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DIII-D RESEARCH OPERATIONS

ANNUAL REPORT

**OCTOBER 1, 1993
THROUGH
SEPTEMBER 30, 1994**

GENERAL ATOMICS

GA-A21935
UC-420

DIII-D RESEARCH OPERATIONS ANNUAL REPORT TO THE U.S. DEPARTMENT OF ENERGY

OCTOBER 1, 1993 THROUGH SEPTEMBER 30, 1994

by
**PROJECT STAFF
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Work supported by
U.S. Department of Energy
under Contract Nos. DE-AC03-89ER51114,
W-7405-ENG-48 and AC05-84OR21400

**GENERAL ATOMICS PROJECTS 3466, 3467, 3470, 3473
DATE PUBLISHED: JULY 1995**

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1. DIII-D PROGRAM OVERVIEW

1.1. INTRODUCTION

Energy production using controlled thermonuclear fusion is on the threshold of scientific realization. Throughout the industrialized world, the requirement for non-polluting, high energy density power sources which do not require expenditure of non-renewable resources has been recognized and is being addressed. An international program of research has developed greater understanding of the physics and technology of magnetic fusion which is leading to substantial improvements in the performance of present day experiments and is demonstrating the technical feasibility of controlled fusion in advance of the next generation devices such as the International Thermonuclear Experimental Reactor (ITER).

In aerodynamics, despite aggressive research programs driven by national defense requirements, the supercritical wing, which enhanced performance and efficiency, was only recently discovered. Similar advances are being made in magnetic fusion research. The exact nature of the plasma geometry providing the best stability, efficiency and reliability is being investigated to reduce the capital cost and increase the economic competitiveness of fusion power stations. Although such factors as increased concentrations of CO₂ in the atmosphere and the future depletion of cheap fossil fuels will change the economic equation in favor of fusion, it is nevertheless critical to demonstrate fusion at the lowest possible cost and with the highest possible margin for ignition. To this end, the DIII-D project is expanding the tokamak operational envelope and providing new physics and engineering knowledge.

The DIII-D tokamak research program is managed by General Atomics (GA) for the U.S. Department of Energy (DOE). Major program participants include GA, Lawrence Livermore National Laboratory (LLNL), Oak Ridge National Laboratory (ORNL), and the University of California together with several other national laboratories and universities. The DIII-D is a moderate sized tokamak with great flexibility and extremely capable subsystems. The primary goal of the DIII-D tokamak research program is to provide data for development of a conceptual physics blueprint for a commercially attractive fusion power plant. In so doing, the DIII-D program provides physics and technology R&D output to aid the International Thermonuclear Experimental Reactor (ITER) and the Princeton Tokamak Physics Experiment (TPX) projects. Specific DIII-D objectives include the achievement of steady-state plasma current as well as the demonstration of techniques for radio frequency heating, divertor heat removal, particle exhaust and tokamak plasma control. The DIII-D program is addressing these objectives in an integrated fashion in plasmas with high beta and with high confinement. The long-range plan is organized with two principal elements, the development of an advanced divertor and the development of advanced tokamak concepts. These two elements have a common goal: an improved demonstration reactor (DEMO) with lower cost and smaller size than present DEMO concepts. In order to prepare for this long-range development, in FY94 the DIII-D research program concentrated on three major areas: Divertor and Boundary Physics, Advanced Tokamak studies, and Tokamak Physics.

Each of these areas by itself constitutes an ambitious program of research. The major goals of the Divertor and Boundary Physics studies are the control of impurities, efficient heat removal and understanding the strong role that the edge plasma plays in the global energy confinement of the plasma. During Advanced Tokamak studies, investigations of new techniques for improving energy confinement, controlling particle fueling and increasing plasma beta were initiated. These techniques include control of the current density profile by various techniques, development of advanced plasma shapes with improved stability and confinement and fueling in part with solid pellets. Included in the category of Tokamak Physics studies are the understanding of energy and particle transport in a power plant relevant plasma, the study of heating by radio frequency techniques and non-inductive current generation.

The versatility of DIII-D permits a wide variety of plasma shapes to be studied. Experiments during 1994 have taken full advantage of this capability. Cross sectional views of DIII-D (Fig. 1.1-1) show a diverted configuration in which the magnetic equilibrium forms a so-called double-null geometry, a single null configuration and a unique small plasma for scaling studies. All of these plasmas were used during the past year in various experiments and illustrate the wide variety of possible shapes which can routinely be produced. To indicate the extensive diagnostic coverage in the machine, soft X-ray and bolometer viewing chords are superimposed on the left and central figures respectively and the points at which the multipulse Thomson scattering system takes data are on the right. Because of this flexibility in shaping and positioning of the magnetic equilibrium and with the development of real time feed back control, a broad range of plasma parameters can be varied under control in real time during a discharge. The DIII-D tokamak characteristics, maximum plasma parameters and experimental results are summarized in Tables 1.1-1 and 1.1-2.

Fig. 1.1-1. Cross sections of DIII-D with flux surfaces of a double-null divertor, single-null divertor and a small specialized discharge superimposed. Soft X-ray viewing chords are superimposed on the double-null equilibrium, bolometer chords on the single-null and Thomson scattering points are on the small equilibrium.

**TABLE 1.1-1
DIII-D CAPABILITIES**

	Present
Vacuum vessel volume	37 m ³
Major radius	1.67 m
Minor radius	0.67 m
Maximum toroidal field	2.2 T
Vertical elongation ratio	2.6
Available ohmic heating flux	12 V-s
Maximum plasma current	3.0 MA
Neutral beam power	20 MW
RF power (ECH)	Upgrade to 2 MW at 110 GHz in progress
RF power (ICRF)	6 MW installed,
Current flattop (divertor at 2 MA)	5 s

**TABLE 1.1-2
DIII-D ACHIEVED PARAMETERS
(NOT SIMULTANEOUS, τ_E IS GLOBAL)**

I/aB	3.3 MA/m/T
$\beta_T(0)$	44% (second stable core)
$\langle\beta_T\rangle$	12.5%
H factor	4
β_N	6
β_P	5.2
$\varepsilon\beta_P$	1.8
$\bar{n}_e$	$1.4 \times 10^{20} \text{ m}^{-3}$
$T_e(0)$	7 keV
$T_i(0)$	22 keV
W	3.7 MJ
τ_E	0.4 s, ($P_{\text{HEAT}} = 4\text{MW}$)
$\bar{n}_e(0) \tau_E$	$0.39 \times 10^{20} \text{ m}^{-3} \text{ s}$
$n_D(0) T_i \tau_E$	$5 \times 10^{20} \text{ m}^{-3} \text{ keV s}$
H-mode duration	10.3 sec

To carry out the DIII-D program, GA provides a combination of extensive institutional experience in the field, a skilled and internationally recognized staff and a unique experimental facility — the DIII-D tokamak. GA has had an active fusion research program for more than three decades. It is the only industrial participant in the U.S. program with a major, integrated effort in all aspects of plasma physics and fusion research, from basic plasma

theory, diagnostics, magnetic fusion device design, engineering, construction, and operation, to fusion reactor technology. In 1994, four DIII-D scientists, Lang Lao, Ron Stambaugh, Ted Strait and Tony Taylor, were presented the Award for Excellence in Plasma Physics Research by the Plasma Physics Division of the American Physical Society for their work on high beta equilibria.

Collaborations with other U.S. and foreign fusion programs are an essential feature of research on DIII-D. These cooperative efforts significantly broaden the base of support for the DIII-D program and provide the human and hardware resources necessary to progress in such an eclectic field of research as fusion physics. Principal among these efforts are the continuing cooperations with the Japan Atomic Energy Research Institute (JAERI), the Lawrence Livermore National Laboratory (LLNL), the Oak Ridge National Laboratory (ORNL), the Sandia National Laboratory (SNL), and the University of California at Los Angeles (UCLA) and University of California at San Diego (UCSD). A list of these and other collaborations at DIII-D is given in Table 1.1-3. In addition to hosting scientists and engineers to participate in DIII-D experiments, GA personnel participate in the experiments at other national and international laboratories and universities. General Atomics scientists are active in the ITER and TPX programs and also are involved in the development of a physics database for the national Transport Task Force (TTF).

**TABLE 1.1-3
COLLABORATORS PARTICIPATING AT DIII-D**

International Laboratories	National Laboratories	Universities
Cadarache (France)	ANL	Cal Tech
CCFM (Canada)	INEL	Columbia University
Culham (England)	LLNL	Jonh Hopkins University
FOM (Netherlands)	ORNL	MIT
Frascati (Italy)	PPPL	N. Carolina State Univ.
Ioffe Institute (Russia)	SNLA	Univ. of Maryland
IPP (Germany)	SNLL	Univ. of New Mexico
JAERI (Japan)		Univ. of Paris
JET (EC)		Univ. of Texas
KFA (Germany)		Univ. of Washington
Kurchatov Institute (Russia)		Univ. of Wisconsin
Lausanne (Switzerland)		UC Berkeley
NIF (Japan)		UC Irvine
Triniti (Russia)		UCLA
Tsukuba Univ. (Japan)		UCSD

The Fusion Energy Advisory Committee (FEAC) was charged by the DOE to develop recommendations on how best to pursue the goal of a practical magnetic fusion reactor in the context of several budget scenarios covering the period FY94-FY98. FEAC identified DIII-D as one of the high priority U.S. Fusion program elements. In its report of September 1992, FEAC noted that the DIII-D program supports both ITER and the development of an optimized tokamak. DIII-D will be the largest operating U.S. tokamak after the Princeton Tokamak Fusion Test Reactor (TFTR) completes its research program in 1995. The DIII-D divertor and advanced tokamak upgrades together with the strong national collaborative program will continue to provide the U.S. with an internationally competitive magnetic fusion tokamak facility connecting with and supporting the design and initial operation of TPX and ITER.

1.2. HIGHLIGHTS OF THE FY94 DIII-D RESEARCH PROGRAM

Successes on DIII-D during fiscal year 1994 include: the development of the improved Motional Stark Effect diagnostic (MSE) permitting direct measurement of the current density profile; installation of an additional 4 MW of radio frequency power near the ion cyclotron frequency; achievement of high performance plasmas for which a substantial portion of the discharge was in the second stability regime; development of improved understanding of the mechanism for turbulence suppression and reduced transport in the H-mode; installation of the advanced C-coil for correction of error fields, which extends the operational envelope of the tokamak; routine operation of the lithium beam diagnostic for probing the plasma edge; operation of the fast reciprocating probe, providing a new capability for boundary diagnosis; development of improved divertor diagnostics, including infrared cameras, probes and bolometry; installation of improved reflectometry; commissioning of a new ECE system providing electron temperature data with excellent spatial and temporal resolution; and installation of a high performance pellet injector. In addition to controlling many aspects of the plasma shape, the digital control system has been used routinely to maintain constant plasma stored energy, to control the antenna loading of the ICRF systems and to control the density in H-mode discharges with ELMs. A list of DIII-D highlights is given in Table 1.2-1 and selected highlights with more detail are described below.

**TABLE 1.2-1
TECHNICAL DIII-D HIGHLIGHTS OF 1994**

- Measured $j(r)$ profiles routinely with possibility for feedback control
- Installed and operated ORNL pellet injector
- Installed ORNL 32 channel heterodyne radiometer for ECE
- Initiated DiMES divertor material studies
- Achieved 21 keV hot ion VH-mode $n_D(0) \cdot \tau_E \cdot T_i(0) = 5 \times 10^{20}$ keV-sec-m⁻³
- Increased average beta from 11 to 12.5%
- Increased $\beta\tau$ from 1.3 to 1.7% sec
- Completed high triangularity plasma shape experiments
- Increased understanding of VH-mode
- Conducted edge fluctuation studies with Li beam emission spectrometer, phase contrast interferometer, reflectometer, microwave scattering, and edge probe
- Conducted transport studies with ECH and FW
- Carried out fast wave current drive physics experiments
- Observed quiescent high poloidal beta with peaked density profile

1.2.1. INVERTED $q(r)$ PROFILES AND THE MOTIONAL STARK EFFECT DIAGNOSTIC

Control of the current density profile may eventually provide the key to production of a successful fusion power plant which is economically competitive. Access to the so called second stability regime, for example, may permit a tokamak reactor to operate at higher values of beta, the ratio of plasma pressure to the confining magnetic field pressure, than has heretofore been possible. The development of the 16 channel Motional Stark Effect (MSE) diagnostic has greatly enhanced our ability to produce, diagnose and control current density, $j(r)$, profiles with intriguing confinement and stability properties. The safety factor $q(r) = rB_T/RB_p$, where r and R are minor and major radii respectively and B_T and B_p are the toroidal and poloidal magnetic fields, is a key parameter for determining the

basic MHD stability of tokamak plasma. Whereas previously the current density profile was inferred from the magnetics, which required that the $q(r)$ profile be normalized at some point, MSE provides a direct measurement.

Inverted q profiles with $q(0)$ greater than one, hence free of sawteeth and with reversed magnetic shear in the core, have been obtained on DIII-D by injecting neutral beam power early in the discharge during the initial current ramp. The early beam power heats the core, which slows the inward current diffusion during the ramp and broadens $j(r)$. Although the profiles produced in this manner are transient, the reverse shear can last up to three seconds, long enough to study these plasmas in detail. An example of such an unusual shear profile is presented later in Fig. 10.2.-1.

With the new MSE capability, a variety of important experiments on non-inductive current drive, stability, confinement and transport have become possible for the first time. In making the measurement, 16 MSE channels along with all available magnetic probes and flux loops, more than 100 independent signals, are used. The uncertainty in central q , $q(0)$, increases nonlinearly with the value of $q(0)$. Therefore inverted q profiles, which are of interest for advanced tokamak research, inherently have a larger uncertainty in $q(0)$ than the more normal monotonic profiles. For $q(0) = 1.0$ the uncertainty is ± 0.05 and for $q(0) = 7.0$, the measurement is accurate to ± 1.0 . In Fig. 1.2-1 the MSE geometry and a set of magnetic field pitch angle data are shown along with the inferred $j(r)$ profile just before a sawtooth. The MSE diagnostic is an example of the close cooperation among fusion laboratories, principal development having been done by Lawrence Livermore National Laboratory at General Atomics.

Fig. 1.2-1. Results from the MSE measurements, obtained at the points where the observation sightlines intersect the neutral beam, have for the first time permitted study of the effects of $j(r)$ profiles. This example is for a moment just preceding a sawtooth.

1.2.2. THE TRANSITION FROM L- TO H-MODE AND THE PHYSICS OF TURBULENCE SUPPRESSION

Understanding the mechanism for the transition from L- to H-mode confinement remains of importance not only for understanding tokamak dynamics and improving performance, but also as a predictor of performance for new devices such as TPX and ITER. Accordingly, transition studies continue to occupy a prominent position in the DIII-D program. Although the presence of a radially directed electric field near the plasma periphery has been identified with confidence as the reason for turbulence suppression in the high confinement H-mode, the radial force balance equation

$$E_r = (n_i Z_i e)^{-1} \nabla P_i - v_\theta B_\phi - v_\phi B_\theta$$

shows that a radial electric field can be generated by cross field heat and particle transport, via the ∇P_i term, by cross field angular momentum transport via the v_ϕ term and/or by poloidal flows via the v_θ term. In a series of experiments on the details of the transition, heat and particle transport were ruled out as the drivers for the transition since the E_r shear began to change well before a major change in the main ion ∇P_i . Several new diagnostics were used to draw this conclusion.

Pressure profiles, and in particular their changes, can be inferred from measurements of the temperature and density with high time resolution. Such measurements were performed with the lithium beam, determining changes in the edge density, the heterodyne radiometer, measuring changes in T_e , and with the fast probes, measuring changes in the density in the scrape-off layer. These diagnostics demonstrated that changes in the radial shear in the electric field began to occur before any corresponding changes in the pressure profile. In Fig. 1.2-2(a) fluorescence profiles from the lithium beam diagnostic, indicative of the density profile, are seen not to change until after the L-H transition, and this is supported by the radiometer measurement of T_e in Fig. 1.2-2(b). The conclusion of these measurements is that ∇P changes as a consequence of changes in E_r rather than as a cause, and that higher derivatives, such as the curvature of the radial electric field, are probably important, a conclusion which rules out several theories of the L-H transition.

1.2.3. MATERIALS RESEARCH USING COLORIMETRY

As research on boundary physics begins to occupy a larger part of the DIII-D experimental effort in support of TPX and ITER, materials development is assuming greater prominence. Steady state heat fluxes in the divertor region on TPX, for example, are expected to reach 7.5 MW/m². The DIII-D facility provides a resource for testing of materials subjected to these large heat fluxes.

The development of colorimetry during CY94 is an example not only of how DIII-D is being used in support of these new projects but also of the interconnected web through which international contributions to the program are made. The colorimetry diagnostic is a joint project of KFA, Jülich and GA. It uses the existing Divertor Materials Erosion Studies (DiMES) system to expose a sample to plasma bombardment. The diagnostic infers changes in the surface, due, for example, to erosion, by measuring at three wavelengths the intensity of a beam of light reflected from the surface. A photograph of two samples and changes in the thickness of its coating inferred from changes in its reflection spectrum are shown in Fig. 1.2-3.

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Fig. 1.2-2. The lithium beam fluorescence profile shows that changes in the density $n_e(r)$ profile begin to occur only after the L-H transition. This is corroborated by data from the 32 channel ECE system on a different discharge showing formation of a transport barrier following the drop in the D_α signal.

Fig. 1.2-3. Two DİMES samples have been coated with an amorphous hydrogenated carbon film and exposed to tokamak discharges (#37 for 8.6 plasma-sec and #40 for 8.4 plasma-sec). The sides and top surface were first coated with a thin aluminum layer (white color) and then a carbon film, approximately 300 nm thick, was deposited on the top surface (brownish tint). Changes in this surface color, when viewed through three filters, measure changes in film thickness on a shot by shot basis. Erosion, visible as dark areas on the right side of each of the samples, is due to slight misalignments relative to adjacent divertor floor tiles. The erosion process can be seen in analyzed data as a function of position across the sample with shot number as a parameter. These samples are approximately 5 cm in diameter.

1.2.4. THE EFFECT OF TOROIDAL PLASMA FLOW ON GLOBAL MHD MODES

High performance discharges in DIII-D have been produced which exceed the limiting β_N value predicted by the Troyon scaling law. Achievement of an understanding of the factors affecting the stability of these discharges has led to a realization of the importance of wall stabilization of a rotating plasma even when the wall is resistive and the rotation velocity is substantially sub-Alfvénic.

The stabilizing influence of a wall on a confined plasma depends on the conductivity of the wall and the geometric parameters such as the wall thickness and the gap between plasma and wall. If the plasma is not undergoing bulk rotation and the wall is perfectly conducting, the plasma can be stable provided it is less than some critical distance from the wall. But if the plasma is static and the external wall has finite resistivity, the plasma is unstable to an external mode regardless of the size of the gap.

The significant new result is the realization that a wall with finite resistivity can be stabilizing for a nearby plasma in motion. The experimental observation on DIII-D was that higher β_N was achievable when the plasma was driven to rotate, for example, by momentum transfer from off-perpendicular neutral beam injection. As the rotation velocity decreased following termination of beam injection, MHD instabilities grew and in some cases the plasma disrupted.

To address the physics of this situation, the MARS code was modified to treat high β equilibria, revealing the stabilizing effect of even a resistive wall. In Fig. 1.2-4 a stability diagram is presented for several values of the rotation speed compared to the Alfvén speed. There is a window of stability matching the DIII-D geometry for sufficiently rapid rotation. The implications of this are that β_N values even higher than the present record values are probably achievable for sufficiently rapid rotation and that passive wall stabilization can be used to good advantage in the operation of the next generation tokamaks as well.

Fig. 1.2-4. Stability diagram for a DIII-D equilibrium using measured rotation and density profiles. Computed are growth rates for different levels of rotation frequency normalized to the frequency of rotation at the Alfvén velocity.

2. DIVERTOR AND BOUNDARY RESEARCH PROGRAM

2.1. BOUNDARY PHYSICS AND TECHNOLOGY OVERVIEW

During FY94 the advanced divertor cryopump (ADP), was instrumental in the addressing of a variety of important tokamak-related issues. The cryopump was used both as a tool in reducing neutral particle recycling (*e.g.*, by partially depleting the wall of deuterium loaded during the discharge formation phase), as well as in specific particle exhaust and wall conditioning experiments, such as in the radiative divertor and helium pumping experiments described below.

The capability for observing and analyzing impurity behavior in both the main plasma and the divertor plasma was significantly enhanced during FY94. Upgrades were made to the SPRED VUV spectrometer. The multichord divertor spectrometer was routinely used in monitoring divertor impurities, and the development of interpretation capabilities using a Monte Carlo impurity transport model progressed. A Russian high resolution spectrometer yielded important data on the temperature of the divertor deuterium atoms. Argon frosting of the cryopump was employed to pump out helium. Several divertor materials issues were addressed with the recent improvements to the existing DiMES diagnostic.

The power flow through the scrape-off layer and into the divertor presents a serious design challenge for next generation tokamaks, such as ITER. To address this problem, the experimental focus continued to be on (1) the scaling of the scrape-off layer (SOL) and divertor plasma transport to gain insight into how the peak heat flux should scale to larger tokamaks and (2) the ways to reduce peak heat flux at the divertor target. In the latter case, experiments concentrated on two promising approaches. Each has been shown to reduce peak divertor heat flux by enhancing the plasma radiated power either by gas puffing deuterium or by gas puffing impurities during the plasma discharge.

