GA–A23056

DIII–D THREE-YEAR PROGRAM PLAN 1999–2001

by
R.D. STAMBAUGH and RESEARCH STAFF OF THE DIII–D TEAM

SEPTEMBER 1999

DISCLAIMER

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ABSTRACT

This draft three year program plan presents a concise view of the research planned on the DIII–D tokamak in the years 1999–2001. Reference is made to GA–A22950, "The DIII–D Five-Year Program Plan," which is a comprehensive discussion of research planned for DIII–D. We have also incorporated in Section 4 a slightly revised version of the "DIII–D 1999 Research Synopsis" which is posted on the General Atomics Website. That synopsis and other information on the Website contain the specifics of the 1999 Experimental Plan. This draft plan will be evolved until about May 1999 by the DIII–D Research Council with inputs from the DIII–D Advisory Committee and others.

CONTENTS

CONTENTS (Continued)

LIST OF FIGURES

LIST OF TABLES

1. OVERALL PROGRAM GOALS

1.1. MISSION OF THE DIII–D NATIONAL FUSION PROGRAM

The overall mission statement of the DIII–D Program is "To establish the scientific basis for the optimization of the tokamak approach to fusion energy production."

The main output of the DIII–D Research Program is a scientific basis. "Scientific" means developing a solid understanding of the underlying physical principles and incorporating it into useful predictive modeling tools. "Optimization" means experimentally demonstrating performance parameters at the theoretically predicted limits for the tokamak magnetic confinement system and achieving to the greatest degree possible an integrated, steady-state demonstration of optimized performance that projects to an attractive fusion power system. The integrated optimization sought and the scientific basis established will allow the definition of optimal paths to fusion energy using the tokamak approach.

1.2. DIII–D NATIONAL PROGRAM RESEARCH GOALS

Working with the DIII–D Research Council, this mission has been elaborated in three additional research goal statements:

1. The DIII–D Program's primary focus is the Advanced Tokamak Thrust that seeks to find the ultimate potential of the tokamak as a magnetic confinement system.
2. Where it has unique capabilities, the DIII–D Program will undertake the resolution of key enabling issues for advancing various magnetic fusion concepts.
3. The DIII–D Program will advance the science of magnetic confinement on a broad front, utilizing its extensive facility and national team research capability.

The DIII–D National Research staff is highly motivated to pursue the Advanced Tokamak (AT) Thrust. Finding the ultimate potential of the tokamak as a confinement system is primarily a scientific motivation. The integration of AT elements into achievable single discharges requires programmatic compromise and tradeoffs evolved over a multi-year period. In order to provide more focus on critical issues in the DIII–D Program, the method of organization of the experimental research was changed in 1998.

The new scheme is a matrix type of approach in which one dimension of the matrix is a set of Thrusts. A Thrust is aimed at a key objective of the research and is given a significant block of run time in which to realize its objectives. The research thrusts and their leaders will change year-to-year to keep up with the evolution of the experimental program. The main motivation for this new scheme was the desire to gain a more purposeful and visible path to the eventual AT integrated plasma scenarios targeted in the Five-Year Plan. This new scheme also makes it natural to create cross-disciplinary teams to pursue integrated plasma scenarios. Most of the thrusts in the 1999 run plan relate to the AT goal of the DIII–D Program. The AT Program in its broad outlines is described in Section 2.2 of the Five-Year Plan. This work pursues Goal 1 above.

The second dimension of the experimental planning matrix is comprised of the four enduring topical areas of fusion energy science: stability, confinement and transport, divertor/edge physics, and heating and current drive. The DIII–D Facility and the DIII–D National Team is a resource of immense value to the U.S. Fusion Program in terms of advancing the science of magnetic confinement on a broad front. DIII–D has a superb diagnostic set, increasingly flexible and capable plasma control systems, an excellent research staff, and a comprehensive set of analysis codes and theory support that enable real learning in depth from the experiments done. The staff recognizes and embraces a responsibility to the greatest extent possible to use that resource to advance the state of fusion energy science knowledge generally.

The managers of these topical areas implement this second dimension of the matrix and have responsibility for the work supporting Goal 3 above. Their continuing leadership of these topical areas over a period of years assures the continued scientific focus of the DIII–D research. A thorough discussion of the scientific topics being pursued in the DIII–D Program can be found in Section 2.3 in the Five-Year Plan.

