiments to investigate fundamental confinement properties and to provide results to the ITER database. The results from this series of experiments were very successful. Ion temperatures of about 30 keV and electron temperatures of about 16 keV were obtained. These results represent some of the highest fusion parameters attained in deuterium plasmas on JET.

TEXTOR (Germany). The RI–mode is an AT mode first discovered on TEXTOR. Its prominent features are confinement of H–mode quality or better with densities above the Greenwald limit and a radiated power fraction approaching 100%. Because the diagnostic instrument set on TEXTOR is limited, the TEXTOR group have initiated a collaboration on DIII–D on RI–mode plasmas. Because of the excellent edge diagnostics set on DIII–D, it is expected that a deeper understanding of the physics of the RI–mode might be obtained on DIII–D.

Tore Supra (France). The primary emphasis for our collaboration with Tore Supra has been ECH physics and technology, noninductive current drive, and plasma and current density profile control. The Tore Supra program includes electron cyclotron, fast wave, and lower hybrid heating and current drive research which complements the DIII–D electron cyclotron and fast wave heating and current drive research.

ASDEX Upgrade (Germany). The ASDEX/DIII–D collaborations are primarily in the area of Divertor research and RF. This includes impurity transport and heat transfer mechanisms involving Edge Localized Modes and MARFEs. Research also includes H–mode confinement studies. RF collaboration includes ICRF and ECH.

JAERI (Japan). The DIII–D Program is a major collaboration between the U.S. and Japan and was implemented by DOE and JAERI. JAERI contributed substantial financial resources and manpower from 1979 to 1986. A U.S./Japan Doublet III Steering Committee meets annually to assess progress and review future plans. GA scientists participated in an exchange at JT–60U, working in the area of NCS, high confinement, and neutral beam current drive. The equilibrium reconstruction code, EFIT, was used to analyze JT–60U NCS configurations. A successful experiment using the GA-designed “Combline” antenna was carried out on the JAERI JFT–2M tokamak. This antenna allows better coupling to the plasma over a wide range of plasma parameters. The highly successful JAERI/DIII–D cooperation continues. This collaborative program entails the long term participation of JAERI scientists in the DIII–D Fusion Research Program.

Russia. DIII–D maintains a broad collaboration program with several Russian Fusion Research Institutes. DIII–D and T–10 tokamak scientists have collaborated closely for a number of years. For several years, DIII–D provided subcontract funds to carry out ECH experiments on T–10 which was equipped with more microwave gyrotron power than was the case at the time on DIII–D. With the TRINITI lab at Troitsk, near Moscow, the main topics are: materials for plasma facing components, divertor spectroscopy, and the use of the Russian developed DINA code for modeling dynamic plasma behavior. With Kurchatov, collaborations were on ECH, electron temperature measurements using an x–ray spectrometer, and remote analysis of charge exchange recombination data. Funded by the Theory Grant, GA is contracting with Moscow State University to support Russian theorists to perform theoretical analysis of plasma physics problems of relevance to understanding the performance of DIII–D.

China. The main thrust of our exchanges with the Chinese Fusion Research Program has consisted of the long term participation of Chinese scientists in the DIII–D Program at the DIII–D site. These exchanges have concentrated in the area of the Thomson scattering diagnostic, the CER diagnostic, and ECH systems.

Korea. The proposed Korean superconducting tokamak (KSTAR) has many similarities in shape and size to DIII–D. DIII–D scientists and engineers have contributed to the preliminary physics and engineering design of this device. It is expected that the collaboration with KSTAR and the Korean fusion program will increase as KSTAR design, construction, and operation progresses.

ITER Experts. The ITER Expert groups provide the forum for prioritizing, coordinating, and communicating tokamak planning, scientific research nationally and internationally. As part of the effort to satisfy the ITER physics research and development needs, seven ITER Expert Groups were established. The Expert Groups, identify ITER research needs within their area of expertise and propose research programs, including suggestions for specific facilities. DIII–D program staff participated in the ITER Physics Committee, chair the Divertor Expert Group, and are members of the Confinement and Transport Group, the Confinement Modeling and Database Group, the Divertor Modeling and Database Group, and the Disruptions, Plasma Control, MHD Group. Many other DIII–D scientists have enthusiastically participated in these groups via extensive presentations, written contributions, database inputs, and participation at working sessions.