An important new addition to these studies in FY94 was combining the radiative divertor concept and active particle pumping of the divertor. Preliminary results show that D₂ injection during ELMing H-mode is compatible with active particle pumping and long pulse plasma operation. Several experiments to inhibit the accumulation of the injected impurities in the main plasma by improving the divertor impurity retention with enhanced SOL plasma flow were also performed. Experiments were continued on the scaling of the SOL and divertor plasma transport. We have found that broad SOL plasmas or density plateaus only appeared following the onset of ELMs, a result that clearly impacts reactor divertor thinking (*e.g.*, "slot" width).

Finally, several diagnostic improvements this past year have significantly improved our ability to diagnose divertor plasmas. These include a new IR camera that views the face of the bias ring (allowing peak heat flux measurement at the optimum pumping location), an improved algorithm for the IR camera systems (making processed data available on an intershot basis), and an improved analysis routine for the bolometer array diagnostic (providing improved 2-D radiation profiles).

2.2. DIVERTOR PHYSICS

A key issue in designing next generation tokamaks is how to reduce the anticipated high peak heat flux at the divertor plates. The DIII-D divertor physics group has concentrated on two promising ways of reducing excessive divertor power loading. Both methods, which fall under the aegis of “radiative divertor” studies, rely on radiating away most of the incoming power before it strikes the divertor target. In the first approach, deuterium gas is injected into the discharge, leading, under certain conditions, to a significant enhancement in divertor radiated power from both hydrogenic and intrinsic impurity (mostly carbon) sources. In the second approach, large reductions in divertor power loading follow the introduction of a recycling impurity gas (*e.g.*, neon), typically enhancing the radiated power predominantly from a band just inside the separatrix of the main plasma. Previously, these studies were done *without* active pumping of particles. During FY94 the compatibility and steady state performance of the radiative divertor *with* cryopumping were explored.

2.2.1. D₂-BASED RADIATIVE DIVERTOR

Deuterium gas injected into ELMing H-mode diverted discharges has typically reduced the total power at the divertor target $\sim 2\times$ and the peak heat flux $\sim 3\text{--}5\times$ with only modest ($<10\%$) degradation in plasma energy confinement. Most of this reduction in heat flux occurred at the sudden formation of a high density, highly radiating region (MARFE) located between the outboard divertor separatrix target and the X-point, and has been associated with sharply reduced particle flux at the outboard separatrix target.

The range in operating parameters in DIII-D has been extended this year (*e.g.*, $q_{95} \sim 2.9\text{--}6.0$ and peak heat flux $\leq 7 \text{ MW/m}^2$), further indicating that this MARFE solution to heat flux reduction is a general and controllable feature and may have relevance in an ITER-like environment. Recent experiments have also shown that steady state operation in the MARFE regime is compatible with divertor pumping, and that the pumping rate is much less sensitive to relative divertor target/pumping plenum location than in a comparable non-MARFE case. With pumping, the plasma could be held at constant density while simultaneously maintaining the MARFE (and a factor of three reduction in peak heat flux), as shown in Fig. 2.2-1. Previously, deuterium puffing produced a constant rise in plasma density. The successful inclusion of pumping demonstrates a crucial element in practical radiative divertor operation. Our next step is to use the cryopump to obtain a divertor MARFE at lower density.

2.2.2. IMPURITY INJECTION

Neon injection without active cryopumping can reduce the peak power flow into the SOL, because, as the neon penetrates into the main plasma, the radiative loss inside the separatrix (“radiating mantle”) increases. During radiative divertor experiments with neon puffing, core neon concentrations of 1–3% have been calculated. Approximately 20–40% of the total number of impurity atoms injected make their way into the core plasma. In order to improve divertor impurity retention, several recent experiments used the divertor cryopump and midplane/divertor deuterium gas puffing to try to reduce the central neon density obtained for a given reduction in divertor heat flux.

It was thought that the added deuterium puff combined with the divertor pumping would increase the plasma flow in the SOL and thereby keep impurities localized in the divertor and out of the main plasma. As yet, this combination of puffing and pumping has not produced a situation where the major fraction of the neon radiation

Fig. 2.2-1. Pumping and feedback control lowers the density in D_2 radiative divertor experiments.

comes from the divertor region. This may mean that we have not yet succeeded in producing a large enough flow in the SOL plasma or that neutral neon leakage from the relatively open divertor dominates the neon transport. Figure 2.2-2 compares unpumped and cryopumped plasmas with similar discharge conditions and shows that reduced heat flux is possible at lower density with pumping. We are now considering how to improve diagnosis of these discharges to help understand these results.

Fig. 2.2-2. First attempts at neon puff with pumping show heat flux reduction at lower density.

2.2.3. SCRAPE-OFF LAYER PLASMA STUDIES

Outside the separatrix a thin layer of plasma, the SOL, carries power and particles into the divertor. This layer can become thicker or thinner, depending on operating regime, and this affects the heat flux distribution downstream at the divertor target. The DIII-D divertor physics group continued its focus on experiments aimed at

characterizing the SOL plasma in different transport regimes. The midplane scanning Langmuir probe was used along with the Thomson scattering system to look for broader scrape-off layer plasmas. Both the Thomson system and the midplane Langmuir probe (which can measure lower plasma density farther outside the separatrix than the Thomson system) were important tools in these studies. We found that the SOL was narrowest in VH-mode, somewhat thicker in ELM-free H-mode, and largest in ELMing H-mode and L-mode discharges. No “plateaus” were found in the ELM-free phase of these discharges. This has confirmed an independent study using the confinement and edge physics databases which found that broad SOL plasmas or density plateaus only appeared following the onset of ELMs.

2.2.4. DIAGNOSTIC AND ANALYSIS UPGRADES

The execution and analysis of the radiative divertor experiments have been aided by several diagnostic improvements. A new IR TV camera viewing the face of the bias ring and the throat of the divertor pump was installed. Previously, the divertor floor immediately adjacent to the bias ring and the inward facing side of the bias ring could not be monitored; hence, observations of the peak heat flux at the optimum pumping location had been impossible. In addition, data from the IR TV system can now be converted to heat flux data and plotted alongside other data between shots. New algorithms for the bolometer data have been applied which more realistically constrain the 2-D reconstruction of the radiation profiles to be flux functions. Increased manpower on the Tangential X-point TV has allowed us to look at the divertor MARFE in a number of different wavelengths and we have compared the spatial profiles of CIII emission to the profiles of radiated emission deduced from tomographic analysis of bolometer data. Figure 2.2-3 compares the distribution of CIII brightness (tangential TV) before deuterium injection with the CIII emission during the MARFE phase of deuterium injection.

2.3. BOUNDARY PHYSICS MODELING

Significant developments in edge physics modeling were made during FY94. Upgrades in our principal edge modeling code (UEDGE) included (1) improvements in generating non-orthogonal grids, (2) more efficient numerical differencing on these grids, and (3) the capability of modeling aspects of impurity transport. In addition to these UEDGE improvements, a separate code (RECLUSE) was developed to couple UEDGE calculated SOL plasma parameters with the neutral particle transport calculations of the DEGAS code.

With these upgrades, we have made important progress in our efforts to model DIII-D discharges. Most of our effort focused on H-mode discharges. Our modeling indicates that the anomalous perpendicular transport coefficients lie in the range $0.03 \leq D_{\perp} \leq 0.25 \text{ m}^2/\text{sec}$, and $0.12 \leq \chi_e \leq 0.25 \text{ m}^2/\text{sec}$. The electron thermal diffusivity increases with SOL heating power for fixed plasma current. The perpendicular transport coefficients are approximately an order of magnitude larger for a low power L-mode discharge than expected for an equivalent power H-mode discharge. Simulation of DIII-D plasmas with these perpendicular transport coefficients has reproduced the upstream density and temperature profiles measured with the Thomson and CER diagnostics, as well as the heat flux profile at the outer strike point. We have successfully obtained UEDGE solutions for the SOL plasma in which the plasma is detached from the divertor plates. This is an essential step in understanding the radiative divertor. With low SOL heating powers and/or high separatrix densities, we can obtain solutions with the total pressure at the divertor a factor of five lower than at the mid-plane, in reasonable agreement with experiment.

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Fig. 2.2-3. Visible CIII emission in the divertor from tangential X-point T.V. views of formation of a radiating MARFE with deuterium injection. The upper figure is before injection. Low levels of emission from outside the vacuum vessel are artifacts of the inversion with the intense CIII emission.

2.4. IMPURITY TRANSPORT AND CONTROL IN DIII-D

The capability to observe and analyze impurity behavior in both the main plasma and the divertor plasma was significantly enhanced during FY94. Upgrades in the SPRED VUV spectrometer analysis routines improved both the spectral information that can be gleaned on a between-shot basis and the accuracy of detailed impurity transport results obtained in conjunction with the MIST code. The multichord divertor spectrometer (MDS) was routinely used in monitoring divertor impurities, and the development of interpretation capabilities using a Monte Carlo impurity transport model continued. Measurement of the Doppler profiles of the D_{α} line by the Russian high resolution spectrometer yielded important data on the temperature of divertor deuterium atoms during D_2 gas injection experiments. The first helium exhaust experiments were successfully performed on DIII-D using argon frosting of the advanced divertor cryopump to pump helium. Important improvements that were made in the operational reliability of the DiMES hardware during this past year have allowed us to examine several important issues related to erosion/redeposition, sheath physics, and material durability. A major step forward in analyzing erosion/redeposition occurred with the commissioning of the "colorimetry" diagnostic, allowing *in-situ* measurements on the DiMES sample. An improved method of boronization has extended the operating range of VH-mode in DIII-D.

2.4.1. SPECTROSCOPY AND MODELING

Spectroscopy is crucial both to the operation of DIII-D and to interpretation of experimental results. A key diagnostic here is the SPRED VUV spectrometer, which routinely provides absolutely calibrated, time-resolved spectra of extreme ultraviolet impurity emissions in the main plasma. Several upgrades in the analysis routines were made this past year, including improved beam attenuation calculations and charge-exchange recombination cross sections, both of which are essential for accurately determining the core impurity content. In addition, inter-shot spectral analysis routines were developed to provide core impurity concentrations between shots. Detailed analysis of the impurity transport is obtained from SPRED's atomic ion spectra, as interpreted by the MIST charge-state equilibrium code.

Absolute calibration of the MDS diagnostic provides visible spectroscopy data at seven distinct poloidal locations across the divertor floor. The MDS has successfully measured the line intensity of carbon, neon and argon in the divertor during radiative divertor experiments. In collaboration with ORNL personnel, the MDS data analysis programs have been upgraded. MDS data can now be analyzed on UNIX workstations and the data viewing program is greatly expanded, functioning similarly to the SPRED spectrometer analysis program. While essential in understanding the radiative nature of the divertor plasma, these spectra require interpretation because of the complexity of the divertor dynamics. To assist in this interpretation, the Monte Carlo impurity code is being developed.

Important new information has been provided by the Russian high resolution spectrometer measurements of D_{α} Doppler profiles during a limited number of D_2 -based radiative divertor shots. These profiles indicate a very low temperature (1–2 eV) for neutrals in the divertor region, that remains nearly constant during ELM events. Knowing the temperature of the divertor neutrals is an important ingredient in modeling the D_2 -based radiative divertor.

2.4.2. HELIUM TRANSPORT/EXHAUST

Helium transport and exhaust has been an active area during this past year. The first He exhaust experiments were performed on DIII-D using Ar frosting of the ADP to pump helium. The exhaust rate of He from these ELMing H-mode plasmas appears to be within the acceptable range for a fusion reactor like ITER, based on a measured value of $\tau_{\text{He}}^*/\tau_{\text{E}} \approx 8$, one of the measures of "goodness" for a fusion reactor. Current reactor modeling codes suggest that this ratio should be $\leq 7-15$. This result is a strong indication that ELMing H-mode discharges are probably acceptable for a reactor, at least from the standpoint of helium removal.

The helium transport work has continued in order to evaluate transport coefficients D_{He} (helium diffusivity), v_{He} (pinch velocity), and χ_{eff} (local thermal conductivity) for a variety of confinement regimes on DIII-D (L-mode, ELM-free H-mode, ELMing H-mode, and VH-mode). Current emphasis in the He transport program will be to produce a database of He transport coefficients over a rather large operational range for DIII-D. He beam injection experiments are currently planned for early in FY95 at DIII-D to investigate the comparison of He beam fueling and He gas puffing with active He exhaust via the Ar-frosted ADP cryo-pump.

2.4.3. DIVERTOR MATERIAL EXPOSURE SYSTEM

Several divertor topics were addressed with the DiMES. Repairs and improvements to the DiMES insertion mechanism during the past year have led to significantly higher operational reliability. Experimental results from a tungsten-coated graphite sample confirmed the modeling conclusions that the net erosion of graphite is about an order of magnitude lower than gross erosion and that the erosion rate of tungsten is much (>7 times) lower than graphite. Other areas of DiMES-related study included: (1) sheath physics (in collaboration with SNL); (2) characterization of the particle flux and energy striking the divertor during a disruption (in collaboration with SNL and ANL); (3) estimation of the deuterium energy at the DIII-D divertor during ELMing and ELM-free shots; and (4) analysis of the durability of pyrolytic boron carbide and silicon carbide in normal divertor conditions (in collaboration with the Russian TRINITI laboratory).

A new diagnostic, colorimetry, was installed and operated on DIII-D during FY94 in collaboration with KFA, Jülich, Germany. Colorimetry measures changes in the thickness of films *in-situ*, and hence erosion and deposition of plasma facing surfaces. This is done by measuring the intensity of light reflected off the film surface at three wavelengths. With the commissioning of colorimetry, measurements of thickness in specially prepared, amorphous films of carbon and of boron can now be made on a shot-by-shot basis. More details appear in the Introduction.

2.4.4. WALL CONDITIONING

Boronization has been shown to reduce both carbon and oxygen central impurity concentrations. In FY94 the boronization process was improved by optimizing the glow discharge current during boronization and using a mixture of 20% diborane and 80% helium. Although it is often difficult to correlate long term changes in wall conditioning with specific improvements in plasma performance, we have noted that after this improvement in the boronization process, several expansions of the DIII-D parameter space were observed that could at least partially be attributed to improved wall conditioning. These enhancements included: a lowering of the VH-mode power threshold; production of double null ohmic H-modes with ELM free periods greater than 1 sec; VH-modes at currents of 0.8–1.0 MA; and a single null 1 MA VH-mode at $q_{95} = 5.1$.

2.5. DIVERTOR CRYOPUMPING AND DENSITY CONTROL

The divertor cryopump, operated reliably this year, allowing it to be used routinely in many experiments. The pump frequently was used simply as a tool to reduce neutral particle recycling by partially depleting the wall of deuterium loaded during the discharge formation phase. The pump was also used in specific particle exhaust and wall conditioning experiments.

Divertor bias experiments in 1991-92 showed the capability of bias to increase neutral gas pressure in the (then unpumped) DIII-D pump plenum and to reduce its sensitivity to geometry, *i.e.*, to strike point location. In order to determine whether divertor bias increases particle exhaust sufficiently to produce usefully lower density high performance plasmas for future Advanced Tokamak experiments, combined bias and pump experiments were performed. The experiments demonstrated a doubling of the pump exhaust rate by bias at a given core plasma density and also showed the expected reduced geometric sensitivity in ELMing plasmas. However, as shown in Fig. 2.5-1, the bias did not overcome the roughly n_e^3 exhaust rate scaling. A similar experiment in ELM-free H-mode, where the divertor plasma is still more tenuous, showed even less exhaust. Thus, the ability to pump low density plasmas is still limited.

An important recent finding indicates that divertor cryopumping alone can maintain low recycling wall conditions. Normally in DIII-D, wall conditioning by helium glow discharge is used before every shot to ensure low recycling and repeatable experimental conditions. However, walls are expected to reach a higher recycling equilibrium in long-pulse tokamaks such as ITER and TPX. In the DIII-D experiment, pumped ELMing discharges were repeated without He glow conditioning. This experimental scenario is a severe test of pumped conditioning, since each plasma startup necessarily involves the injection of a large amount of gas.

Fig. 2.5-1. Biasing the divertor increases the pump exhaust rate.

3. ADVANCED TOKAMAK RESEARCH

The goal of the Advanced Tokamak (AT) program in DIII-D is to optimize both energy confinement time and beta in a noninductively driven discharge relevant to steady state operation. It is convenient to define normalized values for confinement and β and then to discuss the normalization constants which permit comparisons with certain standard conditions to be made. Therefore we use H to compare energy confinement with the baseline set by ITER-89P L-mode scaling and $\beta_N = \beta / (I/aB)$ to compare β with the Troyon limit, $\beta_N \sim 3$. Our target is simultaneously to achieve $H=4$ and $\beta_N=6$, with significant bootstrap fraction, in quasi-steady-state. These values have already been achieved independently in DIII-D for finite durations. Important progress has been made both theoretically and experimentally in elucidating the underlying physics of the performance improvements and perfecting the reliability of the high performance regimes in the past year.

The approach adopted by the AT program to improve performance is to control and optimize the internal profiles of a tokamak discharge. These include the plasma shape factors, current density profile, pressure profile, plasma flow shear, and safety factor. In one of our major experimental campaigns last year, we have demonstrated that plasma triangularity is important for performance while plasma elongation plays a weaker role. Many of our high performance discharges have achieved beta values which far exceed magneto-hydrodynamic (MHD) theory estimates according to Troyon scaling with the plasma wall at infinity. New theoretical understanding has emerged which indicates that plasma rotation is a key to the effectiveness of a resistive wall in stabilizing MHD modes. This theory has been successfully validated by DIII-D results.

The AT program will be moving from a physics validation phase to an active control phase in the near future in order to extend the duration of the high performance discharges and to demonstrate steady-state relevance. Two important tools have been developed for this purpose, a reliable measurement of current and safety factor profile using the Motional Stark Effect (MSE) and the determination of non-inductive current profile components. The third phase of the DIII-D AT program is the integrated demonstration of a high performance discharge. We have identified the second stable core regime to be one of the most promising targets. MHD-transport simulations have quantified the power requirements needed. The attractiveness of this second stable operating regime for fusion power plants was demonstrated.

3.1. THE ROLE OF SHAPING IN ACHIEVING HIGH PERFORMANCE IN DIII-D

As a result of continuing improvements in the DIII-D device, we have achieved new DIII-D record values of the triple products $n_D(0)T_i(0)\tau_E^{\text{th}} = 4 \times 10^{20}$ (keV·s/m³), $n_D(0)T_i(0)\tau_E = 5.6 \times 10^{20}$ (keV·s/m³). While it is expected that the beta limit will depend on shape, we also find that confinement, at finite beta, depends on the plasma shape and that the plasma triangularity is most important for performance while plasma elongation plays a weaker role. The pursuit of higher beta therefore became focused on a study of the shape of the plasma. The particular choices of shape for this study were determined partly from physics motivation, and partly from preliminary design work done on the radiative divertor.

In previous work on DIII-D, it has been shown that β depends on plasma shape in a manner consistent with Troyon scaling. We will use the parameter $S \equiv (I_p/aB_t) \cdot q\psi$, a measure of the toroidally-weighted poloidal contour length, to characterize the plasma geometry. We have observed, in a comparison of single- and double-null diverted plasmas, that the performance of double-null plasmas is superior to that of the single-null. Summarizing those results briefly, we found that: (a) $\beta \cdot \tau$ increases with increasing S ; (b) $\beta \cdot \tau$ decreases with increasing plasma current for $q_\psi < 4$; (c) $\beta \cdot \tau$ increases approximately as $\sqrt{B_t}$; and (d) for double-null plasmas good confinement is maintained with $\beta_N > 3$, whereas for single-null plasmas the confinement begins to degrade at $\beta_N = 2$.

The focus of this experimental effort was to clarify the features responsible for this improved performance. The data indicate that triangularity, defined as $\delta = 1 - [R_{\kappa\max} / (1/2) (R_{\text{out}} - R_{\text{in}})]$ where $R_{\kappa\max}$ is the major radius at the point of maximum elongation ($\kappa = b/a$) of the last closed flux surface and R_{out} and R_{in} are the outer and inner major radii of that flux surface, is the important variable. In the experiment we investigated four double-null shapes: (a) $\kappa \approx 2.1$, $\delta \approx 0.9$; (b) $\kappa \approx 2.1$, $\delta \approx 0.3$; (c) $\kappa \approx 1.8$, $\delta \approx 0.3$; and (d) $\kappa \approx 1.8$, $\delta \approx 0.8$. For each shape, plasma current and neutral beam power were systematically varied. Results discussed here were obtained during ELM-free periods of durations 0.22–1.3 s following the L–H transition.

If we use DIII-D/JET confinement scaling, $\tau_E \sim F \cdot I_p R^{3/2} P^{-1/2}$, where F is an enhancement factor, then, below the β limit, we can write

$$\beta \cdot \tau_E = \text{const} \cdot \left[\frac{S^2 R^2}{1 + \kappa^2} \right] \left[\frac{F}{q} \right]^2.$$

In Fig. 3.1-1 we present the shapes used in this experiment and for each discharge the maximum $\beta \cdot \tau$ versus the geometric quantity $[S^2 R^2 / (1 + \kappa^2)]$. This is a reasonably good descriptor of performance.

The separation of I_p and the safety factor, q , afforded by the geometric variation makes it clear that the degradation in F which occurs with increased plasma current is an effect of lowered q , not explicitly I_p . It is seen that high triangularity and not simply high elongation yields the highest values of $\beta\tau$.