The DIII–D Research Staff also are strongly motivated to see magnetic confinement progress to future next-step devices. The AT work and the broader scientific work on DIII–D can contribute greatly to the definition and the support for these future machine initiatives. Some of those possible next step options are:

ITER which was based on conventional tokamak physics or the RTO-RC ITER which plans more exploitation of and/or reliance on AT physics.

An advanced performance superconducting tokamak (JT–60SU, ARIES-RS) which exploits AT physics toward steady-state.

A copper-coil ignition experiment about the size of JET and using gyroBohm scaling of H–mode, relying on more conventional tokamak physics.

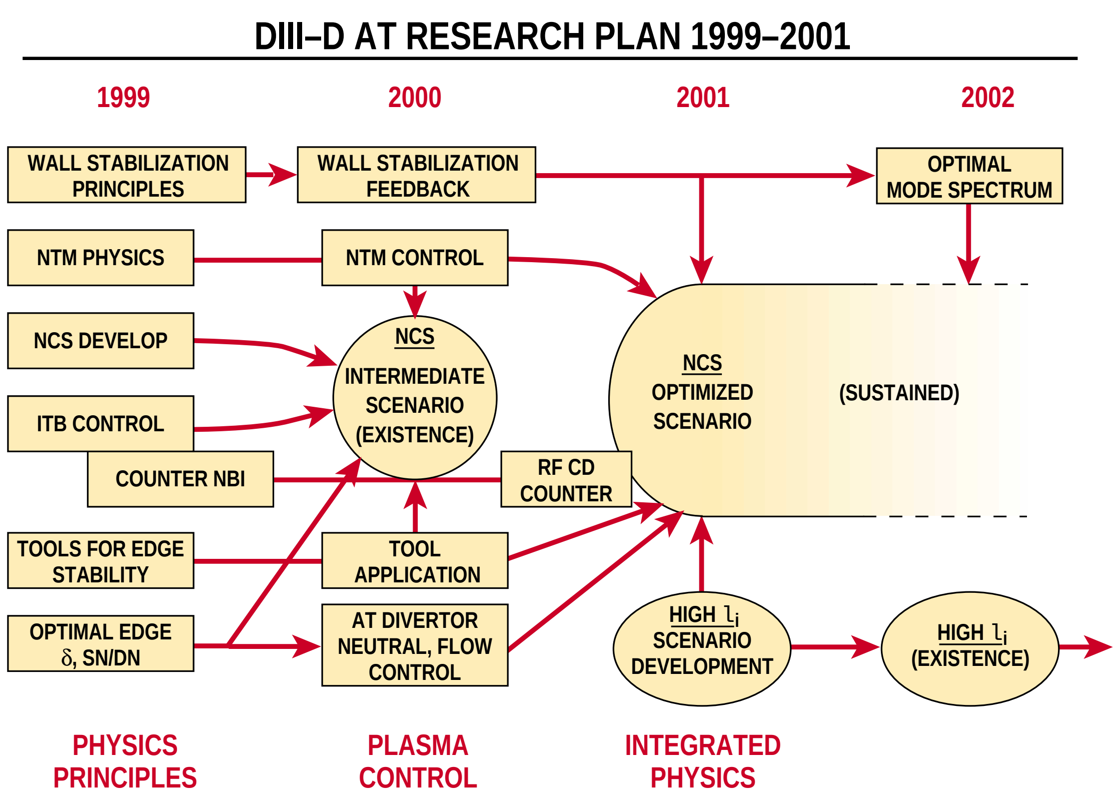
A compact, copper-coil ignition experiment (as exemplified by CIT/BPX/IGNITOR) but enabling studies of or relying on AT physics.

A next-step spherical torus which relies on most elements of AT physics to enable the study of burning plasma physics in long pulse or steady-state.

Research toward Goal 2 can appear either as thrusts or as elements of the Topical Science Area plans. A discussion of how DIII–D research relates to the various future machine possibilities can be found in Section 2.4 of the Five-Year Plan.