7. PUBLICATIONS FY98–FY89

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APPENDIX A
THEORY

HIGHLIGHTS OF DIII–D THEORY 1992–1998

GA has a long history of contributions to fusion plasma theory. We believe there are numerous examples where our theories applied to DIII–D have lead to key experiments and improved tokamak performance. Below we summarize some of our accomplishments from 1992 to 1998

- Demonstrated how sheared toroidal rotation may significantly increase the beta limit for high-n ideal ballooning modes.
- Showed that the low-n external kink mode in a rotating plasma appears to be wall-stabilized even with a resistive wall.
- Correlated the MHD dispersion relation for toroidal Alfvén eigenmodes (TAE) and beta-induced Alfvén eigenmodes (BAE) with catastrophic loss of fast ions in DIII–D neutral beam experiments.
- Developed new insight into how the fundamental processes determining the size and field strength scaling of confinement can be obtained from dimensionally similar tokamak discharges.
- Developed a novel, nonlinear ballooning mode method for numerically simulating three-dimensional homogeneous turbulence in toroidal geometry and applied it to determine the dependence of transport on shear, safety factor, toroidicity, and sheared rotation.
- Demonstrated how a theory based on E x B rotational shear driven by changes in the diamagnetic flows at the plasma edge can explain both the transport bifurcation from L– to H–mode and the further edge confinement improvement in the DIII–D VH–mode.
- Setup a transport profile database for DIII–D data and developed a fast shooting transport code for rapidly testing transport models and assigning a statistical fit.. This later became the forerunner of similar ITER databases and testing codes.
- Motivated many aspects of the DIII–D Advanced Divertor Program by pointing out the role of grad-B directed divertor flows in determining the H–mode power threshold and how poloidal electric fields induced by biasing the divertor plates can be used to control divertor recycling flows and pumping.
- Developed (in collaboration with experimentalists) several steady-state relevant advanced tokamak scenarios with both favorable MHD stability and improved confinement: Second stable core VH–mode, global second stable high poloidal beta, and high-ℓ_i discharges.

- Developed RF current drive and heating codes to show how these advanced tokamak discharges may be accessed in steady state and how bootstrap current and more efficient noninductive current drive concepts may be possible.
- In collaboration with experimentalists, planned, performed, and validated for the first time fast-wave current drive in a tokamak
- Developed generalization of the s-a model for local MHD equilibrium to include finite aspect ratio, elongation, and triangularity.
- Developed new theory for a resistive MHD energy principle in shaped geometry.
- Developed models for active external coil and loop active feedback stabilization of MHD modes.

APPENDIX B
JCT SECONDEES

The ITER Joint Central Team (JCT) was created in 1993. Per agreement with DOE, specially qualified personnel were selected by ITER management to be seconded to the JCT co-center at which their particular expertise was required. While on secondment, the GA employees were fully dedicated to the ITER project and under the direction of the JCT management. The specific employees seconded included:

Individual	Title/Responsibility	Secondment Initiated	Secondment Terminated	Co-Center Location
Charles Ahlfeld	Head of Engineering Division	6/96	11/98	San Diego
Robert Bourque	Cryogenic Systems	9/93	7/98	Naka
Edward Bowles	Power Supply for Plasma and Field Control	10/95	4/98	Naka
Remy Gallix	Assistant Head of Superconducting Coils & Structures Division	4/95	11/98	Naka
Randy Hager	Remote Handling	6/94	11/96	Naka
Doug Holland	Fusion Safety Analysis	2/95	1/97	Naka
Roy Little	System Integration	8/95	6/98	San Diego
Fred Puhn	Head of Design Integration Division	10/92	3/98	San Diego
Doug Remsen	Manager of Radio Frequency Heating Systems	11/93	1/95	Garching
Marshall Rosenbluth	Chief Scientist	1/95	11/98	San Diego
Kurt Schaubel	Pumping and Fueling Mechanical Analysis	1/96	7/98	Garching
Robert Schleicher	Group Leader of Plant Systems Group	11/95	11/98	San Diego
Peter Smith	Assistant Head, Design Integration Division	12/92	4/98	San Diego
Julie Van Fleet	External Relations	12/92	12/94	San Diego
John Wesley	Physics Analysis	1/93	1/99	San Diego

The majority of the GA secondees terminated their secondments before or during FY98.

APPENDIX C
USER SERVICE CENTER

The User Service Center (USC) was funded under the DIII–D contract during FY92 through FY95 in a specified budget category. This is a report on the main activities accomplished during this period.

The USC has always been linked very closely with the DIII–D Computer Systems. Much of its work was done in concert with and in support of the DIII–D program. This work has been reported in the DIII–D Computer Systems section. Work beyond the direct DIII–D support includes general user support for non-DIII–D DOE projects (i.e., APT, ITER), network access, remote collaboration support, local NERSC users support, and unclassified computer security compliance in addition to the day-to-day hardware and software maintenance and upgrades.