While shaping might be thought primarily to affect the outer regions of the plasma, the most dramatic change with plasma shape lies in central heating efficiency. To illustrate this, in Fig. 3.1-2 we compare the electron temperature profiles for shapes C ($\delta=0.30$) and A ($\delta=0.85$), under conditions of similar heating power and safety factor. For the higher triangularity plasma, the temperature gradient is large from approximately the $q=2$ surface to the magnetic axis, whereas at lower triangularity the rise in T_e hence the gradient, is more modest. Similar effects are seen in the T_i profiles. Density profiles are similar in both cases and are very broad, typical of DIII-D ELM-free plasmas.

The $q(r)$ profile also evolves differently depending on triangularity. For high triangularity plasmas, when beam-heating commences, q_0 rises above unity, typically to about 1.5. For the lower triangularity plasmas little change is seen with heating ($q_0 \lesssim 1$).

In the global results shown in Fig. 3.1-1, it appears, to the extent that shaping merely affects the ratio of the plasma current to the safety factor, that we are discussing a continuum in the quality of the discharge. However, when we examine the plasma core, the results appear more as a bifurcation between those plasmas which

Fig. 3.1-1. The dependence of $\beta \cdot \tau_E$ on $[S^2 R^2 / (1 + \kappa^2)] (m^2)$. The τ_E value is the global value, $W/(P - \dot{W})$, with data restricted to $\dot{W}/P < (1/3)$. Each datum is the maximum from a single discharge. The safety factor range is $3.2 \leq q \leq 5.6$.

attain Mercier stability and those which do not. For these high triangularity plasmas the core is in the second stability regime. The high performance phase is transient, and is, in fact, a VH-mode. The low triangularity plasmas remain H-mode.

3.2. PLASMA ROTATION AND RESISTIVE WALL MODES

Experiments in DIII-D have achieved β values which exceed theoretical MHD stability estimates, as determined by sophisticated numerical codes, by as much as 30%. This discrepancy has been resolved during the last year by progress on both the experimental and theoretical fronts. This new understanding indicates that plasma rotation is a key ingredient in obtaining higher β and suggests that increases in β by as much as a factor of two may be attainable.

The source of the discrepancy between earlier theory and experiment was the wall stabilization of the external kink mode. The plasma in the experiment appears to behave as if the vacuum vessel of the tokamak is an ideal conductor allowing higher stable β values. Until recently, theory predicted that a wall with finite resistivity could not stabilize a kink mode but could only reduce its growth rate to the flux diffusion time of the resistive wall, only a few milliseconds in DIII-D.

Fig. 3.1-2. Profile evolution for high (thick line) and low triangularity plasmas at the same plasma current, elongation, and input power (2 MA, 2, 10 MW). Times are relative to the start of NBI.

The missing factor in the above discussion is the toroidal plasma rotation arising from co-injected neutral beam heating. In the experiment, when the plasma rotation slows sufficiently, the plasma β always drops below the ideal limit; often the plasma disrupts. Recently, Bondeson and Ward [Phys. Rev. Lett. 72, 2709 (1994)] computationally demonstrated, using the MARS code, wall stabilization by a resistive wall for a plasma with flow. Plasma dissipation, such as Landau damping or plasma viscosity, was also found to play a key role in the stabilization.

We have extended the MARS code to treat more realistic rotation profiles and a number of possible plasma damping mechanisms. Extensive numerical experiments have been run. The qualitative behavior of the results can be understood using a cubic dispersion relation which describes three modes of the plasma and which clearly indicates the roles of plasma rotation, the resistive wall, and plasma dissipation. When the plasma is stationary, two modes are damped and one mode is unstable regardless of wall position; only the growth rate varies. With sufficient plasma rotation and plasma dissipation—both are necessary—mode coupling between this unstable mode and the two damped modes in the plasma opens a region of stability for the plasma wall location. The “resistive wall mode” is still unstable with the wall close to the plasma and the “plasma mode” is unstable with the wall beyond the stabilization point of an ideal wall, but between these two regions is a stable operating window. Quantitative results from the full MARS code allow comparisons with experiment, *e.g.*, simulating a particular DIII-D discharge we find that the computed critical rotation velocity for plasma stability agrees well with the experimental value.

This new understanding and verification of plasma rotation stabilization offers the possibility of significant increases in plasma β for advanced tokamak operations in DIII-D and TPX.

3.3. MEASUREMENT OF $j(r)$ AND $q(r)$ PROFILES ON DIII-D

The primary diagnostic for measurement of the internal poloidal field on DIII-D is the Motional Stark Effect (MSE) instrument. This instrument relies upon the splitting of the neutral beam Balmer- α line into orthogonally polarized components (σ, π) due to the motional $\mathbf{E}=\mathbf{v} \times \mathbf{B}$ electric field. By measuring the polarization of the σ line, the local pitch angle of the magnetic field B_θ/B_T can be determined.

The MSE system was initially installed on DIII-D with only one channel, but was upgraded to 8 channels in 1992. Although the 8 channel system performed well in the plasma core ($155 \text{ cm} < R < 200 \text{ cm}$), the radial resolution was poor ($\sim 12 \text{ cm}$) at $R > 200 \text{ cm}$ due to limitations of the viewing geometry. In 1994 we have addressed this problem with a significant upgrade to the MSE system. A new 8 channel system (edge MSE) has been added with a viewing geometry that covers the outer portion of the plasma. The radial resolution near the outer separatrix is now $\sim 2 \text{ cm}$. The original MSE system has been realigned to view the plasma core. The two instruments combine to give 16 channels that cover the range $160 \text{ cm} < R < 228 \text{ cm}$ with a chord spacing 4–5 cm. We also improved the MSE optical system by rebuilding the detection optics in a dedicated diagnostic room. This room provides temperature control, cleanliness, and ease of access, which has helped improve the instrument reliability significantly.

In anticipation of feedback control of the q profile, the MSE signals have been coupled via analog fiber optic links to the DIII-D real-time control computer. The MSE signals are now routinely acquired in real-time and the development of algorithms for processing the data in real-time is ongoing.

During operations in 1994, MSE operated reliably on nearly every discharge with neutral beam injection. Several important observations were made with the new MSE system. With the improved resolution at the edge, we now observe the development of a large current density with complex structure at the edge during VH-mode. These edge currents are believed to play an important role in the termination of VH-mode. Also, for the first time, we have produced and measured q profiles with large reversed shear in the core. Through early beam injection during the initial current ramp, profiles with $q_{\min}=1.5$ and q_0 as high as 10 have been obtained.

3.4. DETERMINATION OF NON-INDUCTIVE CURRENT PROFILE COMPONENTS

For advanced tokamak discharges which rely upon current profile modification and control, knowledge about the current profile alone is not sufficient to characterize the noninductive currents driven by auxiliary heating systems using fast waves and neutral beams. For this purpose, a new technique has been developed to separate the measured current profiles into inductive and non-inductive parts. Essentially, if the current profile is known as a function of time, direct application of Faraday's law allows the internal electric field to be determined. This technique has been implemented by using time sequences of EFIT reconstructions, and taking time derivatives of the poloidal flux.

Two important results were achieved this year using this technique. First, the local resistivity profile was measured directly and compared to the model of Spitzer and to neoclassical resistivity. Within the experimental uncertainties, neoclassical resistivity was shown to be appropriate while Spitzer resistivity was clearly too low to describe the results. Second, the non-inductive current profile due to bootstrap currents and neutral beam current drive was measured. This result confirmed the existence of these currents and also showed that the models of these

currents agree within the experimental accuracy. With this new tool, we are now ready to characterize the non-inductive currents driven by the new rf systems (fast wave and electron cyclotron heating).

3.5. ADVANCED TOKAMAK SIMULATIONS

The ratio of fusion power produced to auxiliary heating power required to sustain a tokamak fusion reactor is proportional to the product $\beta\tau_E$, where β is the ratio of plasma pressure to confining magnetic field pressure and τ_E is the energy confinement time. Advanced modes of tokamak operation seek to maximize this product by simultaneously achieving large values for both β and τ_E . Advanced tokamak simulations which are actively under investigation at DIII-D to achieve this goal include: transport modeling of H (high) and VH (very high) confinement discharges; characterization of discharges which have high internal inductance due to peaked current profiles; and modeling of high bootstrap current discharges.

One possibility for high confinement, high beta operation, which includes high bootstrap current fractions, is the "second stable core" scenario. In this scheme the value of normalized beta, β_N , can be significantly increased above its limiting value of about 3.5 obtained during standard modes of tokamak operation. The improvement is due to the bypassing of the fundamental ballooning mode beta limit. By tailoring the plasma current density so that the peak current occurs away from the magnetic axis, regions of reversed magnetic shear are created inside the plasma. Such reversed shear regions are desirable since the region of the plasma where the shear is negative are stabilized with respect to ballooning modes. Experimentally, improved confinement has also been observed in the reversed shear regime. By appropriate fine tuning of the current density profile and by keeping a conducting shell at about 1.2 times the plasma minor radius, it is possible to eliminate the most virulent low n kink modes as well, which allows the plasma to achieve states with high internal energy for extended periods of time.

Second stable core simulations depend on our ability to shape the current profile using a combination of neutral beam, fast wave, and electron cyclotron heating and current drive. In addition, for steady state reactor operation, it is desirable to have a significant fraction of the total plasma current supplied by the bootstrap current which should be well aligned with the desired total current profile. The details of the simulations depend on the energy confinement model assumed. However our simulations indicate that second stable core configurations, which are in an MHD as well as a transport equilibrium state, can be achieved with a variety of plausible confinement models, including the Rebut-Lallia-Watkins model (which is of special interest here due to its prediction of greatly reduced transport in the negative magnetic shear region), the Hsieh empirical model of confinement which fits DIII-D L-mode discharges, and a simpler confinement model required for Intor scaling.

In Table 3.5-1 we summarize the results of the simulations performed using the three confinement models. The improved confinement is given in terms of H, defined as the energy confinement time relative to the ITER89P confinement scaling law. It is seen that high values of normalized beta and confinement, indicated by H parameters greater than one, are achieved simultaneously and the bootstrap current represents a large fraction of the total current. The results are obtained using only modest amounts of current drive, fast wave, and electron cyclotron power to drive current and maintain temperature profiles consistent with the given amounts of bootstrap current. Reliance on externally driven ohmic current is kept to a minimum.

TABLE 3.5-1
RESULTS FOR THE DIII-D SIMULATIONS

	INTOR	RLW	HSIEH
β_N	5.7	5.2	5.4
β_P	2.52	2.50	2.45
H	3.6	3.5	3.8
I_p (kA)	1600	1600	1600
I_{boot} (kA)	1070	975	1170
I_{RF} (kA)	309	349	286
I_{beam} (kA)	172	266	181
I_{ohm} (kA)	49	9	-39
P_{FW} (MW)	6.5	6.5	6.5
P_{ECH} (MW)	7.0	8.0	8.0
P_{NBI} (MW)	6.5	6.5	4.5
ℓ_i	0.57	0.84	0.64
γ^*	0.04	0.04	0.03
W (MJ)	3.8	3.6	3.6
q_{min}	2.57	1.86	2.54

4. TOKAMAK PHYSICS

The Tokamak Physics Program in DIII-D contributes to the fundamental understanding of key physics issues in toroidal confinement systems. The H-mode phenomenon is no longer limited to plasma configurations with a divertor or separatrix. Limiter H-modes and H-modes in stellarators have also been achieved experimentally. DIII-D continues to be a leading contributor to this subject.

Several important results obtained in the past year are highlighted here. With improved resolution in our diagnostics, the causal effect of the radial electric field on the L-H transition has been clarified recently. DIII-D is also a leader in the study of nondimensional transport scaling, and the understanding of the nature of electron and ion transport in L- and H-mode has been one of the major results of transport studies this past year. Experiments in a wide variety of operating modes have called attention to the interaction of MHD modes with fast ions. This appears to have significant consequences for plasma confinement, as well as energy and momentum balance. This will be a subject of further study for the coming year.

DIII-D has made significant contributions to the ITER design database. An example is the ITER elongation studies carried out in DIII-D which produced the necessary information for a design decision on the plasma shape. We will continue to provide similar input to ITER and TPX in the future.

4.1. H-MODE PHYSICS

The hypothesis of stabilization of turbulence by shear in the $\mathbf{E} \times \mathbf{B}$ drift speed successfully explains the observed turbulence reduction and confinement improvement seen at the L to H transition. This paradigm of shear suppression of turbulence as the mechanism for the L to H transition has been verified by quantitative comparison of the predictions of the hypothesis with experimental results from the DIII-D tokamak.

A key question for L to H transition studies is how the radial electric field, E_r , is formed at the plasma edge. The radial force balance equation

$$E_r = (Z_i n_i e)^{-1} \nabla P_i - v_{\theta i} B_\phi + v_{\phi i} B_\theta \quad (1)$$

relates E_r to the main ion pressure gradient ∇P_i , poloidal rotation $v_{\theta i}$, and toroidal rotation $v_{\phi i}$. In the plasma edge, direct measurements show that both ∇P_i and $v_{\theta i}$ are the important terms around the time of the L to H transition, with ∇P_i being the dominant negative term throughout most of the H-mode. Detailed measurements of the time history of the E_r shear show, however, that the shear changes significantly prior to the major change in the fluctuations and in ∇P_i . This indicates that the L to H transition trigger (the dominant contribution to the change in E_r at the transition) is the $\mathbf{V} \times \mathbf{B}$ rotation, not ∇P_i . These observations are inconsistent with theories which depend only on ∇P_i to create the electric field well at the plasma edge. However, theories which include only the $\mathbf{V} \times \mathbf{B}$ term and neglect the ∇P_i term are not able to produce the correct sign of E_r later in the H-mode. Accordingly, a complete theory must include the effects of both terms.

The E_r shear increases significantly before the fluctuation amplitude changes, consistent with increasing E_r shear as the cause of the turbulence suppression. This observation also indicates that there is a threshold in E_r shear required to influence the turbulence. The spatial dependence of the turbulence reduction, determined by the fast stroke Langmuir probe, is consistent with shear suppression for negative E_r shear. For positive E_r shear, the turbulence suppression is consistent with the effect of E_r curvature for modes for which an E_r well is destabilizing. Finally, the transport barrier depends on the phase angle between the density and potential fluctuations inside the E_r well, an effect not included in existing L to H transition models.

4.2. NONDIMENSIONAL TRANSPORT EXPERIMENTS

The scaling of cross-field heat transport with relative gyroradius ρ^* was measured on the DIII-D tokamak in L-mode and H-mode plasmas using the technique of dimensionally similar discharges. For the first time, the relative gyroradius scalings of the electron and ion thermal diffusivities were determined separately using a two-fluid transport analysis. For L-mode plasmas, the electron diffusivity scaled as $\chi_e \propto \chi_B \rho^{*1.1 \pm 0.3}$ (gyro-Bohm-like) while the ion diffusivity scaled as $\chi_i \propto \chi_B \rho^{*-0.5 \pm 0.3}$ (worse than Bohm-like). These transport scalings were obtained with either central radio-frequency heating (ECH and fast wave) or NBI heating. The gyro-Bohm-like scaling of χ_e is the expected result for turbulence generated by small-scale instabilities, but at present there is no clear theoretical explanation for the ρ^* scaling of χ_i . Since the electron and ion fluids had different gyroradius scalings, the effective diffusivity and global confinement time scalings were found to vary from gyro-Bohm-like to Bohm-like depending upon whether the electron or ion channel dominated the heat flux. This last property can explain the previously disparate published results with dimensionally similar discharges on different fusion experiments. Experiments in H-mode with the expected ITER values of beta, collisionality, and safety factor and with the ITER plasma shape also were done. For these dimensionally similar discharges, both the electron and ion diffusivities showed gyro-Bohm-like scaling, $\chi_e, \chi_i \propto \chi_B \rho^*$, as did the global thermal confinement time. The reason for the change in the ion transport scaling between L-mode and H-mode is not known. Scaling the DIII-D transport results along a dimensionally similar path to ITER, results in a predicted H-factor of 5.4. Thus only 41 MW of heating power in ITER would be required to sustain an ignited plasma with 1.5 GW of fusion power; however, additional heating power may be needed if the H-mode power threshold is larger than 41 MW (as is expected).

4.3. FAST ION CONFINEMENT AND ALFVÉN EIGENMODE ACTIVITY IN DIII-D

Neutral beam injection (NBI) presently serves as the primary source of auxiliary heating and non-inductive current drive for the DIII-D tokamak. Fast ion confinement is thus essential for maximizing the performance of discharges such as VH-mode, high normalized beta (β_N), and high beta poloidal (β_p). However, beam driven instabilities such as Alfvén eigenmodes are of great concern because they can lead to a serious depletion of the fast ion distribution, thereby reducing the heating and current drive efficiencies. In addition, excessive heating and damage to the internal hardware can result during long pulse operation. These same issues apply to a reactor for the fusion generated alphas and high energy, negative ion beams (E~1 MeV) which may be used for heating, momentum input and current drive.

To evaluate the level of fast ion confinement, the measured and predicted neutron yields are compared. In DIII-D, beam-target neutrons, generated by the high energy component of the fast ion distribution function contribute significantly to the measured neutron yield, varying from 25% in VH-mode to over 90% in high β_p

discharges. The predicted yield is determined using the measured kinetic profiles as input for the ONETWO transport code. In cases where discrepancies exist, poor fast ion confinement is believed to be the prime explanation. Discrepancies up to 50% have been identified during VH-mode, high β_N , and following the collapse of high β_p discharges. High β_p discharges are also characterized by obvious disagreements between the predicted and measured stored energy and by a predicted Shafranov shift which is inconsistent with both the radial ECE and SXR profiles.

Several observations point to Alfvén eigenmode activity as the cause of poor fast ion confinement. First, toroidally induced (TAE) and/or beta-induced (BAE) modes are observed during periods of poor confinement. Second, when these modes are present, analysis based on the charge exchange diagnostic indicates that the fast ion content in the core of the plasma is depleted. And third, $W_{\text{parallel}} = W_{\text{mhd}} - W_{\text{dia}}$, a relative measure of the parallel stored energy, saturates and/or collapses during these periods.

In discharges where confinement has been problematic, improved fast ion confinement has been observed under two different conditions, with negative I_p ramps and using of lower energy neutral beams. In the first case, it is thought that the negative edge current density enhances the stability of these modes, probably through increased edge shear and a corresponding increase in the damping mechanisms. In the second case, several items may be contributing. These include: a) a reduction of the fast ion pressure gradient resulting from a broader fast ion deposition profile; b) faster thermalization of the beam ions as a result of their slower velocity; and c) a reduction in the drive efficiency resulting from a larger difference between the Alfvén velocity and the injected ion velocity. A dramatic reduction of the Alfvén activity resulting from a decrease in the injection beam energy is shown in Fig. 4.3-1. The high frequency activity with toroidal mode numbers $n=3$ through $n=8$ is all but nonexistent at the lower beam energy.

Fig. 4.3-1. Comparison of the cross-power spectrum during typical bursts in nominally identical discharges with different beam injection energies, 75 keV versus 52 keV. Nominal conditions: $B_t=0.95$ T, $I_p=0.7$ MA, $\langle n_e \rangle = 2.5 \times 10^{13}$ cm $^{-3}$, $T_e(0)=2.2$ keV.

Future experimental plans are being developed to further assess the observed improvements of negative I_p ramps and lower energy injection. In addition, plans to determine BAE mode scaling and evaluate TAE/BAE activity in ITER-like and reversed shear configurations are in progress.

4.4. ITER ELONGATION STUDIES

It has been suggested that there might be an advantage in increasing the plasma elongation in ITER by reducing the plasma width with fixed plasma height. Several confinement scaling relations predict that this change in plasma shape could result in increased performance, expressed here in terms of a volume averaged fusion triple product $\langle nT \rangle \tau$. If this were done, significant cost savings might result since the physical size of the device could be reduced. Alternatively, one might take advantage of the increased distance from the toroidal field coils (and therefore reduced effect of magnetic field ripple) by reducing the number of coils to improve access for neutral beams. There are also disadvantages. Such a change in the plasma shape would probably necessitate addition of internal coils for vertical position stability and moving the plasma away from the shaping coils would make shape control more difficult.

We were requested by the ITER Joint Central Team to perform an experiment to evaluate this effect on performance. The task was to assess the relative performance of plasmas in high and low elongation shapes, with safety factor, q_{95} , electron density, $\langle n_e \rangle$, (to match the collisionality ν^*), heating power, P_{NBI} , and toroidal field coil current fixed at ITER relevant values. Three such scans were performed, at different combinations of q_{95} and $\langle n_e \rangle$.

We observed enhanced performance in the high elongation discharges of each scan (see Table 4.4-1) with increases in $\langle nT \rangle \tau$ ranging from 13 to 39% when the plasma elongation κ was increased from 1.68 to 1.94.

**TABLE 4.4-1
SUMMARY OF RESULTS FROM THE ITER ELONGATION EXPERIMENT**

Case	q_{95}	$\langle n_e \rangle$ (10^{19} m^{-3})	P_{NBI} (MW)	β_N	$\Delta \langle nT \rangle \tau$ (%)	$\langle T_i \rangle / \langle T_e \rangle$ (low $\kappa \rightarrow$ high κ)
I	3.8	4	5.9	1.8	39	1.2 $\rightarrow$ 1.5
II	3.8	6.5	5.7	1.7	13	1.1 $\rightarrow$ 1.1
III	3.1	6	3.9	1.7	32	1.1 $\rightarrow$ 1.1

In case I, the ratio of ion to electron temperature increased to 1.5 in the high elongation discharge. In subsequent scans, we were forced to accept higher densities (and therefore higher ν^*) in order better to equilibrate the ion and electron temperatures (as expected in ITER). The results of case II led to the contention that the enhancement in performance might be contingent on a high ratio of ion to electron temperature. This was refuted by the results of case III, where a high enhancement was once again observed, but with a relatively well equilibrated plasma. The reason for the relatively small increase seen in case II is still not understood.