Competition for experimental time on DIII–D is intense. Priority goes to the Advanced Tokamak work, which occupies most of the thrusts. We seek to reserve about 30%–40% of the run time for the Topical Area Managers to allocate to more broadly motivated studies. The work to support Goal 2 has to find time either as a thrust or in the Topical Areas.

The centerpiece of the three-year plan is Fig. 13 which is reproduced below. This figure diagrams the interplay of machine capabilities and modification and other tools needed to address the principal negative central shear (NCS) and peaked current profile (high ℓ_i) scenarios for performance improvement. The following sections describe these scenarios, their preparatory modeling in DIII–D, and the plans for using the DIII–D Facility to address them.



2. THE ADVANCED TOKAMAK PROGRAM

2.1. PHYSICS ELEMENTS OF THE PRINCIPAL AT SCENARIOS

The goal of the DIII–D program is to establish the scientific basis for the optimization of the tokamak approach to fusion energy production. This scientific research has many elements, but the principal focus of the DIII–D program toward achieving this optimization is the advanced tokamak program. The advanced tokamak program is aimed at improvement of the tokamak concept towards higher performance and steady-state operation through internal profile modification and control, plasma shape, and MHD stabilization. The dependence of the core performance on the boundary conditions, and the operational regimes envisioned, put more stringent requirements on the divertor and edge plasma, leading to inclusion of divertor optimization and control in any tokamak optimization program.

Two characteristics make the optimization of tokamak performance “advanced”: the inherent one and two dimensional dependence of tokamak performance on the plasma profiles, shape, and boundary; and the requirement to develop solutions that are both multidimensional and self-consistent. The performance capabilities and limitations of the tokamak, and requirements for an energy producing tokamak have long been communicated in terms of global zero-dimensional parameters and largely empirical scaling relations. Chief among these scaling relations are the confinement scaling relations and the scaling of beta with normalized current, known as Troyon scaling. More recently we have discovered, both experimentally and theoretically, that the performance of the tokamak plasma also depends largely on the details of internal plasma profiles, details of the plasma shape, and details of the plasma boundary.

This improvement in our understanding depended critically on the development of new diagnostics to measure the important profile parameters, such as the motional Stark effect diagnostic for measuring the internal magnetic field structure, the charge exchange recombination system to measure toroidal and poloidal plasma flows, and many new turbulence measurements. These new measurements lead to discovery and appreciation of new and important physics phenomena in the tokamak, such as the role of sheared $\mathbf{E}\times\mathbf{B}$ flow, and neoclassical tearing modes.

Equally important to new diagnostic capability is the development of new theories and modeling capabilities to put the transport, stability, and current drive projections on a firmer physics basis. An excellent example of the modeling and theory progress is in gyro-kinetic and gyro-fluid approaches for physics based transport calculations, and the appreciation of the importance of sheared **E**×**B** flow in the predictions.

The self-consistency of the parameters and profiles of high performance plasmas is one of the leading challenges of the advanced tokamak program. As well as the details of both the current density and pressure profile impacting the ideal stability limit and stability to non-ideal modes, at high beta the self-generated bootstrap current is necessarily a major component of the total current. Since the profile of the bootstrap depends on not only the profile of the pressure but of its individual constituents (density, electron temperature, ion temperature, …), the pressure profile and the current density profile are not separable. But, the pressure profile is determined by the transport profiles. In turn, the details of the pressure profile and the current density impact the turbulence growth rates and sheared **E**×**B** flow which predominantly determine the transport. In a final advanced tokamak scenario, these interdependencies and complex non-linear relationships must be fully taken into account and fully integrated. This process greatly benefits from and contributes to the development of a strong fundamental (first principal) physics basis for fusion science.

The DIII–D Advanced Tokamak program aims to develop the best possible operational scenario for fusion energy production using the tokamak. There are many opportunities to make improvements, and many complex interdependencies that allow for a multitude of possible advanced tokamak solutions. In this context it is important to recognize that our rapidly developing understanding and new innovations can lead to scenarios that we do not now envision. So, in developing the “scientific basis for optimization of the tokamak’’ we consider of paramount importance to maintain an attitude of research that is open to new discoveries and continual improvements. We therefore try to plan a DIII–D program that is not only targeted toward testing specific scenarios, but is also optimally positioned to take advantage of new discoveries and innovations. This translates directly into developing diagnostic and control capabilities that are flexible and versatile.