The USC is a major node on the ESnet backbone, and as such, provides Wide Area Network access for GA/DOE programs. The USC coordinated the installation and transition to upgraded ESnet routers and faster networks. The USC personnel worked getting reliable network connectivity to Russia (Moscow) which was required for the DIII–D collaboration with the Kurchatov Institute. The USC also worked with the DOE/SAN (now DOE/OAK) unclassified security office to improve computer access for DOE-approved collaborators from sensitive countries. The USC improved the Local Area Network, moving everyone to ethernet and installing a FDDI backbone between the main servers.

During this report period, the USC transitioned the users from the VAX/VMS operating system environment to the UNIX environment. A large, 3-CPU HP computer was installed as were many HP, Digital and SGI workstations. USC staff provided all system administration and general user support for these computers. Special funds were received from the Office of Scientific Computing for procurement of a Desktop Video Conferencing computer. This system enabled GA/Fusion to participate and successfully complete the Remote Experimental Environment project, a collaboration with LLNL, ORNL and PPPL. The USC also integrated the Auspex File Server into the local computer environment making all user home areas and data computational areas available on all UNIX systems.

USC staff provided considerable software support and user help. Many tools were written to help users move into the UNIX environment. Extensive documentation was written. When appropriate, tutorials and basic reference manuals were also provided. The production version of the ONETWO MHD Transport is maintained by USC staff.

APPENDIX D
ITER DIAGNOSTICS

The development of a Fastwave Reflectometer diagnostic was funded through the DIII–D contract in FY97 by the ITER and Technology Group within DOE.

The goal of the task was to test the ability of a reflectometer to identify species mix (D/T ratio) within a plasma for application to ITER. Reflectometer hardware was installed on DIII–D, the hardware was tested, and plasma data was taken. Results indicated that the coupling between the transmitting antennae and the plasma worked fine, however, the coupling between the receiving antennae and the plasma was too small to get good results. Although it was felt such a diagnostic would ultimately accomplish the desired measurements, it would require a complete rebuild of the receiving antennae.

APPENDIX E
HYBRID TUNER

E.1. INTRODUCTION

In order to maximize the power coupled to the plasma from the Fast Wave ICRF transmitter systems, the antenna-to-plasma impedance must be kept well matched. Any variation from the match setting results in reflected power that either poses a threat to the integrity of the transmitter or as a minimum becomes wasted power sent to the dummy load. In general, the antenna-to-plasma match is reasonably stable as long as the plasma edge conditions can be kept stable. Unfortunately, this condition is not meet for ELMing H–mode plasmas — an area of plasma research which is of great interest to the DIII–D program. With this in mind, the DIII–D RF Group embarked upon several efforts to devise a transmitter antenna matching system that would be more tolerant of changes in edge conditions, one of these schemes was to use a Hybrid Tuning System (HTS). In 1995, a contract was placed with Advanced Ferrite Technologies (AFT) to provide an HTS that could operate at 60 MHz, at the 2 MW level.

E.2. HYBRID TUNER SYSTEM (HTS)

The HTS is an electronically controlled matching device to match an ICRF transmitter to a dynamic load impedance, such as presented by an antenna facing a tokamak plasma. The dynamic nature of the plasma requires a device which can follow a time varying load impedance. In the HTS this is accomplished by using stubs with planar geometry transmission lines which have ferrite material glued to the center conductor of the line. By rapidly varying the magnetic field penetrating the ferrite, using currents in coils driven by a switched mode transistor supply, the characteristics of the line are appropriately modified to create the desired match. The HTS achieved an unprecedented matching speed of approximately 2 ms in high rf voltage conditions, which is the capability needed for most tokamak operations. A schematic of the HTS is shown in Fig. E–1.

In the purchase order given to AFT for this prototype HTS it was specified that acceptance tests could be done at an appropriate laboratory other than at DIII–D since the rf power capability to do these tests does not exist at AFT itself. There are several reasons for the allowance of an external laboratory for acceptance testing. First, the tests at DIII–D would create an extra burden on the technical staff normally fully subscribed to rf operations on DIII–D, and could take away from the ability to support experiments.

Second, if any difficulties were to arise in acceptance testing, then returning the HTS to the factory in Germany from DIII–D for repair would stretch the schedule. Third, the ICRF staff at IPP has been extremely interested in this technology also and they were willing to support these tests with their own labor. This is how the acceptance tests came to be at IPP, under an arrangement developed between IPP and AFT.