In interpreting these results, we must keep in mind that performance expressed as $\langle nT \rangle \tau$ determines not the total fusion power, but rather the fusion power density. Since the plasma volumes decreased by about 15% when the elongation was increased, the total fusion power in the best cases is expected to increase by only about 15%. This is confirmed by the observed neutron rates.

5. OPERATIONS

5.1. OVERVIEW

The principle functions of the DIII-D Operations group are to ensure reliable tokamak operations, develop new capabilities, and refurbish existing systems. During the year there were 47 days of machine operations for experimental research and a similar number of days for preparation, calibration and testing of tokamak and diagnostic systems. The availability of the tokamak during scheduled research operations was 75%. Over half of the year was spent in the installation of new capability.

A number of new capabilities have been developed to meet the needs of the experimental program. Two new ICRF systems were installed, including two 2 MW generators, transmission and coupling systems, and fast wave antennas in the vacuum vessel for each (described in Sec. 6.2). A new correction coil (C-coil) was installed to reduce the field asymmetries intrinsic to the magnetic coil system. This coil consists of six saddle coil segments surrounding the tokamak at the midplane connected in diametrically opposite pairs (Fig. 5.1-1). These coils were

Fig. 5.1-1. Layout of the C-coil as installed on DIII-D.

powered by three power supplies obtained from Lawrence Livermore National Laboratory and refurbished with new control systems. The C-coil has allowed the minimum density for operation of DIII-D to be cut in half. A pneumatic pellet injector has also been installed on DIII-D, in collaboration with Oak Ridge National Laboratory, which will allow modification of the density profile. Argon frosting techniques applied to the neutral beamlines have allowed relatively long pulse (4 sec) helium injection to be achieved.

Important changes both in the machine hall and in the remainder of the building have been made to support the expansion of the ICRF program. Modification of the vacuum vessel to allow installation of two new rf antennas at the locations previously used for movable limiter blades was completed. New bumper limiters were installed to protect the antenna faces. The areas immediately outside the vessel and toroidal field coil were also cleared to provide access for the transmission line.

The areas of the laboratory building to the south and east of the machine hall were reconfigured and upgraded to provide space for the rf tuners and address facility needs for the next several years. Space was provided for the tuners on balconies above the normal work areas. The space immediately below was reconfigured to provide better access to the main machine hall entrance and between the front and back parts of the building. New space near the machine hall was provided for diagnostics along with critical additional space for data acquisition. The vacuum services group moved to improved quarters at the rear of the building.

Refurbishments were carried out to major machine systems in order to help improve the reliability of the facility. A major upgrade of the data acquisition and control computers was begun with the installation of a new AEG computer for the primary data acquisition task. The User Service Center capability was also upgraded with several new state of the art computers to support data analysis and user computation. The primary power supply and discharge control system was modernized to a new programmable logic controller (PLC) based system which has proven to be more reliable and has greater flexibility than the previous system. An ongoing program to upgrade the power supplies and interfaces for the main coil systems has significantly improved reliability while providing increased flexibility to meet operating needs. A number of facility support systems (heat exchangers, cooling towers, etc.) have been upgraded to replace systems that had reached the ends of their useful lifetimes.

The preventive maintenance program is now functioning effectively to ensure that the equipment receives timely service, and it is serving as a basis for evaluating our maintenance activity.

Forty-nine diagnostics are now operating on DIII-D, and the task of maintaining them is significant. Attention is now being paid to the refurbishment of productive older systems to ensure that they remain reliable. Recent additions to the diagnostic set emphasize better understanding of the divertor configuration.

5.2. TOKAMAK OPERATIONS

The year 1994 was characterized by a significant number of system upgrades and new hardware installations that both increased the capability of the tokamak and focused on improved reliability. The main program elements are summarized in Fig. 5.2-1. Much of this work was performed during a six month vent from November through April and during an extended break between operating periods in the summer. The major operational highlights of this year included the installation and successful operation of the new ohmic heating system control (EPSSIC), a highly flexible and user friendly gas distribution system, and the six section error correction coil (C-coil).

Fig. 5.2-1. DIII-D FY94 weekly operations schedule.

Following two weeks of operation in October, the tokamak was vented for six months to install two Fast Wave Current Drive (FWCD) antennas and a number of new diagnostic systems. After a commissioning period for the new systems and a period of tokamak operations for the purpose of cleaning and conditioning the vessel following the vent, five weeks of experiments were performed. This was followed by an intensive six week construction and maintenance period in August and September during which the C-coil installation and tests were completed. A brief vent in August was conducted for the installation of a Faraday shield on the original FWCD antenna. Operations resumed at the end of September. In addition to experimental operations, the tokamak has also been operated this year for diagnostic calibrations, tests of new hardware systems, and tests of repaired or modified systems prior to their use in experiments. In order to accommodate the large amount of testing and calibration, the "maintenance" weeks are now typically double shift with the maintenance and new construction during the first shift and operations during the second shift. In the third and fourth quarters of FY94, approximately 20 days of non-experimental operations were performed. This work included tests of the new C-coil and C-coil supplies, feedback tests for the ohmic heating coil supply and the new digital control system algorithms, and diagnostic calibration for the MSE, lithium beam, ASDEX pressure gauges, Thomson scattering, and piezvalve flow sensors.

For the year as a whole, there were 47 days of operation scheduled for physics experiments during which time the tokamak availability was 75%. The largest single source of downtime was three days lost for repair of two poloidal field power supplies that were damaged as a result of a cooling water leak in an adjacent power supply. This incident has focused our attention on the need to improve the quality of our hardware installations and in particular to concentrate on improving the test procedures following any repair or installation of new hardware.

The most critical operations system that was upgraded this year was the ohmic coil control and interlock system (EPSSIC). This upgrade was completed on schedule and the related upgrade to the abort interlock system was completed ahead of schedule. The final system incorporated a combination of a medium speed programmable logic controller, high speed CPU, and hardware interlock and protection circuitry. This design provides a highly flexible system for future growth, improved identification of system failures, and high degree of reliability because of system design and redundancy. Because of the critical nature of the systems, extensive test procedures were developed. Problems were identified and repaired during this testing phase and the system has operated without failure during the year.

As part of our effort to improve systems that cause significant downtime, a major upgrade of the gas distribution system was undertaken during the vent period. The new hardware provides significantly more flexibility in the choice of gases, and valves have been added to accommodate the more demanding requirements of the radiative divertor experiments being performed this year. The new system also has an improved vacuum design to reduce impurities in the working gas. A highlight of the new system is the user friendly graphical interface coupled with a programmable logic controller for system control. These two features provide flexibility in control and ease of upgrade, and have virtually eliminated operator error which was the major source of downtime associated with the previous system. As a result of our satisfaction with the new user interface, we are proceeding with the replacement of the old user interface for the vacuum and air/water cooling system. This work is expected to be completed in early 1995.

A major development this year was the installation and successful operation of the new error field correction coil (C-coil). The full operational system associated with the C-coil includes six saddle coil segments surrounding the vessel midplane, four power supplies, a patch panel for making intersegment connections and for connecting the supplies to the coil, instrumentation for measuring the coil currents and voltages, and a real-time control algorithm that calculates the proper currents in each of the coil segments. The six section coil can produce an $n=1$, 2, or 3 field either to reduce the existing machine error field or produce other error fields to study their effects. Initial results of C-coil operation to reduce error fields indicate the minimum obtainable density without locked modes has decreased significantly. In our standard 1.6 MA discharge, the C-coil has permitted densities more than a factor of two lower than those achieved using the previous crude error correction coil, the $n=1$ coil. The low density locked mode threshold is reduced to a minimum (Fig. 5.2-2) for a particular set of phasings for the C-coil segments which yield a specific perturbative field which cancels intrinsic error fields from the DIII-D coil set. The lower densities have already made it possible to obtain VH-mode operation at lower plasma currents and have provided the required low density target plasma for the rf current drive experiments. In addition to error field correction, the coil has also been used in conjunction with the $n=1$ coil, to magnetically brake the plasma rotation. The short L/R time of the new coil makes it ideal for rapidly affecting the rotation and studying its effects on high beta stability.

Fig. 5.2-2. Low density locked mode threshold: stable operation at low density depends on the phase of the C-coil.

5.3. DISRUPTION STUDIES

5.3.1. INTRODUCTION

Tokamak discharges are sometimes terminated by a disruption, an instability resulting in a very sudden loss of the plasma thermal and magnetic energy. Due to the significance of disruption related thermal and mechanical loads for future fusion devices such as ITER, experiments dedicated to studying disruptions have been performed in the DIII-D tokamak both to further characterize disruptions and to determine a method to provide reproducible disruptions with high heat fluxes for future experiments using DiMES. This year, for the first time, experiments dedicated to disruption studies were performed, with the initial concentration on forced disruptions using Argon impurity injection in ohmic and beam heated discharges. This dedicated time led to a large increase in the amount of data describing the disruption phases. The measurements include the first temperature and density profiles obtained through the thermal and current quench with the Thomson scattering system operated in the burst mode and the first direct measurements of the flattening of the current profile using the MSE diagnostic. The disruption database continued to automatically receive information including disruption related parameters on shots with disruptive termination.

5.3.2. FY94 DISRUPTION STUDIES

Analyses of the disruption rate and disruption frequency from the disruption database were updated and forwarded to TPX for input to the TPX design. During the analysis of data from the disruption database it was discovered that small volume plasmas have a higher disruption rate than our standard full size plasmas and some of

the shapes were similar to plasmas in CMOD which is also reporting higher disruption rates than found on the average in DIII-D.

Radiative collapse disruptions were the first disruptions studied using dedicated machine time; both the thermal quench and the current quench behavior were examined. Before the disruption, an $n=1$ mode was observed, which grew prior to the thermal quench and had further rapid growth prior to the current quench. Measurements were made of temperature and density profiles obtained before, during and after the thermal quench and after the start of the current quench, and of current profiles through the disruption. The temperature profiles show the cooling of the edge at the start of the thermal quench followed by a central temperature collapse at the final thermal quench with a further temperature decay after the current quench. Density profiles measured through the disruption show an increase at the edge at the start of the thermal quench, and the development of an edge density bump, while the central density remained relatively unchanged. The current profile, measured using the MSE diagnostic and the plasma equilibrium, flattened at the final thermal quench as shown in Fig. 5.3-1. The current measurements of the

Fig. 5.3-1. The current density profile flattens prior to the final thermal quench.

scrape off layer current show a toroidal asymmetry ($n=1$) with the tile current peaking late in the decay of the current. Disruptions, during which plasma struck a DiMES sample, also were made and the sample is being analyzed to determine the ion energy spectrum incident on the divertor.

5.4. RADIATION MANAGEMENT

5.4.1. FY94 RADIATION MANAGEMENT

Radiation management tasks include monitoring the site boundary radiation, monitoring the dose exposures of individuals during between shots access and vessel entries, ensuring compliance with legal limits, DOE guidelines and DIII-D procedures, monitoring material for activation, maintenance and operation of the radiation monitoring detectors (neutron and gamma), and maintaining a database of radiation measurements and of personnel dose exposures.

The total neutron radiation at the site boundary for FY94 was 4.1 millirem, and the total gamma radiation was 1.4 millirem, giving a total site dose from DIII-D operations of 5.5 millirem for the year. This is well below both the DOE SAN annual guideline limit of 40 millirem and the California annual limit of 100 millirem.

The total dose exposure received by personnel was kept below the DIII-D procedural limits of 25 millirem per day, 100 millirem per week, and 300 millirem per quarter (1200 millirem per year). The highest dose accumulated by an individual from between shots access and vessel entries (but not operations) for FY94 was 307 millirem. A total of 135 individuals received doses with 108 of the doses being below 25 millirem. All doses were logged in the database of personnel radiation doses.

The ALARA subcommittee established goals for the coming year. The goals are 0.15 person-rem per mrem of site boundary dose and a maximum of 350 mrem per person from in vessel work.

Audits of the radiation monitoring program were held by the Criticality and Radiation Safety Committee, and by Health Physics. The checkout test procedure performed before the start of operations after a long down time was added to the preventive maintenance list. A design review of the radiation monitoring system was held and actions for improving the fail safe features of the system were determined. These have now been incorporated.

5.5. NEUTRAL BEAM OPERATIONS

5.5.1. OPERATIONS SUMMARY

Eleven weeks of plasma heating experiments were supported by neutral beams in FY94, and eight beam systems were available to support the experiments.

Water leaks in the neutral beam ion source gradient grid assemblies are a serious concern. Three ion sources had water leaks due to hardware failure in FY94. One bellows of the 330° left source gradient grid water cooling line developed a crack in December, and plasma grid modules of the 150° right and left sources developed water leaks due to failure of braze joints between bellows and rail holders in April and September, respectively. These

three ion sources were successfully repaired at DIII-D after receipt of spare parts from Burle Industries with the approval of PPPL. They were tested and then conditioned and returned to operational status. Analysis of these faults including metallurgical tests have not identified a cause for these failures. Possibility of fatigue failure due to vibration is being investigated.

5.5.2. SYSTEM AVAILABILITY

Availability of the neutral beam system by month is shown in Fig. 5.5-1. The "Available" category is based on the beam system required for the physics experiments. The difference between the "Available" and "Injecting" categories represents beam systems which were available but were not used for injection during physics experiments. The various causes for downtime are shown in Fig. 5.5-2. Cryogenic and power supply problems are the two largest downtime categories.

Fig. 5.5-1. Neutral beam availability by month.

Fig. 5.5-2. Causes of downtime by category (percent of total operating time).

5.5.3. SYSTEM IMPROVEMENT, MAINTENANCE AND TESTING

Several improvements have been made to the neutral beamlines to improve the reliability and capability of the system. A neutralizer gas interlock system was installed, which will terminate the beam pulse if gas flow into the neutralizer is not sufficient to prevent several beamline internal components from overheating by excessive energetic ion bombardment. This system has been checked out on all eight beam systems and has worked very well since being put into service.

An argon frosting system for helium gas pumping inside the beamline was designed and installed on the 150° and 210° beamlines in FY94 to enable production of helium neutral beams for helium transport and exhaust experiments. The argon frosting system has been successfully tested and can pump helium gas from two ion sources for five seconds. This will provide four second helium beam pulses from each ion source (as compared to 0.5 second helium beam operation without helium gas pumping).

A portable RGA/leak-checker system has been built and checked out. After baking, it had excellent base pressure of 6×10^{-8} Torr. This system replaces the old RGA system in the ion source lab to improve capability for source maintenance and repair.

In general, the arc chambers of the eight ion sources are in very good condition. Scheduled preventive maintenance and inspections were performed on all eight ion sources. This task included external visual inspection, measuring resistance between the plates of the arc chamber, checking filament resistance with low current, leak-check of the cooling water channels internal to the source, and replacing Langmuir probe tubing and some water fittings.

The four torus isolation valves (TIV) between the beamline and torus were found to be leaking badly. After removal of the paddles of the TIVs from the beamlines and performance of required maintenance, these paddles were reinstalled and the air pressure for valve locking was tested to meet the manufacturer's specification. Based on the past two year's experience, these TIVs will need yearly maintenance.

The internal components of all four beamlines were inspected during the five month shut-down period, and no new damage was found except for minor melting spots at the 150° beamline magnet louvers. The magnetic field strength of the 150° bending magnets has since been increased slightly, which should prevent further damage to the louvers. The gaps at the apexes of six calorimeters (except 210°) were found to have opened up enough that energetic particles could leak through and onto the TIVs. A copper plate was installed at the apex of each of the eight calorimeters to prevent this leakage.

5.6. COMPUTER SYSTEMS

During Fiscal year 1994, data from 2875 individual tokamak cycles were acquired. This includes plasma shots, calibration shots, and other test shots. On the largest single shot 109.5 Megabytes were acquired. Over 230 Gigabytes of data were collected during the year (this compares to 127 Gigabytes in FY93). This major increase in the size of the database and in computer usage motivated substantial changes in our computer systems both for data acquisition and analysis at DIII-D and for general computational use at the User Service Center.

The data collected from a DIII-D shot previously had been amassed into a single large computer file. During FY94, this mode of operation was changed. Multiple files are now collected for each shot; generally each file is

made up of data from particular diagnostic experiments. The advantages of this change include that shot data are available to the user more quickly after a shot, more data easily can be collected without unduly affecting the length of the shot cycle, separate diagnostic systems can use the shot collection software to automatically handle and archive little used data, and more useful data can be stored on disk since all files for a particular shot do not need to reside on the disk. Client/server software was developed so that users can transparently access data which can be located on many different computers and workstations. Also, software was modified to handle the multiple files, archive it efficiently, and manage the disk space of the involved systems.

A new UNIX-based real-time computer system is in place, replacing two old 16-bit proprietary MODCOMP machines for data acquisition. The UNIX system uses a faster CAMAC serial highway, VME bus and much faster Motorola 88110 processor. The overall speed is four to five times faster than the old system, using only one computer and one serial highway. Development efforts included a new database, new programs to cycle through setting and reading the CAMAC modules, and programs to gather, format and archive the data. Over 50 Mbytes of data were moved to the new system, with only a small amount remaining to be transferred as of November, 1994. This deployment has provided DIII-D with more reliable, faster, flexible capabilities which were not previously available.

A very fast Hewlett-Packard (HP) workstation was purchased and located in the DIII-D control room to allow users to view and analyze data quickly during operations. When the tokamak is not operating, the workstation is available for general use. This reflects our general movement away from the VAX systems to HP UNIX systems for the physics users. In order to make this workstation immediately useful, the REVIEW program was ported from the VAX/VMS systems to the HP systems. This program is the primary tool for examining waveform data between shots. The program is now running routinely on HP workstations from X-window terminals. Because of the significantly greater cpu power of the UNIX systems, the program plots data faster, thus enhancing data examination and operational decision making between tokamak shots.

During this past year a major computer system was selected and installed at the User Service Center. An HP T-500 Computer was integrated into the User Service Center computer configuration. The T-500 has three CPUs and 512 MB of memory. It is available for general purpose computing, and is open to all Fusion computer users. There are 16 Gigabytes of disk space dedicated to DIII-D shot data. The T-500 also hosts a 1 Gigabyte scratch area that is mounted on all the HP computers. The scratch area is used heavily during DIII-D operations as a repository for between shot EFIT MHD calculations. These calculations are now available to all interested parties, not just those few on the dedicated workstation in the DIII-D control room. The hardware selection process involved an internal committee to evaluate equipment relative to existing and future needs, recommendations and contact with many outside organizations including other DOE users, and a final external review.

The GA Corporate computer center procured an Auspex File Server for the purpose of providing a central file service capability. The Fusion Division took advantage of this powerful processor and added 18 Gigabytes of disk space for their exclusive use. These disks are available on all HP computers so that users see a common file system. This has been a very successful addition to the Fusion computing resources.

The ONETWO transport/MHD code was ported from the CRAY computers at NERSC to our local HP systems. ONETWO consists of about 63,000 lines of Fortran code, plus another 40,000 lines in math libraries, graphics and ECH support codes and utilities. Only about 500 of these lines now differ across platforms, for 99.5% portability. User throughput and interoperability with other Fusion codes are both vastly improved over the old CRAY configuration, resulting in far greater usage of the codes than before. The code is in production use and has seen about one thousand executions since mid-September.

5.7. RELIABILITY AND AVAILABILITY

5.7.1. INTEGRATED PREVENTIVE MAINTENANCE PROGRAM (IPMP)

There were 2482 maintenance tasks identified for all of FY94. Of this number; 2162 preventive maintenance (PM) and 259 corrective maintenance (CM) tasks were completed. There were an additional 486 tasks which were scheduled and deferred, and there were 99 tasks not completed due to lack of manpower. During FY94 safety training and related safety issues were added to the IPMP for tracking and implementation. Additionally, the IPMP data base is being used more proactively: 1.) to improve the reliability of equipment based on life histories; 2.) to identify equipment with problem histories; 3.) to identify equipment needs based on inventory list of equipment status and; 4.) to locate equipment within the facility.

At the end of FY94 there were 1313 pieces of equipment being tracked by the IPMP. This is a 9% increase over FY93.

5.7.2. SIGNIFICANT EVENT REVIEWS (SER)

In March of 1991, management initiated a Significant Event Review process to understand the root cause of equipment and/or operational problems and thus eliminate the recurrence of these events. There were six significant events this year each of which caused more than 4 hours of DIII-D tokamak downtime. There was also one event, a fire in a chopper which powers one of the DIII-D field shaping coils, that did not cause any downtime, but caused equipment damage. The cause of the fire was that the chopper had been incorrectly connected during a bake cycle. All procedures and the checklist had been followed; but these procedures did not take into account the configuration of the previous day's unusual testing mode. As a result of this, procedures and the checklist for connecting the choppers and checking the setup have been revised. Four of the significant events were caused by human errors which we are trying to minimize by reviews, training and education. A significant event not traceable to human error also occurred when failed water hoses sprayed water over the machine delaying operations. Many of the hoses on the DIII-D tokamak are in the process of being replaced.