To make significant progress in our research, it is important nevertheless to focus on testing specific scenarios while being alert for discovery. It is important to set aggressive and measurable goals (targets) toward which to focus our efforts. We take our best present understanding of the physics and our best vision of the future embodiment in an energy producing system and develop scenarios, which we can test experimentally in the DIII–D device.

Consideration of physics and energy production lead us naturally to two principal advanced tokamak scenarios. These two scenarios are negative central magnetic shear (NCS) and high internal inductance (high ℓ_i). These two scenarios do not encompass all the known approaches to tokamak improvement, but rather provide some focus to the challenges that confront us. The profiles and conditions of the two scenarios are quite different, but it is recognized that a fully optimized scenario might lie somewhere in the space between the two.

The viability of a tokamak as an economically and environmentally attractive power plant requires both sufficient energy confinement time, τ_E, for ignition margin, and sufficient volume average toroidal beta, $\beta_T = 2\mu_0 \langle p \rangle / B_T^2$, for adequate fusion power density. Further improvements in the tokamak reactor concept can be made if these improvements in β_T^{max} and τ_E are obtained in steady-state discharge conditions (Kikuchi 1993). We are seeking scenarios that have the potential for high beta, high confinement consistent with steady state, and consistent with divertor scenarios that can provide adequate heat removal, particle and helium ash control, and impurity control.

A minimum necessary condition for an attractive fusion energy producing system is high energy gain. Some insight into possible operational scenarios is obtained by considering the energy gain for a steady state system, given by Eq. (1):

$$Q \propto \frac{P_{fus}}{P_{CD}} = \frac{P_{fus}}{\frac{n\,R\,I_P}{\gamma_{cur}}\left(1 - f_{BS}\right)} \propto \frac{\gamma_{cur}\,\varepsilon_{eff}\,\beta_N^2}{n\,q\left(1-\xi\sqrt{A}\,q\,\beta_N\right)} \quad . \tag{1}$$

In Eq. (1), P_{FUS} is the fusion power, the P_{CD} is the current drive power, γ_{CUR} is the current drive efficiency, ε_{EFF} is the effective inverse aspect ratio, **A** is the aspect ratio, q is the safety factor at the plasma edge, β_N is the normalized beta and f_{BS} is the fraction of the total current that is the self-driven bootstrap current. In any steady state scenario, care must be taken to minimize the current drive power required. One can view two separate approaches (NCS or high-ℓ_i) to minimizing this current drive power: (1) maximize the bootstrap fraction, or (2) maximize the efficiency of current drive (γ_{CUR}/nI). If the bootstrap fraction becomes a major fraction of the total current, the current profile becomes naturally hollow with the maximum off-axis and the central portion of the plasma has negative central shear, NCS. If the emphasis is placed on increasing the current drive efficiency, it is natural to drive the current on axis where the temperature is highest (current drive efficiency is proportional to electron temperature) and where the effects of trapping are minimal. Axial current drive leads naturally to peaked current densities, with large positive magnetic shear in the outer plasma region. A schematic of the resultant current profiles is shown in Fig. 1. The actual current profile for these two cases depends on establishing consistency among the profiles, stability, and transport.

Unless the NCS scenario has fully 100% bootstrap driven current, it is important to maintain relative high current drive efficiency in both the NCS and high ℓ_i scenarios.