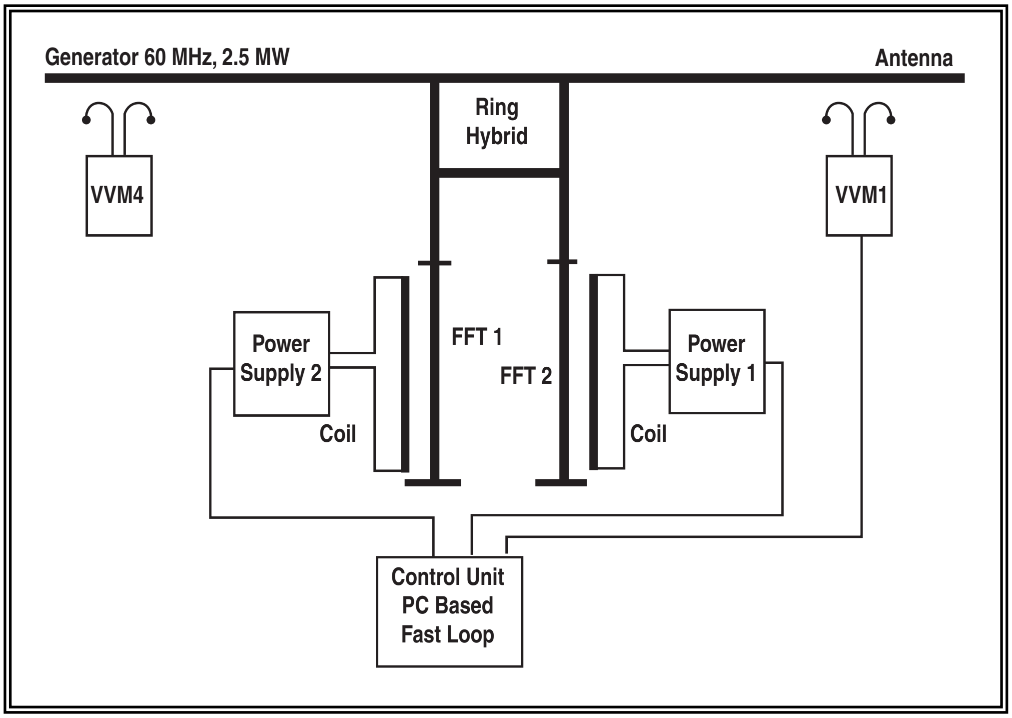

Fig. E–1. Hybrid tuner: basic.

E.3. ACCEPTANCE TEST

The HTS demonstrated power handling capability of 1.7 MW for a 9 s pulse at the design frequency of 60 MHz. This is the maximum limit for the IPP transmitter at this frequency. Using a spark gap in the transmission line to cause a sudden change in the impedance of the load seen by the HTS, on a submillisecond timescale, the HTS demonstrated a complete matching time to the arc of approximately 2–3 ms. This was done with a short 20 ms transmitter pulse; otherwise, the energy in the arcing spark gap would completely melt portions of the transmission line around it. Figure E–2 shows the forward and reflected power for such a case, where the arc strikes with the application of rf power. The reflected power (lower trace) drops to a low value as the HTS feedback seeks the match. There are low level undulations on the reflected power after the first match is achieved, due to the HTS following the slight variations in the arc impedance as the arc moves around the sharpened tip of a bolt inserted into the end of a section of transmission line. The cycle time of the feedback computer is about 100 μs, so this is the amount of time required to send out the next command to the power supply. The case in Fig. E–2 was with 300 kW of transmitter power with an unmatched reflection coefficient of 0.9 in magnitude.

The experience with the transmission line arc for the acceptance tests provided a graphic illustration of the need for reliable arc detection in conjunction with such a fast tuning system. The HTS itself has very sensitive photon detectors viewing the ferrite lines which shut down the transmitter upon sensing any light from arcing in these stubs. However, an extended transmission line system, such as is used on ASDEX-U or DIII–D, is difficult to monitor completely with photon sensors. The ASDEX-U system is using the detection of broadband rf noise below the operating frequency as an arc detector, together with some reliance upon standard reflection signals from directional couplers.

The HTS is a nicely constructed package, occupying about 20 cubic meters of physical volume. Figure E–3 shows a photograph of the HTS. The two black cylinders near the floor house the ferrite loaded strip lines. The HTS is completely controlled by a computer program running in real-time on an IBM-type PC. The switch mode power supply occupies one standard laboratory rack, not shown in Fig. E–3.