5.8. MECHANICAL ENGINEERING

5.8.1. TOKAMAK SYSTEMS

A vent early in the report period was concluded on time and the vacuum vessel was pumped down on March 18. The following in-vessel tasks were completed: installation of the new Langmuir probes; installation of the edge MSE diagnostic; installation of the 0 and 180 degree FWCD antennas; insulation of upgrades to the ADP ring; installation of the in-vessel hardware for the bias ring IRTV; installation of the divertor interferometer; rework of the tile current monitors; installation of the UCLA microwave reflectometer; installation of the 165 degree bolometer shield; installation of two moveable glow discharge electrodes; and calibration of several diagnostics. The vessel was subsequently vented from April 6 through April 15 to complete the installation of the Faraday shields and adjacent tiles, install new bumper limiter tiles, and correct ADP ring voltage standoff problems.

5.9. ELECTRICAL ENGINEERING

5.9.1. OVERVIEW

The primary responsibility of the electrical engineering group is to make all power systems available for operations. In addition, the group completed improvements to the specific systems which had contributed most to downtime during FY93 operations and developed some new capabilities. In particular, the new EPSSIC interlock system has provided new flexibility in determining the cause and sequence of events in aborted shots. Four power supplies were modified and installed to serve the magnetic field correction coil system. A new gas delivery system was placed in operation during the year. A new CER CCD diagnostic system was completed and will be used to acquire data in FY95. Finally, the down time attributable to the poloidal field (F) power system CAMAC interface and the circuit breakers for the toroidal field (B) power supply and ohmic heating (E) power supply has been significantly reduced by recent refurbishments.

5.9.2. OPERATION SUPPORT

The electrical systems performed well throughout the year, and were used in forward and reverse toroidal and poloidal field configurations during operations. Minor delays in operations resulted when a fire caused by a patch panel setup error destroyed one chopper. A water leak in one of the neutral beam modulator regulator systems also caused minor delay. Precautions have been implemented to prevent reoccurrence of the patch panel error, and to minimize the impact of water leaks in the modulator regulator equipment area.

Tests to validate and characterize models of the DIII-D coil/vessel and electrical power systems also were completed. The models are being used in the development of new control schemes for the DIII-D tokamak.

Approximately 5414 man hours were used by the electrical engineering group to complete 1483 preventive maintenance tasks this year.

5.10. DIAGNOSTICS

5.10.1. OVERVIEW

The DIII-D diagnostic effort in FY94 emphasized expanding the divertor diagnostic set but also included some improvements and additions to the core diagnostics. The emphasis on the divertor diagnostics was aimed at exploiting the pumped divertor and the relatively good diagnostic access available with the existing divertor configuration while preparing for the radiative divertor installation planned for FY96. Concrete steps were taken toward making the critically important detailed measurements of the density and temperature in the divertor with the installation of a two channel microwave interferometer viewing across the divertor legs, and with the design of three new divertor diagnostics, a divertor Thomson scattering diagnostic, a fast stroke Langmuir probe in the X-point region and a divertor reflectometer. There are 49 major diagnostic systems now operating on DIII-D with six new diagnostic systems in some stage of development. Table 5.10-1 lists the major diagnostic systems along with the quantities measured. In addition to the development of new diagnostic systems, we refurbished three diagnostics that

TABLE 5.10-1
PLASMA DIAGNOSTICS

Diagnostic	Quantity Measured	Comments
<u>Operations-Related Diagnostics</u>		
Magnetics	$\dot{B}$, Magnetic flux, I_p	2 poloidal (58), 1 toroidal (8) mag. probe arrays with 50 kHz response; 41 flux loops, 9 diamagnetic loops, 30 saddle loops, 3 I_p Rogowskis
Hard X-rays		2 toroidal locations, 4 detectors, 1 kHz response
Plasma TV	Visible light	Radial view, divertor, rf antennas
IR cameras	Heat load to armored surfaces	Upper divertor; inside wall, 2 toroidal locations, lower divertor
H_α monitors	H_α radiation, recycling	H_α filtered, 16 locations, 10 kHz response
Neutron detectors	Fusion and photo-neutrons	3 toroidal locations, 200 Hz response
Soft X-ray arrays	Internal fluctuations	1 vertical, 1 horizontal, 32 ch. ea. ~ 4 cm resolution, 4 toroidal locations, 12 ch. ea. > 250 kHz response,
<u>Electron Profiles</u>		
Multipulse Thomson profile	$T_e(r, t)$, $n_e(r, t)$	8 lasers, 6 ms, vertical profile, 40 pts., < 1 cm in edge resolution, $20 \text{ keV} > T_e > 10 \text{ eV}$
CO ₂ interferometers	$\bar{n}_e(t)$	Vertical: 3 chords; radial: 1 chord
ECE Michelson	$T_e(r, t)$	Radial profile, each 25 ms
Microwave radiometer	$T_e(r, t)$	32 ch. (ORNL), 0.2 ms, 2 cm resolution
<u>Fluctuations</u>		
Microwave reflectometer	$n_e(r, t)$, $\bar{n}_e(r, t)$	UCLA collaboration – broadband; system gives profile in 5 ms, narrowband system, 400 kHz bandwidth
Correlation reflectometer	Radial correlation lengths in $\bar{n}$	UCLA
FIR scattering	$\bar{n}_e(r, t)$	UCLA collaboration – 4 MHz bandwidth
CO ₂ scattering	$\bar{n}_e(r, t)$	MIT collaboration – phase contrast 1kHz, 100 MHz bandwidth
Fast magnetic probes	$\bar{B}_\theta$	1 MHz resolution, 2 arrays totaling 7 probes
Li beam	Edge $\bar{n}_e(r, t)$	Turbulence measurement near the edge

TABLE 5.10-1
PLASMA DIAGNOSTICS (Continued)

Diagnostic	Quantity Measured	Comments
<u>Ion Temperature and Rotation</u>		
Charge exchange recombination	$T_i(r, t)$, $v_\phi(r, t)$, $v_\theta(r, t)$	16 channels bulk plasma; 16 channels edge plasma; 3 mm T_i edge resolution; 6 mm v_ϕ , v_θ , edge resolution
<u>Impurities and Boundary Parameters</u>		
Visible bremsstrahlung	$Z_{\text{eff}}(r, t)$	Radial profile, 16 chords, 1 kHz
Bolometer arrays	Radiated power	Tomographic reconstructions, 3 arrays, 69 ch., 1 kHz
Graphite foil bolometers	Heat load to the wall	7 on the outer wall, 2 inside wall, 2 lower divertor, 1 upper divertor
SPRED (dual range)	Impurity concentrations	280-1200Å; 100-290Å, spectrum every 1 ms; edge profiles with scanning capability
H α TV	Divertor H α	LLNL
Divertor IR cameras	Heat load to divertors	LLNL, GA, 125 ms for a profile, upper, 2 lower
Langmuir probes	Edge $T_e(t)$, $n_e(t)$	On divertor tiles
Fast stroke Langmuir probe	Edge $T_e(t)$, $n_e(t)$	SNL, UCLA
DiMES calorimetry	Surface erosion	Divertor Material Exposure System
Penning gauges	He pressure	ORNL – He pressure under the divertor baffle
Divertor microwave interferometer	$\bar{n}_e$ in divertor legs	2 chords (LLNL)
Fast pressure gauges	Neutral pressure	Near divertor region (ASDEX, LLNL), toroidal array, 2-3 ms response
Tile current monitors	Poloidal tile currents	Includes tile currents in the ADP, toroidal array
Multichannel divertor visible spectrometer (MDS)	Divertor impurity concentrations	ORNL – 7 channels
Divertor Baratron gauge	Neutral pressure	ORNL – verified ASDEX gauge measurements

TABLE 5.10-1
PLASMA DIAGNOSTICS (Continued)

Diagnostic	Quantity Measured	Comments
<u>Fast Ion Diagnostics</u>		
Scintillator	Neutron fluctuations	50 kHz response time
E B charge exchange	Beam-ion density profile	Conventional charge exchange analyzer coupled with H α detectors used to measure beam neutral density profile, scannable, spectrum in 1 ms
Fusion products	Fast neutrons, tritons He ³ , other fast ions	UCI collaboration; movable probe assembly
<u>Current Drive Diagnostics</u>		
Motional stark effect (MSE)	$B_p(\tau)$	8 radial channels; 2 cm resolution, 5 ms response
ECE Michelson	Microwave emission	Detects tail population in the electron distribution function, profile in 25 ms (LLNL and Maryland)
SXR pulse height	High energy X-ray spectrum	Russian collaboration; detects tail population in electron distribution; spectrum every 100 ms
RF probes	RF radiation	Phase and amplitude of ICH rf on inside wall
<u>In Development</u>		
Divertor reflectometer	Peak n_e in divertor	12 ch. in lower divertor (LLNL)
Divertor Thomson scattering	$n_e(\tau)$, $T_e(\tau)$ in divertor	Design work only in FY94 (LLNL, GA)
Beam emission spectroscopy	$\bar{n}_e(\tau)$ in the core	Design work only in FY94
Divertor SPRED	Impurity concentrations in divertor	LLNL
Visible SPRED	Impurity concentrations in divertor	Many chords
X-point fast stroke probe	$T_e(\tau)$, $n_e(\tau)$ in divertor	UCSD, SNL

were known to be operating with critical components near or beyond their life expectancies. Many of the diagnostics systems on DIII-D have been operating for many years (some were initially built for Doublet III) and we expect the refurbishment effort to continue throughout the life of DIII-D.

5.10.2. NEW OR UPGRADED DIAGNOSTICS

In FY94 the DIII-D resources for new diagnostics were concentrated on the divertor diagnostic set. Measurement of the electron density contours in the divertor is desirable in order to validate divertor models and to develop methods for heat flux control. Because of the complicated plasma geometry in the divertor and access restrictions, it is extremely difficult to make complete density measurements in the divertor. We are approaching this problem by installing four complementary diagnostics.

A two channel interferometer was installed by LLNL with views across the inner and outer legs of the divertor. The interferometers provide line averaged density measurements across the legs and will be used to calibrate a Thomson scattering diagnostic that has been designed and is now being fabricated by LLNL. The Thomson scattering diagnostic will provide a very detailed density and temperature map of the divertor (made possible by scanning the location of the X-point). A fast stroke Langmuir probe that will plunge from the bottom of the tokamak into the divertor and an upward looking reflectometer will complete the divertor density diagnostic suite when they are installed in FY95. The design for both diagnostics was essentially completed in FY94.

In FY94 a tangentially viewing IR camera with a view of the inside surface of the bias ring was installed. This camera allows monitoring of the inside portion of the bias ring that is exposed to large amounts of power during pumping experiments. The camera has optics that allow it to view the RF antenna at 0 degrees during ICRF experiments.

In collaboration with IPP Jülich, a colorimetry diagnostic was added that allows *in situ* measurements of erosion and re-deposition of thin films on the DiMES sample.

Improvements to the core diagnostic set included the addition of eight spatial channels on the MSE diagnostic by LLNL. Doubling the number of channels allowed accurate and detailed measurements of the current profile to be made. This unique capability quickly yielded a number of exciting new results, including measurements of very hollow current profiles with central safety factor of greater than five. In collaboration with ORNL, the ATF ECE diagnostic was installed on DIII-D. This diagnostic provides electron temperature profiles with submillisecond time resolution and spatial resolution of about 2 cm at 32 radial points.

Three aging diagnostics have undergone major refurbishment. A new detector system using a fast readout CCD has been built and tested for the edge CER diagnostic. These new detectors replace obsolete image intensifiers and will be installed in FY95. The CO₂ density interferometer system was overhauled. Two new CO₂ lasers were installed and two others were refurbished, along with the replacement of other optical components. The 112 soft X-ray detectors were replaced with modern detectors and front end electronics. The new detectors have greatly improved noise characteristics.

5.10.3. DIAGNOSTICS UNDER DEVELOPMENT

The largest and most important new diagnostic planned for installation in FY95 is the divertor Thomson scattering diagnostic. This diagnostic will provide detailed temperature and density measurements in the divertor region. The divertor Thomson scattering diagnostic will be installed in the spring of 1995 and will operate until the radiative divertor is installed. The geometry of the radiative divertor is such that it may be incompatible with the divertor Thomson diagnostic. We are therefore planning to install diagnostics in the divertor that will take advantage of the Thomson scattering measurements while we have them available. With the density and temperature measurements from the Thomson diagnostic and spectroscopic measurements, it will be possible to model the impurity concentrations in the divertor. With this in mind, in FY95, LLNL will install a SPRED instrument with a view through the divertor and GA will install a visible spectrometer with good spatial resolution in the divertor.

The divertor reflectometer and the X-point fast Langmuir probe will be installed in FY95. The X-point probe will allow measurements of the density and temperature in the divertor (with more restrictions than the Thomson diagnostic), but more importantly it may provide flow measurements in the divertor.

6. PROGRAM DEVELOPMENT

6.1. ECH UPGRADE PROJECT

There were only two activities associated with the ECH development program in FY94, both were performed in companies outside of GA: Gycom, a Russian company, was placed under contract to GA to fabricate a 1 MW 110 GHz long pulse gyrotron; and Varian was supported by the DOE I&T Division to also develop a 1 MW 110 GHz long pulse gyrotron. Details of their activities are given below.

Gycom activities were phased, and used a short pulse gyrotron to validate the rf design and a long pulse gyrotron to address the heat load issues. Gycom tests of their short pulse prototype gyrotron achieved 1 MW pulses 50–100 μ sec in duration. The efficiency of the tube at the 1 MW output level was $36\pm 2\%$. No undesired mode was observed, either at 110 GHz or other frequencies. Gycom then rebuilt this short pulse gyrotron with slight modifications to the gun. This tube has now operated at more than 1 MW with an efficiency of 40%. The performance of the tube is such that no further changes to the rf design are anticipated. The production tube is still on schedule to go into test in December, 1994.

With funding from the DOE I&T Division, Varian is developing a 110 GHz 1 MW gyrotron with an internal mode converter similar to the Russian design. On March 19th, Varian produced rf power from their gyrotron for the first time. An important note on this achievement is that rf production came less than two weeks after first conditioning efforts were started. Varian has reached 500 kW, 1 μ sec with this new gyrotron. Some unexpected oscillations were observed in the launch section of the tube. The additional power loss caused by these oscillations may limit the power, but it seemed that >200 kW cw should have been achieved, and this was the ITER goal. However, during testing Varian was only able to achieve about 50 kW average power. This power level was limited by arcing in the dummy load. Arcing was believed to have been caused by an appreciable non-gaussian component in the rf power exiting the window. The plan is to rebuild the tube with a new launcher configuration to improve the mode purity of the output and to eliminate the post cavity oscillations that have been observed. Before ending testing, Varian was able to test GA's distributed window aluminum mockup, and, at a 50 kW power level, 96% or better throughput was observed with little or no observable effect on the operating conditions of the gyrotron. This result is consistent with our expectations.

Varian also completed the design of a new tapered launcher and mirror system. Cold test prototypes were made and shipped to MIT and the University of Wisconsin for testing. Based upon the successful cold test data, Varian rebuilt their 1 MW internal mode converter gyrotron with a new launcher and mirror system. This new configuration produces a flat output profile and should allow the tube to operate at output power levels up to 300 kW cw. During bake out the tube developed a vacuum leak on one of the mirror cooling circuits and the tube had to be opened for replacement of the cooling tubes. The new gyrotron should be back into bake out by early December.

6.2. 4 MW FAST WAVE UPGRADE PROJECT

The FWCD program on the DIII-D tokamak is a collaborative effort between General Atomics (GA) and the Oak Ridge National Laboratory (ORNL), with ORNL providing the design and fabrication of the FWCD antennas and GA providing the technology to generate, transmit, and couple the high power rf to the antenna. Both parties participate in the experimental activities.

The 4 MW upgrade consists of two systems each having its own 2 MW transmitter connected to a four strap antenna by an all ceramic insulated coaxial transmission line. The transmission line is configured to provide the flexibility of adjusting the phasing of the straps for electron heating ($0, \pi, 0, \pi$) or for current drive ($0, \pi/2, \pi, 3\pi/2$), while providing matching between the antenna-plasma impedance ($\approx 1-4 \Omega$) and the 50Ω impedance that the transmitter requires for optimum power delivery. The transmission system must also compensate for the mutual inductance between the straps. This has been achieved by using a decoupler developed earlier on DIII-D.

An overall schematic of one of the two new transmitter/antenna systems is shown in Fig 6.2-1. The two systems are topologically the same, although the routings of the transmission lines are different in order to conform to the DIII-D building layout. The transmitters, transmission lines and the four strap antennas are described in more detail in the following sections.

Fig. 6.2-1. Schematic for one of the 2 MW 30-120 MHz fast wave current drive systems.

The high power rf transmitters are being supplied by THOMCAST AG, formally Asea Brown Boveri Infocom (ABB). These transmitters are of the same type as those in service on the ASDEX-Upgrade experiment at the Max-Planck Institute for Plasma Physics. The maximum output power depends on the frequency and VSWR matching presented by the load. This power capability is given in Table 6.2-1, and even though the maximum power decreases above 80 MHz, in general, and throughout this report, the transmitter is referred to as having a 2 MW capability.

The transmitter consists of four stages of rf amplification. The input signal is fed to the 50 W rf preamplifier through 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 4CW 150000 tetrode made by Eimac. The output circuit is a coaxial quarter wave 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 frequency settings, recalling the stored tuning element locations is all that is required. The specifications for the transmitter are given in Table 6.2-1.

Table 6.2-1
Specifications of the ABB Type VU 62 B1 Transmitter

Frequency Range	30 MHz to 120 MHz	
Bandwidth	±0.25% -1 dB ±75 kHz to ±300 kHz -1 dB	
Output Power	from 30 MHz 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 sec maximum	
Duty Cycle	10% maximum	

Each of the two transmission lines consists of sections of 23.3 cm diameter and 5.6 cm diameter coaxial lines, a switch, a splitter, a dummy load, phase shifters, stub tuners, gas barriers, test sections and diagnostic equipment as depicted in Fig. 6.2-2. The transmission lines and components have ceramic insulators which provide 25 to 50 ohm impedance depending on the dimensions, to maximize the power delivered from the transmitter. Both of the transmission lines to the two antennas consist of the same equipment, but due to the DIII-D building layout, one line is approximately 60 meters longer than the other. For the sake of discussion, only one transmission line will be referenced in the discussion of the two systems.

The design of the transmission line system is straightforward. The generator power is split into the A and B sides with a 3 dB hybrid junction, optimized for the primary 60 to 120 MHz band of operation. A four-port, two-way switch is used to direct any reflected power through the hybrid to the dummy load.

Each of lines A and B contains a standard line-stretcher/stub-tuner pair, and side B has the main 360° phase shifter. There is a 5-way junction at the end of each line where the 23.3 cm line transitions to a pair of 15.6 cm 25 Ω lines forming a resonant loop with a pair of current straps on the antenna, and a decoupler/nuller which compensates for the circulating power that is created by the mutual coupling between resonant loop A and resonant loop B.

Fig. 6.2-2. Block diagram of the RF transmitter amplifier stages.

Current drive efficiency is maximized by using antennas which launch waves at the peak $\eta_{||}$ in the plasma for FWCD while simultaneously minimizing the power launched in the opposite direction. The variables affecting antenna directionality are the number of radiating elements, their width and configuration, strap-to-strap spacing, the spacing from the back wall and sides of the port box, frequency, phasing, and the distance from the plasma edge.

The FWCD antennas for DIII-D were designed to be modular with each antenna consisting of four modules mounted side by side. The two antennas are 180° apart toroidally. Each module of an antenna is fed by its own vacuum feedthrough, capable of handling 1 MW for pulses up to 10 seconds long. This means that each antenna could support a total power of 4 MW with appropriate antenna/plasma loading.

Each antenna module consists of a cavity box, Faraday shield, current strap, and vacuum coax. The cavity box, Faraday shield and current strap are tilted to match the pitch of the magnetic field lines at the surface of the plasma for optimum rf coupling. Cooling channels are fabricated into the modules. The Faraday shield rods specifically do not have water flowing through them so that if a rod were melted or damaged a water leak into the tokamak would not occur. The Faraday shield rods are therefore only edge cooled and are the limiting factor for the pulse length.

The current straps are grounded at two locations, one end of each strap is grounded at the bottom of the cavity, while the other end is grounded to the side wall of the cavity midway up the wall. The feed to the strap is attached to the back of the strap where the folded strap comes closest to the cavity wall. The folded strap design allows the antenna to have effective performance at frequencies above 80 MHz while being effective at lower frequencies as well. The cavity walls are recessed to increase directionality. All parts have been nickel plated to reduce sputtering and thus maximize voltage standoff capability.

6.3. RADIATIVE DIVERTOR UPGRADE PROJECT

In the Radiative Divertor Upgrade Project (RDP) we are designing a new divertor for DIII-D. We have used input from tokamak experiments with the present advanced divertor and modeling to guide the design. The radiative divertor is designed to reduce the heat flux on the divertor plates by radiatively dispersing the power in the scrape-off layer.

In present experiments, (see Section 2.2), we have demonstrated heat flux reduction using either deuterium or impurity gas puffing. Other experiments have started to explore density control with the cryopump in both D₂ and impurity puffing experiments. Several important results from these experiments are being used to design the radiative divertor:

- D₂ puffing reduces the divertor heat flux 3–5 times, with a corresponding increase in the radiation in the divertor which extends from the X-point to the strike point. There is a threshold plasma density for this behavior, and the core density often increases with D₂ puffing.
- D₂ puffing produces a partially detached divertor, *i.e.*, the plasma pressure at the divertor plate (compared to the midplane) is reduced only near the separatrix.
- Impurity puffing produces a similar decrease in the divertor heat flux, but the radiation is divided roughly equally between a mantle around the core and the divertor and there is a rise in the core Z_{eff} to ~3.
- Impurity puffing produces a totally detached divertor, *i.e.*, the power to the scrape-off layer is reduced by the core radiation.