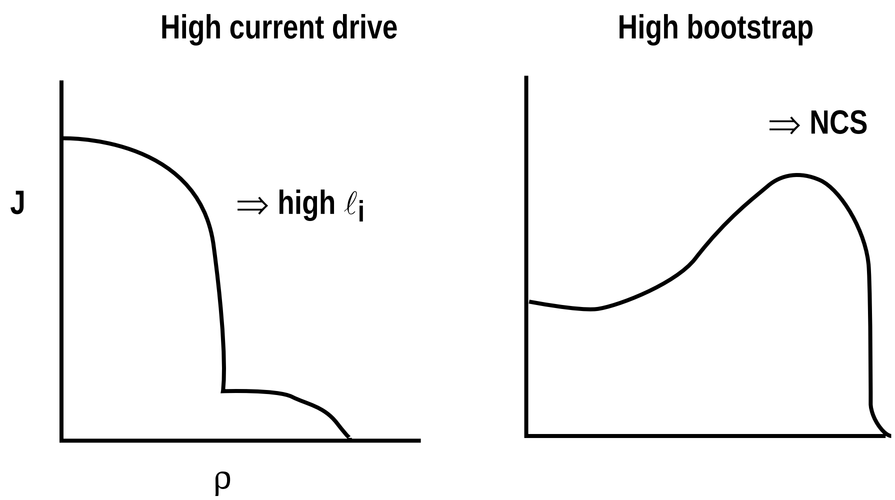


Fig. 1. Steady state considerations also lead to two "natural" current profiles.

The need for high current drive efficiency pushes steady state operational regimes to higher temperature and lower density than might otherwise be the optimal in a Ohmically pulsed scenario. The higher temperature and lower density, impose new challenges for heat removal and impurity control for the divertor. This lower density, higher temperature operation motivates the inclusion of divertor optimization as an important element in the DIII–D AT program.

Simple physics considerations also lead to the same operational scenarios, (1) NCS and (2) high l_i. We show in Fig. 2, the general dependence of ideal ballooning stability and ion temperature gradient (ITG) driven instabilities on the magnetic shear, $S_M = \rho/q\ (\partial q/\partial \rho)$. These general dependencies, known for a long time, clearly show that both low or negative magnetic shear and high magnetic shear are favorable for stability of ballooning modes and ITG modes. These physics considerations lead to the same two general classes of scenarios given above; (1) low or negative shear → NCS, and (2) high positive shear → high l_i. It is worth noting that the magnetic shear (in the large aspect ratio circular limit) observed experimentally in Ohmically driven discharges is near 1, nearly the most unfavorable value for ballooning and ITG mode stability. So one might expect that the ability to modify the current profile toward either larger positive or negative magnetic shear would lead to positive benefits.

2.1.1. General NCS Considerations

The NCS scenario has the potential for a high bootstrap fraction at moderate q: the bootstrap fraction can approach unity at q_{95} = 5–6. Furthermore, there is the potential for the bootstrap current to be well aligned with the total current, resulting in low, total current drive requirements. The hollow current profile, and the resultant region of negative central magnetic shear derive naturally from the bootstrap current. The bootstrap current is proportional to the square-root of the local aspect ratio times the pressure gradient, both which go to zero on axis, so that the bootstrap current profile is naturally hollow. In addition, the higher axial q and lower poloidal field in the core have the effect of increasing the total bootstrap fraction. The high bootstrap fraction results in lower total current drive, but highly localized, off-axis, precision current drive is needed. Electron cyclotron current drive is well suited for the precise off-axis current drive needed. Because of the potential of the NCS scenario with respect to fusion energy, we have chosen it as the leading scenario on which to focus.

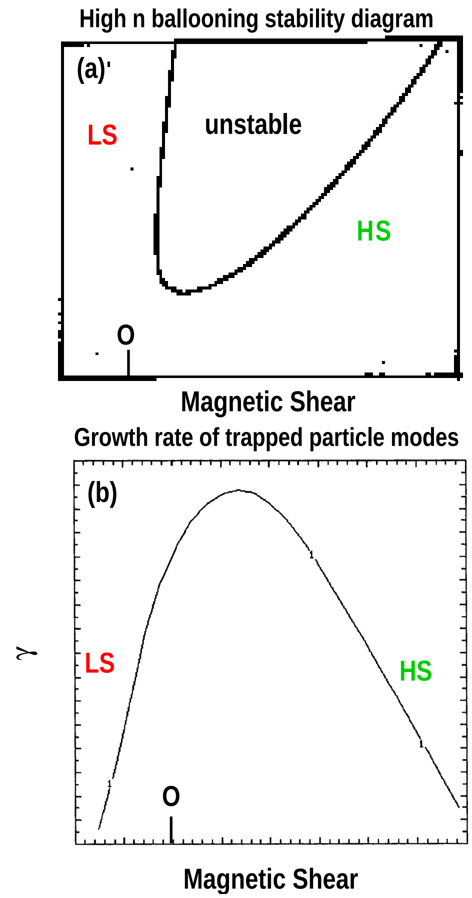


Fig. 2. Both low magnetic shear (LS) and high shear (HS) are favorable for: (a) higher beta, (b) reduced turbulence and reduced transport. Magnetic shear is $s \propto R/B_T^2 q^2 \, d\rho/dr$.