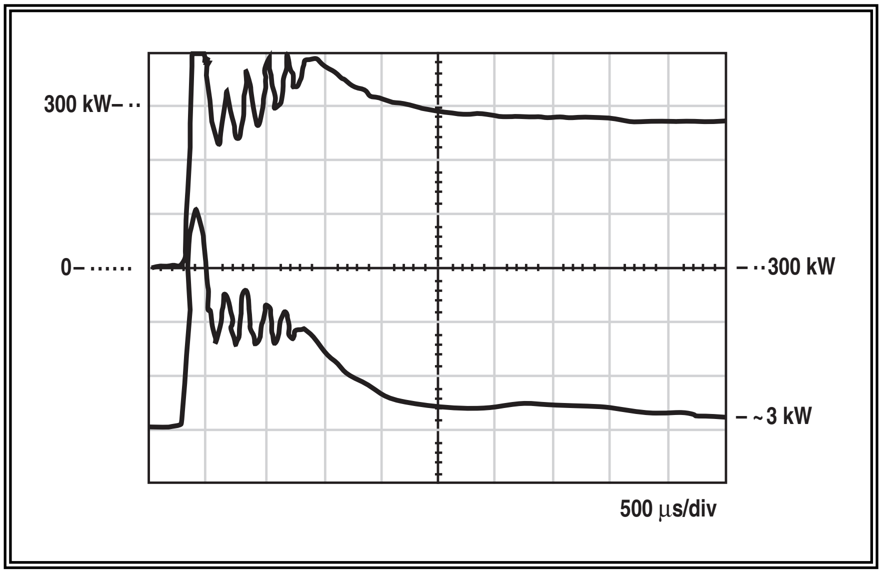

Fig. E–2. HTS matches the transmitter to an arc on the spark gap. Upper trace: forward transmitter power. Lower trace: power reflector to the transmitter.

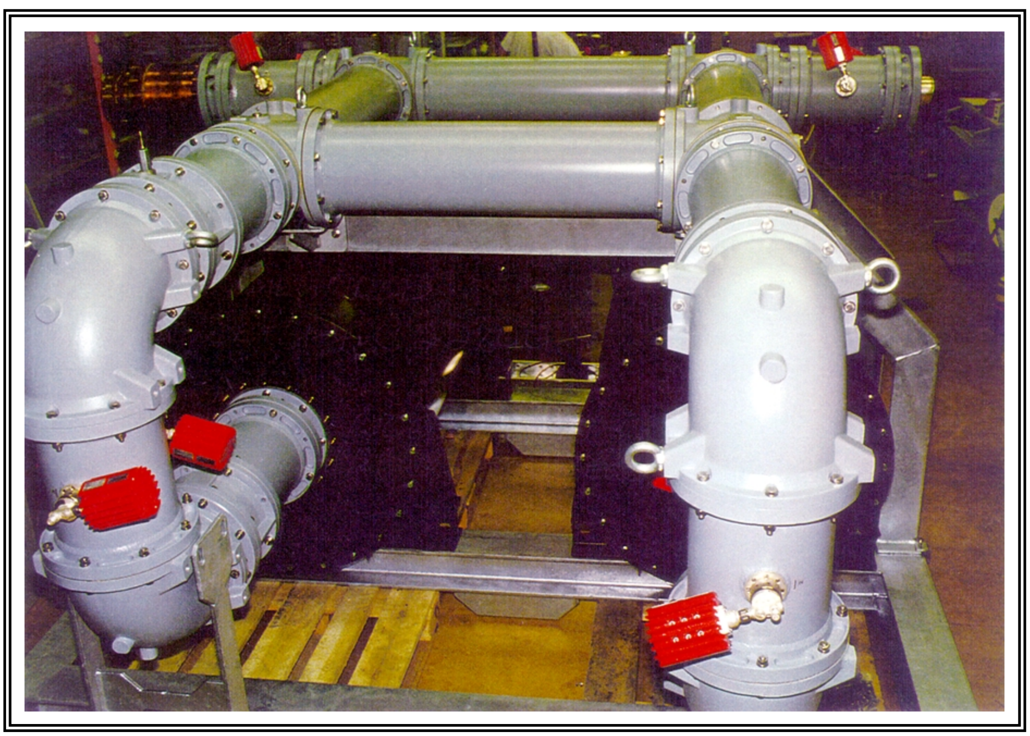

Fig. E–3. Photograph of the HTS.