The effects on the plasma core, *i.e.*, the increase in density with D₂ puffing and the increase in Z_{eff} with impurity puffing indicate the need for better control of particle flow between the divertor and the core. The radiative divertor addresses this need by including slots that close off the divertor. Essentially, we are building a divertor “around” our experimental observations of a localized radiation zone, with the goal of improving the control of core effects.

As shown in Fig. 6.3-1, the baseline design includes slots and pumping at four locations: the inner (top and bottom) and outer (top and bottom) strike points. Each cryopump, similar in design to the existing pump in the advanced divertor, is independently controlled. The divertor structure permits operation in either single or double-null plasma configurations. The capability to apply bias voltages is an integral part of the design and is based on experience gained on DIII-D and the Tokamak de Varennes. Biasing capability will be included in the initial design, but it will be installed during one of the modifications to the Radiative Divertor following initial installation.

Computer models have also been very important in guiding the design, particularly the slot width and length. The UEDGE fluid plasma model calculates the plasma conditions in the scrape-off-layer after comparison of data at the plasma midplane and divertor plate. The DEGAS Monte-Carlo neutrals model then calculates the neutral distribution on this (fixed) plasma solution. These results have shown that a 23 cm slot (X-point to outside target distance) with a width of about 1.5 cm (midplane flux line) should effectively control the neutral flow to the plasma core. DEGAS indicates that the core ionization should be reduced by a

Fig. 6.3-1. The DIII-D radiative divertor is a flexible slot design.

factor of about seven compared to an open divertor. Furthermore, the core ionization rate is reduced to values less than the beam fueling rate. We are currently developing impurity transport models, both in UEDGE and a Monte-Carlo approach called MCI, to calculate the effect of the slots on impurities.

The radiative divertor will impact the overall operation of DIII-D in several ways: 1.) the cryopumps will provide particle control for high triangularity plasma operation; 2.) the divertor hardware will reduce the maximum DIII-D plasma volume, thereby decreasing the maximum plasma current; and 3.) the divertor slots will limit the range of plasma shapes that can be run in DIII-D. DIII-D experiments have shown that high triangularity and particle control are necessary ingredients for high performance discharges. As shown in Fig. 6.3-1, the radiative divertor is built around a high triangularity ($\delta=0.9$) plasma shape. In addition, the slots and cryopumps provide pumping of this shape, which is not possible in the current advanced divertor. The advanced divertor can only pump low triangularity shapes because the pump is located at the outboard side of the divertor. The radiative divertor moves the pump inward, and adds three new pumps.

The radiative divertor has also been designed to be very flexible. As shown in Fig. 6.3-2, the height and width of the slots can easily be varied. This is important both to optimize the design based on experimental results and to benchmark various configurations with computer models. Currently, the combined UEDGE and DEGAS models indicate that a slot width of 1.5 cm (flux line measured at the plasma midplane) is optimum for reducing 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

Fig. 6.3-2. Modular radiative divertor hardware provides flexibility.

wider, neutrals can leak around the plasma and enter the core at the midplane. The nominal slot width for the RDP design is 1.5 cm, and by adding thicker or thinner graphite tiles, we can easily vary the slot width by about 3.5 cm. We estimate that this can be accomplished in about 3 weeks of total down-time on DIII-D. As shown in Fig. 6.3-2, the length of the slot can be increased from 23 cm 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 would be done each year during the major DIII-D construction period.

The Radiative Divertor Project has been reviewed twice during the past year. In January, a significant part of the DIII-D advisory committee review was focused on the RDP. We were encouraged to keep the design flexible and address both divertor and advanced tokamak issues. A separate RDP review was held in June, where both physics and engineering concepts were presented. The committee reacted favorably to the design, and indicated that the RDP was responsive to both the divertor and advanced tokamak programs on DIII-D. A number of design issues were brought up by the committee and were addressed in writing. The successful completion of these reviews resulted in the approval of the project in October, 1994. Our current schedule calls for detailed design in 1995, followed by completion of installation in December 1996.

7. SUPPORT SERVICES

7.1. QUALITY ASSURANCE

Fusion Quality Assurance (QA) engineers, inspectors, and support personnel maintained a high level of activity during 1994. Significant projects supported were the Advanced Divertor Program (ADP), the Divertor Material Exposure System (DiMES), the DIII-D Graphite Tile Upgrade Project, the Fast Wave Current Drive (FWCD) antenna for the Ion Cyclotron RF Heating System (ICRH), the Electron Cyclotron Heating system (ECH), and the Neutral Beam Injector System (NBIS).

7.1.1. DESIGN SUPPORT

The QA group performed the following specific jobs:

1. Reverse-engineered a flange weldment for the ICH Group.
2. Measured the 75°R-1 port flange orientation for the Design Group.
3. Assisted engineering in devising a measurement technique to establish approximate radial and elevation locations of the C-Coil support trays.
4. Reverse-engineered two lithium beam ceramic breaks for the Diagnostics Group.
5. Provided as-built data for the Thompson laser experiment to the Design Group.
6. Assisted the C-Coil Group to prepare as-built design drawings by providing elevations of features adjacent to the C-Coil.
7. Reviewed and approved all DIII-D design drawings, specifications, procedures, and procurement requisitions. Participated in design reviews and chaired the Material Review Board (MRB).
8. Reverse-engineered two Faraday shield hole pattern mockups for the ICH Group.

7.1.2. INSPECTION SUPPORT

1. The majority of QA activities during the year involved receiving and source inspections of purchased and fabricated material, parts, subassemblies and assemblies. Inspection activity was particularly heavy for the DiMES project, the ADP, the Neutral Beam System repair and the Fast Wave Current Drive antenna project (FWCD).
2. The FWCD project required an extraordinary effort to monitor supplier conformance to the design requirements. GA hired a temporary employee in Maine to perform in-process and final source inspections of FWCD coaxial line components at R.F. Technologies. The units were also inspected upon receipt at GA, and the Impact-O-Graph impact recorders were analyzed to determine if abnormal

transit shocks occurred. Significant problems were encountered with R.F. Technologies' quality of workmanship.

3. QA performed hardness tests, ultrasonic NDE tests, chemical and physical tests of high strength bolts, characterized weld distortions, witnessed hydro and air pressure tests, assisted the Operations Group in measuring the location of DIII-D bumper limiters with respect to the inner wall tiles, performed borescope inspections, and verified machine setups prior to machining operations. QA also provided measurements and adjustments to various machine features, performed tubing straightness evaluations, monitored repairs and provided realignments of removed and replaced components, monitored modifications of penetrations in the neutron shield walls, performed dye penetrant evaluations on the ICRH tubing, performed bench and field accuracy tests on alignment scopes for the Thompson tower, characterized source grids of Neutral Beam Injector sources on the Coordinate Measuring Machine (CMM), performed load versus deflection measurements on spring washers for the FWCD antenna project, and numerous other varied inspection tasks.

7.1.3. TOOLING/LAYOUT/ALIGNMENT SUPPORT

The QA group provided the following support activities for layout verification in the vicinity of DIII-D:

1. Completed the quarterly column and footing subsidence tasks, and the semi-annual pit wall crack inspections. Performed a special inspection of all structural features following the January 14 Northridge earthquake.
2. Measured the attitude and relative elevations of many DIII-D components using optical instrumentation. Assisted DIII-D engineering in measuring various features of the NBI beamlines, and assisted the Design Group in laying out column support locations for the FWCD antenna transmission lines.
3. Positioned several optical tooling monuments on the Thompson tower to detect any movements caused by structural modifications necessary to accommodate the new pellet injector.
4. Performed many as-built measurements of port boxes to assure adequate clearances for new equipment, and verified location and orientation of equipment during and after welding.
5. Measured and documented beamline bellows "cuff gaps" periodically on NBI injectors, and assisted the DIII-D Physics Group with alignment and calibration of the lithium beam diagnostic.
6. Performed a series of measurements inside the DIII-D vacuum vessel to ascertain the radial location of FWCD antenna shields and bumper limiters. Also utilized optical tooling instruments and custom made targets to characterize the relationship of the DiMES changer flanges to the axis of the DiMES assembly.
7. Performed several comprehensive grid alignment and spacing characterizations of NBI common long pulse sources on the CMM.

7.1.4. QA SYSTEM IMPROVEMENTS

GA has developed and released a new Fusion QA manual complying with DOE Order 5700.6C. DIII-D Project personnel were major contributors to the manual and the set of implementing fusion procedures. The new quality program is being implemented in the DIII-D Program, the DOE Tokamak Physics Experiment (TPX) Plasma Facing Components (PFC) Project, and the Inertial Confinement Fusion (ICF) project.

7.1.5. OTHER SUPPORT

Additional support was provided by QA in the following areas:

1. Fusion QA personnel conducted hoisting and rigging equipment inspections, and monitored a large number of high consequence lifts.
2. QA staff also reviewed drawings, procedures, technical specifications, procurement requisitions, supplier disposition requests, material certifications, attended various design and operations meetings, expedited inspection efforts, tracked and processed incoming procurements.
3. Substantial progress has been made towards locating and tagging all Fusion measurement and test equipment with appropriate calibration system status labels, and updating the calibration recall list.

7.2. PLANNING AND CONTROL

The Planning and Control 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 FY94 included work on the 4 MW ICH and radiative divertor upgrades, installation of the ORNL pellet injector, C-coil installation, diagnostic upgrades, machine operations and maintenance, and a major six-month vent.

7.3. ENVIRONMENT SAFETY AND HEALTH

7.3.1. 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. The publication "DIII-D Safety Procedures for Facility and Equipment Operation" contains policies and procedures that must be adhered to at the DIII-D site. Fusion works closely with GA's Licensing, Safety and Nuclear Compliance organization in areas such as health physics, industrial hygiene, environmental permitting and industrial safety.

The Fusion Safety Committee provides 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 and technicians. 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 cryogenics or chemicals.

In addition to the Fusion Safety Committee's oversight of activities at DIII-D, two individuals are dedicated full-time to onsite "preventive" safety involvement. Their activities include writing and reviewing procedures, developing and conducting special training classes, conducting inspections and follow-up, interfacing and coordinating with GA's Licensing, Safety and Nuclear Compliance organization and providing continuous oversight to assure compliance with established safety policies, procedures, and regulations.

The DIII-D Emergency Response Team consists of individuals involved directly with maintenance and operation of the DIII-D equipment. They are trained in cardio-pulmonary 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 the company Emergency Medical Technicians (EMTs) arrive.

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 personnel and outside consultants.

A report is provided to the Fusion Safety Committee where corrective action assignments are made. DIII-D inspections include:

Inspection	Frequency
Site Inspection (GA Personnel)	Monthly
Electrical Consultant	Yearly
CAL/OSHA Consulting Service	Yearly
Insurance Carrier Inspection	Yearly (Multiple)
S.D. City Fire Department CEDMAT	Yearly
GA Safety Committee Hazards Survey	Yearly
DOE-SF Safety Review Inspection	Every 2 years

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

All new employees and collaborators must go through thorough and comprehensive safety indoctrinations by the Senior Fusion Safety Officer and 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. Subcontractors also receive a similar indoctrination.

7.3.2. FY94 SAFETY

7.3.2.1. Training

Training is all-important to the safety of both personnel and equipment. Due to the complexity of the DIII-D site and its potential hazards, numerous safety training classes are conducted. Subjects of the classes include, but are not limited to: confined space entry, back injury prevention, radiological safety, laser safety, hazard communication, cryogenic safety, crane and forklift operation, lockout/tagout, the national electrical code, machine shop tool usage and basic industrial safety requirements.

7.3.2.2. Inspections/Reviews

Seven internal safety inspections of the DIII-D site were conducted by representatives from GA Safety, Fusion Safety, Fusion Management and Fusion Facilities Engineering with a total of 282 discrepancies noted. At the close of FY94, there were zero items remaining on the discrepancy list.

The San Diego Fire Department's Combustible, Explosive and Dangerous Materials (CEDMAT) team inspected the DIII-D facility.

An outside electrical inspector was contracted to inspect the DIII-D facility for electrical discrepancies. There were 38 items that were found to need modification or improvement and at the close of FY94 all items had been corrected.

7.3.2.3. Other Activities

The Fusion Safety Committee met 24 times this fiscal year to discuss various safety issues. The Committee:

1. Reviewed and approved 24 Hazardous Work Authorizations (HWA) after appropriate recommendations and changes by the Safety Committee and select safety subcommittees and;
2. Reviewed four incidents that involved no injury, two minor first aid incidents, four accidents that required minor off-site medical treatment and two lost time back injuries.

A noise monitoring and hearing conservation survey was conducted at Building 34. Noise amplitude readings were also made in the noisy areas of the facility as requested on the last DOE inspection. The findings were well below the current CAL/OSHA and DOE limits.

The following DIII-D safety procedures were reviewed and updated.

1. DIII-D emergency response procedures
2. Cryogenic practices at the DIII-D facility
3. Power dispatcher procedure
4. Laser safety procedure

7.4. VISITOR AND PUBLIC INFORMATION PROGRAM

Tours of the DIII-D facilities are open to organizations and institutions interested in fusion development (colleges, schools, government agencies, manufacturers, and miscellaneous organizations). These tours are conducted on a noninterference basis and are arranged through the DIII-D tour coordinator whose responsibilities include security, arranging tour guides, and scheduling tours. During 1994 1,901 people toured DIII-D for a total of 12,952 during the last ten years.

DIII-D personnel have also taken an active role in supporting science education within the community. The partnership between the fusion group and Kearny High School included student and teacher internships, donation of surplus equipment, school visits and lectures by staff members and participation by students and teachers in GA sponsored projects. These activities, primarily undertaken by volunteers, have enriched the educational program at Kearny and in the community at large and resulted in the naming of the GA-Kearny High School partnership as the outstanding partnership in San Diego in 1994.

8. CONTRIBUTION TO ITER PHYSICS R&D

8.1. ITER PHYSICS SUPPORT

During the past year the DIII-D Program made numerous voluntary Physics R&D contributions to the ITER Engineering Design Activity. These include:

1. Additions to the H-mode confinement data base;
2. Benchmarking DIII-D against the ITER Rebut-Lalia-Watkins transport model;
3. Measurements of helium transport and exhaust;
4. Study of advanced tokamak scenarios with high triangularity and reverse shear current profiles;
5. Measurements and calculations of fast-wave and ECH current drive efficiency;
6. Measurements of divertor power, particle fluxes and dynamics in the scrape-off layer.

In addition to the above work, DIII-D also carried out two experiments of specific relevance to ITER, a plasma elongation study and an ITER demonstration discharge. In these experiments we worked closely with the Joint Central Team to develop the experimental mini-proposals. The DIII-D elongation study showed that higher elongation increased performance, however questions regarding cost and difficulty of vertical control resulted in the ITER configuration's not being changed. In the second type of ITER specific experiments carried out on DIII-D, ITER demonstration discharges were produced which provided information on transport scaling and showed values of H/q up to 0.58, slightly less than the ITER design value of 0.66.

The DIII-D team is providing a number of U.S. ITER experts: K. Burrell (Transport); J. DeBoo (Confinement Data Base); G. Porter-LLNL (Divertor Modeling); R. Stambaugh (Chairman-Divertor); and T. Taylor (Stability). In addition T. Simonen is a member of the Physics Committee.

As a member of the U.S. industrial team, GA is now beginning to contribute to U.S. Home Team tasks involving diagnostics, plasma equilibrium and control and RF current drive modeling. Although this activity is not funded through the DIII-D Program, it uses experienced DIII-D staff members.

In addition to the above DOE funded activities, GA invested internal R&D funds in support of ITER. These activities have included construction of a helium cooled divertor component, and development of a helium cooled blanket design, rf components, and plasma control algorithms.

8.2. ELECTRON CYCLOTRON HEATING AND CURRENT DRIVE FOR ITER

Electron cyclotron heating (ECH) and current drive (ECCD) are major components of General Atomics' strategy for current profile control in the DIII-D advanced tokamak scenarios. ECH and ECCD have technological and physics advantages for ITER, and advances in source development make them credible options for very high power applications in the near term. As a source of auxiliary heating to ignition, remotely generated EC power can be injected through simple, small penetrations into the torus. Auxiliary current drive can probably be applied locally, providing control of the $j(r)$ profile leading to improved performance.

A comprehensive modeling code, CQL3D, for calculating the interaction between EC waves and non-circular toroidal plasmas, has been developed as a collaboration between General Atomics and the National Energy Research Supercomputer Center. The rf power densities are sufficiently low with ECCD that the distribution function remains approximately Maxwellian. The CQL3D code incorporates effects of momentum conserving electron-electron collisions which contribute typically ~20% to an increased ECCD efficiency arising from the current carried by the bulk electrons due to collisions with the tail. The code is bounce averaged, 2D in momentum space and 1D in the plasma radial variable, and is fully relativistic. Noncircular plasma geometry and the finite beam width of the injected radiation are accounted for. The code has been benchmarked by comparison with neoclassical conductivity and fundamental EC absorption.

This CQL3D code has been exercised to predict ECCD efficiency for several combinations of parameters. Current drive efficiency up to values of $\eta=0.30$ (10^{20} A/m²-W) was obtained at a frequency of 220 GHz, with injection 40 deg from perpendicular to the toroidal magnetic field for a full magnetic field equilibrium with central electron temperature of 30 keV. Fig. 8.2-1 shows results from CQL3D of a scan of η versus injection angle, with

Fig. 8.2-1. Electron Cyclotron current drive figure of merit vs. launch angle 24 MA ITER target plasma with $T_{e0} = 30$ keV

frequency as a parameter. The parameter of these curves can alternatively be considered to be magnetic field. The frequency 170 GHz is being considered for ITER, and the higher current drive efficiencies can be achieved by reducing the toroidal field from 6.0 Tesla used in these calculations in proportion to 170 GHz divided by the frequencies indicated on the curves.

The functional dependence of efficiency shown in Fig. 8.2-1 principally results from an interplay among the electron velocity at which the EC absorption occurs, the radial location of absorption considering toroidal trapping effects, the tokamak aspect ratio through effects of parasitic second harmonic absorption, and plasma temperature and density profiles. Density has a minor parametric effect, up to the vicinity of the cutoff density, which is $3.6 \times 10^{14} \text{ cm}^3$ at 170 GHz. The injection angle at which ECCD efficiency is greatest depends principally on the minor radius of the current drive location. Thus at 170 GHz, the current drive location varies from the magnetic axis at injection angle 20 deg to 60% of the minor radius at injection angle 50 deg. The radial width of the profile of the injected current is 0.1–0.2 times the minor radius. This provides a means for some off-axis current profile control.

9. TPX PHYSICS SUPPORT

The activities of the DIII-D program in support of TPX physics development during FY94 fall into two categories. We are involved on a continuing basis in a number of tasks which are directly associated with the TPX project, and which have been undertaken at the request of the project. In addition, because of the close relationship between the advanced tokamak and radiative divertor development goals of the DIII-D program and the long range mission of TPX to provide steady-state advanced tokamak operation, many of the experiments undertaken on DIII-D during the past year are directly relevant to TPX and, in fact, are called for in the TPX Physics R&D Needs document.

Tasks initiated by the TPX program include the development of an operating plan for TPX. In this effort we worked with the TPX Program Advisory Committee (PAC), which established a Working Group on Experimental Goals and Milestones. This has been a joint effort involving the PAC and the TPX physics team to develop a detailed operations plan for the TPX tokamak. The operations plan will give a realistic and substantive program for approaching the TPX mission. It will also provide a credible basis for estimating the needs for manpower, operating time, and upgrade hardware needed to accomplish the TPX mission. We developed a detailed draft plan listing major goals and milestones, as well as secondary milestones. It covers the three principal TPX research areas of plasma control, advanced tokamak operating regimes, and steady state divertor. The draft experimental plan document is essentially complete. The milestone list has been reviewed extensively, by the PAC and other TPX physics staff, and suggested changes have been incorporated. The draft of the text of the plan has also been reviewed, and we expect to issue Version 1.0 early in CY95.

In other activities, work was started on development of alternative current drive scenarios (*e.g.*, ECCD) for TPX. The standard TPX equilibria were reproduced using EFIT. The reversed shear case will be used for off-axis current drive modeling. This work will be continued in FY95. We also discussed applying the expertise developed at DIII-D on tokamak plasma control, as well as on current profile measurement and control, to TPX.

The prioritized list of physics R&D needs prepared by the TPX physics group was carefully considered in developing the DIII-D 1994 program, a major portion of which was devoted to advanced tokamak studies, and to divertor/SOL work. The experimental planning exercise for operation of DIII-D in CY94 was initiated in November. Because of the commonality between the DIII-D long range plan and the overall physics goals of TPX, the 1994 DIII-D experimental plan addresses many of the TPX R&D issues. Thus far, there has been good coupling between the DIII-D planning effort and TPX physics.

Among the 1994 experiments carried out on DIII-D of direct relevance to TPX in the area of divertor and SOL work were studies of the need for extensive helium glow discharge cleaning when using an active pump, a series of experiments on the effects of various injected gases on the level and localization of radiation, initial studies of feedback control for long-pulse reduction of the divertor heat flux, and helium transport and exhaust studies. In the advanced tokamak area, some of the directly applicable experiments were experiments on plasma rotation, including efforts to modify the rotation profile with counter-injected neutral beams, and studies of stabilization of kink modes by plasma rotation past a resistive wall. Specific studies of advanced tokamak operating

modes included extension of the VH-mode operating regime to a range of plasma currents, and further work on beta limits in high internal inductance plasmas. Other relevant experiments were optimization of the C-coil to correct for error fields and allow lower density operation, and studies of Alfvén instabilities in the presence of energetic neutral beams. The DIII-D experimental results for the year are described in detail elsewhere in this report.

10. COLLABORATIVE EFFORTS

10.1. DIII-D COLLABORATION PROGRAMS OVERVIEW

Although the major part of the fusion research which is carried out on DIII-D is performed by General Atomics scientists under General Atomics management, in the last few years DIII-D has become a type of national fusion research users facility. The total number of national and international collaborators on DIII-D in FY94 was over 200 including both full and part time collaborators. Through our national and international collaboration programs, GA is playing a major role in the worldwide advancement of fusion research towards the goal of a fusion power reactor. Our collaboration with the DOE national laboratories has brought new recognition to our program and broadened its scope. Our international collaborations have helped the U.S. to maintain its role as a strong partner in the ITER project.

The recent increase in our collaboration with Russian fusion research institutes and scientists has helped maintain an active Russian fusion program as well as providing GA and the U.S. fusion program with an economical method to expand DIII-D relevant research.

The DIII-D collaborative programs have contributed to the results presented in the previous sections of this report and are discussed in more detail in the following.

10.2. LAWRENCE LIVERMORE NATIONAL LABORATORY

Lawrence Livermore National Laboratory (LLNL) personnel participated in both the edge physics and advanced tokamak areas on DIII-D in FY94. Approximately 16 physicists are involved in the collaboration; roughly half are located in San Diego, and the other half commute between LLNL and GA. We also have a two engineers and three technicians located in San Diego working on the collaboration.

LLNL work in the edge physics and divertor area has included both experiments and modeling. One focus of divertor activity has been radiative divertor experiments in support of the radiative divertor design. Previously, both D₂ and neon puffing experiments have been performed, and divertor heat flux reduction by factors of three to five has been observed. The goal of the experiments this year was to control effects on the core by feedback and active cryopumping. In D₂ puffing experiments, we were able to reduce the core density by using these techniques, while maintaining nearly the same heat flux reduction. In impurity puffing experiments, we also obtained heat flux reduction at lower densities. To better study the radiation in these discharges, we have improved the bolometer inversion techniques. Analysis using the new inversion showed that in the case of D₂ puffing, most of the radiation is from the divertor, and with impurity puffing, the radiation is divided roughly equally between the divertor and a mantle surrounding the core.

LLNL staff also participated in several experiments to characterize fundamental divertor quantities, including heat flux symmetry during double null operation and double null radiative divertor performance. We have continued to study the width of the scrape-off layer as a function of discharge parameters in L-, H-, and VH-mode discharges. A broader scrape-off layer seems to be coincident with ELMs.

We completed three diagnostic projects and initiated a large diagnostic project during this year: installation and operation of a divertor interferometer; installation and operation of the Edge MSE system (discussed in the Advanced Tokamak section); installation and operation of the bias-ring IR TV system; and construction of a divertor Thomson scattering system. The divertor interferometer is a 250 GHz system which measures the line-averaged density across each leg of the divertor. Due to hardware and signal limitations, we have focused on measurements of the outer divertor leg. The signal passes through the plasma twice; a mirror has been machined into the ring of the advanced divertor to reflect the microwave signal. The system works well for L-mode and some H-mode plasmas, but has difficulty tracking the large density changes during rapid ELMs in H-modes. We are now comparing these data with other density measurements and the UEDGE code.

The bias-ring IR TV is a joint project between LLNL and GA, which was installed and operated this year. This system uses a slightly tangential view to image the region under the bias ring (*i.e.*, the pump throat) so that the divertor heat flux can be measured when the strike point is in the pump throat (and hidden from view of the normal divertor IR TV). We are working on the cross calibration of the two systems using plasma data where the strike point is swept slowly from the normal IR TV view to the bias-ring IR TV view with the pump warm. We have also obtained some preliminary data during pumping experiments.

The divertor Thomson scattering system is an extension of the existing core Thomson scattering system and involves a close collaboration between GA and LLNL. GA is providing the core laser and LLNL is duplicating the existing detection system and engineering the modifications to the system. When this extension is complete, it will provide plasma temperature and density information along a 20 cm long vertical chord that extends up from the floor of the vessel. Eight data channels, each providing one cm spatial resolution can be placed at twelve locations along this vertical chord. The remaining four channels are available for other diagnostic uses. Thus far, all of the new beamline hardware has been installed and successfully tested for alignment stability during recent operations. Peripheral beamline components such as energy monitors, alignment cameras, and apertures are being installed. Baking and vacuum proof-testing of the novel 180 cm diameter window will take place in January. Numerous data acquisition software and control system modifications are in progress and on schedule for performance of pre-tests early in CY95. Full system checkout and completion is expected by the end of April, 1995.

Development of the UEDGE code has moved forward significantly in several areas over the past year (see also section 2.3). The UEDGE code now calculates on non-orthogonal grids so that realistic divertor plate configurations can be modeled. We have also developed a separate code called the RECLUSE code which permits generation of a DEGAS grid which extends the flux-surface based grid used in the UEDGE code to a realistic vacuum wall and divertor plate geometry. The extension of this UEDGE grid then permits analysis of the neutral particle flow using DEGAS. This combination has been important for Radiative Divertor and TPX divertor design.

We have recently added the capability of modeling impurity transport to the UEDGE code. The code can now treat impurities in any of three different ways. Previously, we assumed the impurity density was a fixed fraction of the background plasma density and calculated the impurity radiation from non-coronal tables; the radiated power was part of the power balance. We can now also treat the impurities in an "average ion" sense as a second species of ions. The impurity charge state is determined by the local density and temperature, and the impurity density is determined by solving the fluid equations which include the relevant forces on the impurity ions. Alternatively, we are able to treat each of the impurity charge states as a separate ion fluid. This multi-charge state capability is currently in the debug stage.

We have used the UEDGE code to analyze several DIII-D discharges. The results, as discussed earlier, indicate the anomalous perpendicular transport coefficients lie in the range $0.03 \leq D_{\perp} \leq 0.25 \text{ m}^2/\text{sec}$, and

$0.12 \leq \chi_e \leq 0.25 \text{ m}^2/\text{sec}$. The electron thermal diffusivity increases with SOL heating power for fixed plasma current. The perpendicular transport coefficients are approximately an order of magnitude larger for a low power L-mode discharge than expected for an equivalent power H-mode discharge.

LLNL involvement in the advanced tokamak effort on DIII-D has been centered on measurement of the current profile with the Motional Stark Effect (MSE) diagnostic. We hope ultimately to use feedback control of the current profile with the MSE and a non-inductive current drive source such as fast waves or electron cyclotron waves. Upgrades to the MSE system discussed earlier were completed in March 1994 and have been operational for the entire 1994 experimental campaign. In anticipation of feedback control of the q profile, the MSE signals have been coupled via analog fiber optic links to the DIII-D real-time control computer. The MSE signals are now routinely acquired in real-time and the development of algorithms for processing the data in real-time is ongoing.

During operations in 1994, MSE operated reliably for nearly every discharge with neutral beam injection. Although several important observations were made with the new MSE system over the past year, two highlights include the measurement of edge currents during H- and VH-mode and the measurement of reverse shear q profiles.

With the improved resolution at the edge, we now observe the development of a large current density with complex structure at the edge during H- and VH-mode. These edge currents are believed to play an important role in the termination of the VH-mode. It appears that, depending on the discharge conditions, the VH-mode can terminate with either q_0 returning to unity or with the development of edge currents and a steep edge pressure gradient. Stability analysis of these discharges with the new current profile measurements is underway and experiments for lengthening the VH-mode period are being planned for 1995.

Another important development has been the measurement of reverse shear q profiles on DIII-D as shown in Fig. 10.2-1. By applying early beam injection (5 MW at 300 ms) during the initial current ramp, profiles with $q_{\min}=1.5$ and q_0 as high as 10 have been obtained. Although profiles produced in this manner are transient, the reverse shear can last up to three seconds. The reversed shear profiles are critical for the Advanced Tokamak program on DIII-D because they allow access to the regime of second stability.

In addition to the MSE diagnostic upgrade, we conducted further experiments in stability and confinement of high β_p discharges, continued investigations of anomalous fast ion losses, and began time dependent modeling of tokamak discharges needed for current profile control. From analysis of 0.4 MA high β_p experiments (1993), we found that core transport coefficients χ_{eff} and D_e approached ion neoclassical values during the time of enhanced core confinement, similar to observations made for the PEP mode in JET, but without an inverted q profile. During 1994, higher current experiments at 0.6–0.7 MA also showed enhanced core confinement. Analysis of these discharges is underway.

The influence of Alfvén Eigenmode instabilities on fast ion confinement has been studied during 1994. These instabilities, which degrade the quality of high performance discharges, such as VH-mode, high normalized beta (β_N), and high beta poloidal (β_p), are also a concern for fusion generated alphas and high energy negative ion beams ($E \sim 1 \text{ MeV}$) which may be used for heating, momentum input and current drive in a fusion power plant. The connection between Alfvén Eigenmode activity and poor fast ion confinement is inferred from several observations: a) the appearance of these modes during periods of poor confinement; b) charge exchange data correlating central fast ion losses with bursts of Eigenmode activity; and c) saturation and/or collapse of the parallel stored energy during problematic periods. Improvements in fast ion confinement have been observed using negative I_p ramps and reduction of the injected neutral beam energy. These effects as well as the parametric dependencies of these modes will be explored in 1995.

Fig. 10.2-1. The MSE diagnostic has made it possible for the first time to study the effects of different $j(r)$ profiles. This example shows a highly inverted $q(r)$ profile inferred from the MSE data.

A modeling effort using the Corsica code in support of DIII-D advanced tokamak operations was also begun. Corsica is a time-dependent, 1-1/2 dimension free-boundary equilibrium (TEQ) and transport code that has been under development at LLNL. Using a full description of the DIII-D coil configuration, Corsica can provide modeling of time-dependent effects associated with profile control scenarios. It presently runs on Cray computers and workstations under the BASIS interface which provides flexibility in adding or changing physics models. We have begun to use Corsica in the modeling of time dependent effects associated with our earlier high β_p experiments. In results to date, we have bench-marked operation of the equilibrium solver TEQ (used in design applications for TPX and ITER), with that of the EFIT routine used to analyze equilibria on DIII-D and many other experiments. Corsica was used in a hybrid mode with experimentally measured density and temperature profiles to model the slow, time-dependent flux evolution of DIII-D high β_p experiments.

10.3. OAK RIDGE NATIONAL LABORATORY

The ORNL collaboration during the past year included work in the areas of the Advanced Divertor Program, boundary physics, FWCD, plasma shaping, and pellet injection. In addition to hardware installation and commissioning of the ORNL pellet injector and FWCD antennas, data analysis was a major focus of activity.

10.3.1. ADVANCED DIVERTOR AND BOUNDARY PHYSICS PROGRAM

Helium Transport Studies and CER Spectroscopy. Substantial benefits could be realized in plasma performance if next-generation tokamaks, such as the ITER, could be designed to operate in an enhanced

confinement regime, such as H-mode or VH-mode. One possible difficulty, however, is that the core plasma helium (He) particle confinement time is expected to be longer than that for L-mode. For burning fusion devices, continuous purging of the He ash is essential. Estimates show that newly created He ions must be removed from the system within 7 to 15 energy confinement times to maintain a continuous burn ($\tau_{\text{He}}^*/\tau_E \leq 7-15$). The ratio $\tau_{\text{He}}^*/\tau_E$ is a global figure of merit for a reactor, where $\tau_{\text{He}}^* = \tau_{\text{He}}/(1-R_{\text{He}})$, R_{He} is the global He recycling coefficient, τ_{He} the helium particle confinement time, and τ_E is the energy confinement time. Recent measurements on several tokamaks have demonstrated that He can be readily exhausted from the plasma core in L-mode with $\tau_{\text{He}}^*/\tau_E \approx 12$.

During FY94, DIII-D experiments have focused on investigating enhanced confinement regimes, such as H-mode and VH-mode. These DIII-D He transport and exhaust experiments give an encouraging preliminary view of the suitability of the H-mode for reactor plasmas. The core transport rate was studied as a function of neutral beam power (and/or ELM frequency) and a demonstration of He exhaust from an H-mode plasma in a diverted tokamak was performed. The exhaust rates of He from these ELMing H-mode plasmas on DIII-D appear to be within the acceptable range for a fusion reactor based on a measured value of $\tau_{\text{He}}^*/\tau_E \approx 8$.

To simulate the presence of He ash in DIII-D, concentrations of 10% He (relative to n_e) were puffed into the plasma during an otherwise steady-state phase of the discharge. A lower single-null divertor (double-null for VH-mode) configuration was used with a toroidal magnetic field of 2.1 T, a major radius of 1.67 m, and a plasma current of 1.6 MA. To investigate a variety of confinement regimes, the NB power was varied, starting from modest levels of 2 MW and 4 MW, to achieve L-mode and ELM-free H-mode, respectively. At higher NB powers, ELMing H-mode and VH-mode occur. The transport of He is determined from the temporal evolution of the absolute He density profile in the plasma core measured using charge-exchange recombination (CER) spectroscopy (4686 Å transition), in conjunction with NB injection.

The He density profile measured by CER and the electron density profile measured by Thomson scattering exhibit the same shape. In Fig. 10.3-1, a comparison of the electron density and He density profiles measured in L-mode, ELM-free H-mode, ELMing H-mode, and VH-mode is displayed. The profiles in each case were taken ~1.0 s after the He gas puff, allowing the He enough time to come to equilibrium in the discharge. This observation suggests a strong similarity between deuterium and He transport and an encouraging absence of preferential accumulation of He in the plasma core.

To compare the He diffusivity and thermal diffusivity, a local power balance analysis was used to determine a single-fluid heat diffusivity χ_{eff} from the ONETWO transport code. The He diffusivity, D_{He} was determined from the time evolution of the measured He radial profile. A comparison of D_{He} and χ_{eff} for L-mode and ELM-free H-mode shows that $D_{\text{He}}/\chi_{\text{eff}} \sim 1$ over most of the plasma radius, consistent with previous measurements on TFTR.

As the injected power is systematically increased, edge localized modes (ELMs) become prominent with the ELM frequency increasing at higher NB powers. These ELMs expel both plasma energy and particles from the plasma edge. In terms of He transport properties within the core, both D_{He} and χ_{eff} increase as the NB power is increased. For ELMing H-mode, the quantity $D_{\text{He}}/\chi_{\text{eff}} \sim 1$ for $\rho < 0.5$ and appears to be insensitive to changes in input power. For $\rho > 0.5$ $D_{\text{He}}/\chi_{\text{eff}}$ increases with increasing NB power indicating that He is more quickly transported out of the plasma, probably due to the ELMs. This suggests that a favorable trend exists for enhancing He transport relative to energy transport as the ELM frequency is increased and makes an ITER scenario based on ELMing H-mode conditions more feasible.

Fig. 10.3-1. Comparison of measured helium density and electron density profiles in: (a) L-mode, (b) ELMing H-mode, (c) ELM-free H-mode, and (d) VH-mode. The solid curve is a spline fit to the measured electron density profile

Boundary Physics and Particle Control. Boundary Physics analysis efforts during FY94 have been focused on several areas: (1) a study of particle exhaust capabilities with the cryopump; (2) a study of the dependence of divertor parameters on the line-average density; and (3) a study of global particle confinement time for a variety of confinement regimes.

Over the past year considerable effort has gone into modeling the strong dependence of the baffle neutral pressure as a function of the separatrix-ring gap location. Reasonable agreement was achieved with the data for the pumped case, but more work is needed to understand the unpumped case. Additional work in this area will continue in FY95.

To study the dependence of divertor parameters on the line-average density, a density scan in ELMing H-mode was completed. The inner strike point peak heat flux was found to increase almost linearly as the density was reduced, but the peak particle flux reached a maximum for the intermediate density cases, thereby suggesting that partial detachment of the plasma might be occurring at high density.

The global particle confinement time, τ_p , has also been investigated during FY94 for a variety of confinement conditions to determine its scaling. The ratio of τ_p/τ_E in various confinement regimes is the primary result of this study. To date, this ratio has been computed for both L-mode and ELM-free H-mode discharges. For L-mode, this ratio is found to be approximately two, and for ELM-free H-mode, the ratio is approximately four.

10.3.2. ORNL FAST WAVE CURRENT DRIVE PROGRAM

Two new long-pulse fast wave current drive antennas were completed and installed on DIII-D in April, 1994. These antennas were designed to handle up to 4 MW of ICRF power in the frequency range from 60 to 120 MHz for pulse lengths of up to 10 s. Figure 10.3-2 shows a photograph of one of the new antennas. The Faraday shields of these antennas are coated with 100 μ of plasma-sprayed boron carbide to reduce the possibility of impurity influx from the antenna. All elements of the antenna, except for the Faraday shield, are water-cooled for long-pulse operation. The antenna arrays can be phased symmetrically or with $\pm 90^\circ$ phasing between adjacent elements. The antennas are fed from a single 2 MW transmitter through a matching and tuning network incorporating a decoupler stub to maximize power handling and simplify the impedance matching.

Fig. 10.3-2. Long-pulse fast wave current drive antenna for DIII-D.

The antenna in the 0° port was first operated in July, 1994. This antenna has performed as expected up to 1.1 MW, the maximum power so far available from the new transmitter. All operation to date has been conducted using current drive phasing for the antenna elements.

The existing antenna mounted in the 285° and 300° ports has been operated in two configurations: without a Faraday shield and with a bare inconel single-tier, tilted Faraday shield similar to that on the new long-pulse antennas. Heating efficiency was observed to be greater with a Faraday shield and power handling was also improved. Boronization of the DIII-D vessel appears to be effective in eliminating impurity generation from the exposed inconel on the Faraday shield.

Two diagnostics of particular importance to the FWCD program were installed and operated, an amplitude modulated (AM) reflectometer and an electron cyclotron emission (ECE) radiometer. The new AM differential-phase reflectometer on the 285° – 300° antenna was used during these experiments to obtain detailed measurements of the edge electron density profile, data that are critical to the understanding of the antenna-plasma coupling. The

reflectometer data show a modification of the edge density profile in front of the antenna for high-power operation. The density profile is pushed away from the antenna and steepened with the application of high RF power. This reflectometer operates over the 33–50 GHz frequency range using the extraordinary mode with 100 MHz modulation of the microwave probing signal. By measuring the phase delay of the resulting modulation envelope, a direct measure of the edge electron density gradient in this highly turbulent region of the plasma is made.

The ECE radiometer has been installed to provide fast response time measurements of the electron temperature profile, capability that will be used to infer the electron power deposition profile and electron heat diffusivity during FWCD and direct electron heating ICRF experiments. This instrument is a 32-channel, solid-state, heterodyne, filter-bank, millimeter-wave radiometer which replaces the GRECE grating polychromator system and uses the antenna, radiation transport path and calibration hardware designed for that system. The radiometer measures second harmonic ECE in the frequency range of 83–114 GHz, corresponding to the spatial range of $R = 1.67\text{--}2.29\text{ m}$ at $B_T = 2.0\text{ T}$. Under typical operating conditions, most of the outboard side of the electron temperature profile is covered with a radial resolution of about 2 cm and a response time of about 200 microseconds (7 kHz video bandwidth), which is adequate for electron thermal diffusivity and power deposition profile measurements. Video amplifiers with 100 kHz bandwidth have been built and will be installed during CY95.

10.3.3. PELLET INJECTOR PROGRAM

The pellet injection system was shipped to DIII-D in January 1994 and installation was begun in February. The injector was installed on a stand and connected to vacuum pumps and the DIII-D LHe delivery system. The injector is installed at the 135R0 port and injects pellets horizontally just above the midplane of the machine. The injector is configured with three barrels: 1.8 mm diameter with a 10 Hz repetition rate, 2.7 mm with a 5 Hz repetition rate, and 4 mm with a 2.5 Hz repetition rate. Figure 10.3-3 shows the injection system installation on DIII-D. Commissioning of the injector began in May and first pellets were injected into DIII-D in July. Both the 1.8 mm and 2.7 mm barrels are fully commissioned and produced reliable pellets at their specified delivery rate. The 4 mm barrel still remains to be aimed and tested. The pellet diagnostics include two light gates for each barrel, a common microwave cavity for mass measurement, and a common photodiode for viewing the ablation light emission. The diagnostics have been fully tested and have produced useful data for the initial pellet experiments. Completion of the commissioning of the DIII-D pellet injector and its control system is scheduled for FY95.

Initial testing of the pellet injector on DIII-D consisted of injecting small numbers of pellets into the end of DIII-D discharges. From these experiments it has been possible to obtain basic pellet penetration and deposition information. The plasma diagnostics on DIII-D are able to measure the pellet density and temperature perturbation with good spatial and temporal resolution. In particular, the Thomson scattering diagnostic can measure the density deposition profile from the pellet with more precision than has been previously available on any fusion experiment. Density profiles measured by Thomson scattering have been obtained before and after pellet injection. Typically, the 1.8 mm pellets penetrate to $\sim 40\%$ of the plasma minor radius. Experiments are planned for early FY95 to use the pellet perturbation to study particle transport in H-mode plasmas.

10.4. UNIVERSITY OF CALIFORNIA, LOS ANGELES

UCLA contributes to the overall DIII-D program in two main areas. The first is physics studies, especially the application of turbulence measurements using FIR scattering and reflectometry for study of the L–H transition and H– and VH-mode phenomena. The second is diagnostic development, involving the development, demonstration and integration of advanced diagnostics systems onto a mainline fusion device, in this case DIII-D.

Fig. 10.3-3. Photograph of pellet injection system as installed on DIII-D.

The current focus of activity in this area is on reflectometer development, including density profile and internal rf wave measurements, and the application of reflectometer systems to the challenging divertor environment. Both UCLA research programs are fully integrated within the overall DIII-D program, and UCLA staff participate in the DIII-D program at all levels. A separate research group within UCLA also operated, in collaboration with Sandia National Laboratories, two reciprocating Langmuir probe arrays on DIII-D. Effective November 1, 1994, this research project and associated staff transferred to UC San Diego. A summary of this group's activities can be found in Section 10.5 below.

At present, UCLA has responsibility for all the mm-wave and FIR turbulence diagnostic systems on DIII-D. Specific systems installed on the machine are inboard and outboard (high/low field side) fixed frequency reflectometers, a frequency tunable correlation reflectometer, profile measurement reflectometers at two toroidal locations, a new reflectometer system specifically designed to provide localized internal measurements of the amplitude of the rf waves used for heating and current drive, and a multichannel heterodyne FIR scattering system. The design of a new reflectometer system for the divertor is also actively under way. These systems have been used to support a wide range of experiments on DIII-D, and a selection of important developments during the year is outlined below.

10.4.1. DENSITY PROFILE MEASUREMENTS

Significant improvements have been achieved in the accuracy and reliability of broadband frequency-swept (FM) reflectometer measurements on DIII-D. This has been accomplished through several improvements, the most important of which has been the application of digital complex demodulation (CDM) analysis software, which can extract signal phase with subfringe accuracy. Using this new analysis technique, results from the broadband FM

system look very promising. Reflectometer measurements show excellent agreement with Thomson profiles in Ohmic and H-mode plasmas, and good agreement with Thomson data in L-mode. An example showing a typical comparison of measured reflectometer and Thomson density profiles in a low density discharge with auxiliary rf power is shown in Fig. 10.4-1.

Fig. 10.4-1. Example of typical density profiles measured via reflectometry and Thomson scattering in low density RF discharges. Note that the reflectometer system defines the central density under these conditions, and also provides superior edge resolution.

Obtaining high resolution edge density profiles in support of the rf program has been a particular focus of this work, as the reflectometer offers superior edge density profile resolution compared with Thomson scattering. Accurate edge profile measurements are very important for the rf program because the efficiency of the rf wave/plasma coupling process depends on the details of the edge density profile. Figure 10.4-2 shows an example of measured edge density profiles as a function of rf power for a fast wave launcher operating without a Faraday shield. These profiles were used by ORNL to calculate the theoretically expected antenna loading resistance, which was then compared to the experimentally measured value.

Agreement was also reached among UCLA, ORNL and GA to bring an ORNL amplitude modulated (AM) reflectometer to DIII-D from the PBX tokamak. This system is now operational on DIII-D and UCLA has provided support in the installation and operation of the system. The ORNL system is being used as part of an ongoing UCLA program to test and compare different reflectometer profile measurement techniques, including FM, AM and pulsed radar techniques and is also used in combination with the UCLA reflectometer for density profile measurements in front of the FWCD antenna.

Fig. 10.4-2. Example of measured edge density profile evolution as a function of fast wave power.

10.4.2. INTERNAL FW MEASUREMENTS

Current drive and rf heating play an important role in current tokamak experiments, such as on DIII-D, and are also planned for future devices such as ITER and TPX. However, there are still many unresolved questions regarding the details of fast wave rf absorption and propagation. Reflectometry can help to address these questions by providing spatially localized *internal* measurements of the rf wave amplitude, since reflectometry measures density fluctuations induced in the plasma at the rf wave frequency, and the density fluctuation amplitude depends on the rf wave amplitude. A development program to demonstrate internal rf wave measurements using reflectometry was initiated on DIII-D in FY92-93, and expanded during FY94. In addition to the original, fixed frequency reflectometer installed last year, a new frequency tunable system has been added. An example of the system's ability simultaneously to detect fast wave signals at different frequencies is shown in Fig. 10.4-3.

Fig. 10.4-3. Example of signals from fast wave monitor reflectometer system, showing the ability to detect different frequencies simultaneously.

10.4.3. INSIDE LAUNCH REFLECTOMETER SYSTEM

Inside launch (high field side) reflectometry is necessary for routine core access on DIII-D and is a very attractive option for ITER and TPX. On DIII-D, core access using the normal X-mode right hand cutoff is restricted to low density plasmas by absorption at cyclotron harmonics, while O-mode polarization is incompatible with flat H-mode density profiles. Routine core access can, however, be obtained using the left hand cutoff and inside launch. Another attraction is that, for high temperature reactor relevant plasmas, relativistic corrections for the left hand cutoff (inside launched) are relatively small, whereas these substantially affect the location of the normal right hand cutoff. A development program to investigate the use of the left hand cutoff and inside launch reflectometry commenced during the year with the installation of an upgraded inside launch reflectometer system. The system allows for a continuously variable input polarization, and thus provides access to all plasma cutoffs. Measurements commenced with the investigation of plasma asymmetries using O-mode propagation. This was done to confirm system operation by comparing with previous DIII-D measurements of an inboard/outboard asymmetry in turbulence response at the L-H transition. Measurements have since commenced using the left hand cutoff for core access; initial results are promising.

10.4.4. DIVERTOR REFLECTOMETER

Historically the divertor has tended to be an under-diagnosed region of the plasma. With the increased demands being made on divertor performance and the importance of the divertor region to overall tokamak operation, it is essential to increase diagnostic coverage in order directly to quantify divertor performance, to provide divertor plasma characterization, and to provide vital input for divertor modeling efforts. In order to yield improved density information, a new UCLA reflectometer system will be installed in the divertor during a next major machine vent early in CY95. The system was designed during the year and consists of a framework supporting two circular waveguides which will be installed in the 285 degree V-1 port. This location provides access to the lower X-point as well as the inside and outside lower divertor legs. The framework and waveguide are built as one piece and can be installed and removed during a clean vent, allowing for easy modification. The system is designed to provide data on both density fluctuations and density profiles and will add significantly to divertor diagnostic capability. A possible upgrade to the system will use electron cyclotron absorption techniques to determine the electron temperature along the divertor leg. These systems will be transferred to the radiative divertor when it is installed in 1996. Given the challenging nature of the divertor, the complex geometry, *etc.*, this is a highly developmental program.

10.4.5. EDGE AND CORE TURBULENCE

Measurements of edge turbulence during the year using far-infrared (FIR) scattering and reflectometry have included several features relevant to a more detailed theoretical understanding of the L-H transition. The measured turbulence spectra are Doppler shifted and broadened by plasma $\mathbf{E} \times \mathbf{B}$ drifts, allowing the relative timing between changes in electric field and turbulence magnitude to be determined. In all discharges, a significant Doppler shift (increased electric field) is observed prior to the observed turbulence reduction associated with the L-H transition, as if a threshold value of shear is necessary to reduce turbulence.

Core turbulence is observed to decrease on the same time scale and in the same spatial region as the observed decrease in energy transport during the enhanced confinement regime known as VH-mode. Calculations to explicitly determine the spatial dependence of the turbulence across the plasma radius indicate that the fluctuations are reduced in the plasma core ($0.5 < r/a < 0.9$), as the confinement increases during the H- to VH-mode transition. Simulations of the scattered spectra based upon a generic gradient driven mode and including shear suppression effects from published theories reproduce many of the observed characteristics of the scattered spectra. The turbulence magnitude appears to peak inside the separatrix and fall off rapidly toward the discharge center.

10.5. UNIVERSITY OF CALIFORNIA, SAN DIEGO

The University of California, San Diego program is a continuation of work begun in 1989 at University of California, Los Angeles. The research project has been transferred to the Fusion Energy Research Center in the School of Engineering at the University of California, San Diego effective November 1, 1994. Principal research topics from the past year include:

- the formation and impurity screening properties of "enhanced" scrape-off layers in double null divertor discharges and,
- quantitative tests of various L to H transition models based upon shear suppression,

and are summarized briefly below.

University of California, San Diego, in collaboration with Sandia National Laboratories, is responsible for two fast reciprocating Langmuir probe arrays on the DIII-D tokamak. The first, located on the outboard midplane, has been operational since November, 1990 and has made several key contributions to the DIII-D program since that time. The second probe, called the "X-point probe", will be installed into a V-1 port in the lower divertor chamber of the DIII-D tokamak in February-March, 1995. The X-point probe will measure the plasma density n_e , electron temperature T_e , plasma potential ϕ_p , density and potential fluctuations $\tilde{n}$ and $\tilde{\phi}$ and associated transport, and main ion flows across the outer divertor leg and in the vicinity of the lower X-point. These data are crucial for understanding the high recycling and radiation enhanced divertor regimes.

The fast reciprocating Langmuir probe arrays each provide unique capabilities. These systems have high spatial (2 mm) and temporal (0.4 μ s, limited by digitizing rates) resolution. The probe arrays simultaneously measure n_e , T_e , ϕ_p and the electrostatic turbulence in the plasma edge. Because the density and potential fluctuation measurements are correlated, the probes permit the evaluation of the radial particle and heat transport due to electrostatic turbulence. In addition, the probe arrays possess the unique capability to measure both parallel and $E \times B$ main ion flows via Mach probe techniques and to determine the phase velocity of the electrostatic turbulence.

10.5.1. ENHANCED SCRAPE-OFF LAYERS IN DOUBLE NULL DIVERTOR DISCHARGES

The existence of "enhanced" scrape-off layers (SOL) in double null divertor discharges has previously been reported for some VH-mode discharges in DIII-D. These SOLs were enhanced in the sense that the impurity screening appeared to be higher than in similar ELMing H-modes. Measurements with the midplane probe have verified that two types of enhanced SOLs exist: extremely broad SOLs in double null L-mode discharges; and SOLs in ELM-free H-mode and VH-mode discharges which appear to have higher levels of impurity screening.

In the past year, a series of experiments dedicated to investigating the formation of both types of enhanced SOLs was completed. Broad SOL profiles have been seen only in L-modes in double null divertor discharges. VH-modes do not possess broader SOLs than ELMing H-modes, contrary to the general interpretation of measurements of SOL profiles in ELM-free H-modes which were essentially identical to the VH-mode profiles. There are some indications that the screening of trace levels of neon puffed into the SOL in H-modes with long ELM-free periods may be enhanced over ELMing H-mode levels, similar to the VH-mode results reported earlier.

10.5.2. THE CONNECTION BETWEEN E_r SHEAR AND TURBULENCE SUPPRESSION

Substantial progress has been made in the past year in the comparison of experimental results on L to H transitions with predictions of various models by: 1.) studying L to H transitions in discharges just above the power threshold ($P_{inj} \approx 1.3 P_{th}$) where the temporal evolution of the edge radial electric field E_r is slowed; and 2.) utilizing the high spatial resolution of the probe array to compare the dependence of the suppression of turbulence in the E_r shear layer on E_r shear with predictions of models based upon shear suppression of turbulence.

Measurements in discharges with $P_{inj} \approx 1.3 P_{th}$ clearly show that E_r shear begins to increase before any substantial change to the main ion pressure gradient, indicating that the formation of the E_r shear layer at the time of the transition from L to H mode is due to the $V \times B$ rotation terms in the lowest order single ion species force balance equation. This result is inconsistent with models of the L to H transition based solely upon changes in the main ion pressure gradient. The E_r shear also increases before both the onset of the fast drop in D_α emission in the plasma periphery and any significant change in the fluctuation levels (Fig. 10.5-1). At the onset of the fast D_α drop, the fluctuation levels are suppressed within about 30–40 μ s, as shown in Fig. 10.5-2. This behavior suggests that a

threshold value of E_r shear exists for turbulence suppression, which is contrary to predictions of the non-self-consistent models of E_r shear suppression of turbulence (Biglari, Shaing, Zhang and Mahajan).

Fig. 10.5-1 Variation of (a) the radial electric field E_r 4 mm outside ($\times$) and 8 mm inside ($\bullet$) the separatrix, (b) the power in the fluctuations as measured by FIR scattering, and the D_α emission for a discharge with two L to H transitions (indicated by the vertical lines). The separation between the two E_r traces indicates the increase in E_r shear beginning several milliseconds prior to each transition. The fluctuation power does not change until the fast D_α drop.

Comparison of the spatial variation of the turbulence suppression with E_r shear shows that three regions exist in the plasma periphery (Fig. 10.5-3): 1.) a region where $E_r' < 0$ where the turbulence suppression scales as $|E_r'|^{-2/3}$ (high shear) and as $|E_r'|^{-1}$ (arbitrary shear), in reasonable agreement with the shear suppression models; 2.) a region of $E_r' < 0$ where the turbulence amplitude is proportional to the E_r shear, contrary to the E_r shear suppression hypothesis; and 3.) the near SOL where E_r' is small but a large degree of turbulence suppression is observed. It is possible to explain the departure of the scaling of the turbulence amplitude from $1/|E_r'|$ in the region of positive shear due to the effect of E_r curvature, E_r'' , for modes for which an E_r well ($E_r'' > 0$) is destabilizing, such as electron drift waves. The measured E_r'' is several times larger than that required to affect the stability of electron drift waves.

Fig. 10.5-2. Variation of the raw floating potential ϕ_f (a) and ion saturation current (b) and of the mean power in the ion saturation current fluctuations from a wavelet analysis (c) from 200 μ s before to 200 μ s after the start of the fast D_α drop at the L to H transition. All three traces show that the turbulence is suppressed within 20–40 ms of the start of the fast D_α drop.

These comparisons with L to H transition models show that the models are incomplete. Many models are single point, fluid models which do not properly treat the condition where both pressure gradient and E_r scale lengths approach the ion gyro radius in magnitude. A fully kinetic approach may be required to handle correctly the modifications to the ion orbits due to the large gradient in E_r . Further, existing models of the L to H transition have a single fluctuating field quantity, and radial transport that scales as the fluctuation amplitude to some power. Inside the E_r shear layer, near the point where $E_r' = 0$, the suppression of density fluctuations is minimal (20%) and potential fluctuations are enhanced (150%). In this region, the H-mode radial transport barrier is maintained by the phase angle between the density and potential fluctuations ($\alpha = \pi$). Such behavior cannot be treated within any existing models of the L to H transition.

Fig. 10.5-3 Profiles of the radial electric field, E_r , (black line; left axis) and the relative fluctuation amplitude $\Theta \equiv \left[(\tilde{n}/n)_H / (\tilde{n}/n)_L \right]^2$ (symbols; right axis) near the separatrix. The relative fluctuation amplitude is divided into three groups: E_r shear < 0 (●), E_r shear > 0 (open squares) and near the SOL E_r shear $= 0$ (filled triangles). Where E_r shear < 0 (●), the relative fluctuation amplitude agrees with the shear suppression models, while in the region where E_r shear > 0 (open squares), the fluctuation amplitude is proportional to the E_r shear, not inversely proportional, in contradiction to the shear suppression hypothesis.

10.6. INTERNATIONAL COOPERATION

The GA international collaboration program continues to provide a broad source of innovative ideas and opportunities which support the DIII-D research program. Major collaborations are continuing with JET in England, Tore Supra in France and T-10 in Moscow. In addition to the benefits gained from DIII-D staff assignments in these and other laboratories, foreign scientists visiting GA have made significant contributions to DIII-D program goals.

Contributions to the DIII-D programs were made by: S. Konoshima (one year assignment) from JT-60U working on the effect of impurities on H- and VH-modes and performing the function of JAERI representative to DIII-D; F. Weschenfelder (one year assignment) from TEXTOR working on DiMES; V. Trukhin from T-10 working with the X-ray spectrometer; a team from the T-10 ECH program; and a team from TRINITY lab. We have started a theory exchange with a group of Russian theoreticians to work on specific DIII-D related topics in H-mode and divertor physics and we have initiated a collaboration program with China which has brought four scientists from the PRC to GA for six month exchanges.

A summary of the progress made by DIII-D staff members in support of the international collaboration program is given below.

10.6.1. JET (ENGLAND)

JET is a large dee shaped tokamak, and, except for size and strength of magnetic field, is very similar to DIII-D. Our collaboration with JET remains the largest of our international collaborations. The main topics are divertor physics and technology and high performance. Two long term exchanges in the divertor area were completed this year. One involved participation in experiments with the recently installed Mark I divertor, and the other involved helping with the design of the Mark II divertor. GA also participated in JET high performance experiments. Since the installation of the new divertor, JET has not been able to achieve the same levels of performance as previously. The reasons for this are thought to be; a smaller plasma volume, less triangularity of the plasma, and higher recycling of particles from the walls (some of which are bare metal). Exchanges on plasma rotation and high power ICRF were also conducted.

10.6.2. ASDEX-U/W7 STELLARATOR (GERMANY)

The Steering Committee meeting for the DOE/ASDEX collaboration was held in March at MIT and ASDEX plans for the future were presented. It was agreed that the collaboration should concentrate on the following three areas: divertor physics; H-mode and energy confinement physics; and control and disruptions. After the Steering Committee meeting Dr. Kaufmann, the head of ASDEX Upgrade, visited GA for about two weeks to hold technical discussions on various areas of tokamak physics and also to discuss the planning of several future DIII-D/ASDEX exchanges. This year GA participated in ECH experiments on Wendelstein 7A and is presently involved in an ongoing exchange on the comparison of measurements of the edge plasma with numerical modeling.

10.6.3. TEXTOR (GERMANY)

Felix Weschenfelder from TEXTOR is at GA for one year to perform colorimetry measurements with the DiMES experiment. Colorimetry was developed at TEXTOR as a technique to observe the carbon erosion and redeposition on extended limiter areas. It measures the interference color of a transparent co-deposit of carbon, boron and hydrogen to determine the thickness of the layer. With the observation of a changing color pattern on a limiter, the erosion or deposition rate or even an erosion contour over an extended area can be measured. Using the colorimetry on DIII-D, the erosion rates on a precoated DiMES sample will be measured. A DiMES sample also will be coated during a boronization. After the boronization, the DiMES sample will be exposed during tokamak discharges and then, from measurements of the change in the color of the deposit on DiMES between discharges, erosion rates will be determined. More details on this topic can be found in the Introduction.

10.6.4. TORE SUPRA (FRANCE)

At the U.S./Tore Supra International Collaboration Steering Committee Meeting at Cadarache on November 15-16, the GA proposals for exchanges on pellet injection, study of impurities in poloidal and ergodic divertors and LHCD enhanced confinement were presented.

Four separate exchanges took place at Tore Supra. One was a month long exchange on pellet injection into tokamak plasmas. Two involved the measurement and control of the current profile in the tokamak. And one was a participation in experiments on pumped limiters, to compare the performance of the pumped limiter in Tore Supra with the pumped divertor in DIII-D.

10.6.5. JT60-U (JAPAN)

The DIII-D U.S./Japan Steering Committee exchange was held in Naka/Mito, Japan. Tokamak physics of interest to both sides was discussed with regard to the DIII-D/JAERI exchange program. JAERI expressed interest in two experiments which they recommended to be carried out on DIII-D, radiative cold pumped divertor in ELMy H-mode or VH-mode, and high beta operation with high bootstrap current. There was also some interest expressed in the possibility of operating DIII-D at 3-4 Tesla, which might allow DIII-D to reach break-even. On JT60-U GA participated in several experiments to vary the current density profile by the use of Lower Hybrid Current Drive. It was found that by varying $n_{||}$ it was possible to generate either broad or narrow current profiles. Some discussions were held concerning the possibility of using the GA designed combine ICRF antenna on JFT-2M. Although there is interest in this experiment by JFT-2M scientists, some modifications to the experimental hardware will have to be made.

10.6.6. GAMMA-10 (JAPAN)

GA scientists participated in a short exchange with Gamma-10 on experiments on the thermal dike divertor concept. This concept involves using radiation to reflect electrons from the divertor target plates back to the plates. The experiments during this exchange used a biased mesh to produce the same effect. These experiments yielded the first evidence that the central temperature was increased by the mesh biasing.

10.6.7. T-10 (RUSSIA)

General Atomics scientists participated in the fall 1993 and the spring 1994 campaigns on T-10 in Moscow. The experiments featured work on suppression of MHD activity by ECH and ECCD and work on high power ECCD at the second harmonic resonance. The experiments used four gyrotrons operating at 140 GHz and one tube at 160 GHz providing approximately 400 kW per tube for 300 msec pulses. A Michelson interferometer from GA was installed and operated during the T-10 campaign. The GA group participated in both the planning and execution of the T-10 experiments. The data acquisition system on T-10, which is being upgraded with the help of equipment provided by GA, now connects most computers at the Kurchatov Institute Division of Plasma Physics and allows shared access to data and programs from nearly anywhere in the division. This capability was extensively exercised to follow the progress of the experiments and produce a series of calculations with the ASTRA code of the electric field behavior during sawteeth.

10.6.8. TRINITI (RUSSIA)

GA has been involved in an exchange program with the TRINITI lab at Troitsk for the past few years. The main topics are: materials for plasma facing components such as graphite with boron and silicon carbide coatings; plasma and divertor spectroscopy; and the use of the Russian developed DINA code for modeling dynamic plasma behavior.

11. FY94 PUBLICATIONS

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