The NCS scenario does have some very specific challenges. The first challenge is stability. Stability to ballooning modes is a necessary condition for achieving high beta, and therefore an important consideration. The NCS scenario avoids ballooning mode limitations because the region of high pressure gradient is in the region of low or negative shear, where there is access to the second regime and no limiting pressure gradient is calculated. Furthermore, the negative shear region is stabilizing to neoclassical tearing modes where the pressure gradient is expected to be large, and if the minimum value of q is above 2, the absence of low order rational surfaces should further diminish the importance of the modes.

There are several MHD instabilities that remain a challenge to the NCS scenario. Strong pressure gradients in the region of negative shear can be destabilizing to resistive interchange modes. Modeling indicates that these modes are stable if the magnitude of the negative shear is kept modest. Double tearing modes are calculated to be unstable as a consequence of the double value of q. However, these are rarely observed in the experiment, and modeling indicates modest rotational difference in the plasma between the two surfaces (as observed in the experiment) is sufficient for stabilization.

The NCS scenarios have quite low l_i and are generally unstable to the external/global kink, in the absence of a conducting wall. However, the broad current density profile, broad pressure profile, and strongly shaped plasmas couple very strongly to a nearby wall. Modeling indicates that $\beta_N > 5$ stable to n=1 and 2, is easily obtained if a conducting wall is located at $r_W/a <$ ~1.5. The modeling calculations for n=1 are shown in Fig. 3. However, the real wall is resistive and the plasma is subject to the resistive wall mode. The stabilization of the resistive wall mode then is key part of validating and optimizing the NCS scenario. The DIII–D program is taking two approaches to stabilization of the resistive wall mode; passive stabilization with a rotating plasma in the presence of a resistive wall, and active feedback stabilization with non-axisymmetric external coils.

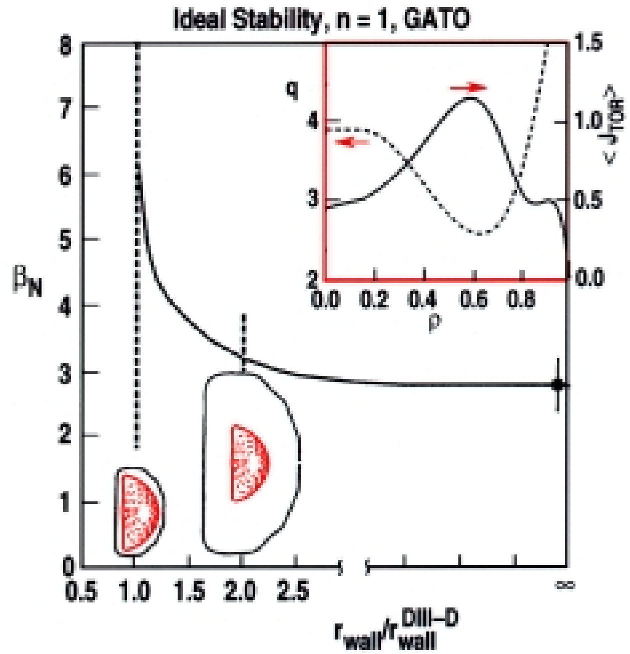


Fig. 3. Maximum stable beta increases for closer wall position: ideal n=1 stability using DIII–D plsma shape and DIII–D wall. Insets are typical current density and q profile (Taylor1995).

It is important to note that reasonably high beta values can be calculated for the NCS scenario without a conducting wall; β_N values <4 are calculated, very similar to the high l_i scenario. So if wall stabilization proves not to be so attractive in a reactor embodiment, there remain attractive NCS and high l_i scenarios.

For the NCS scenario, perhaps the most challenging physics lies in the consistency of the profiles. A range of current density and pressure profiles can be identified that are consistent with high beta stability. In particular, it can be shown that broad pressure profiles are required for high beta stability and alignment of the bootstrap current (Fig. 4). However, the combination of the q profiles, pressure profiles, and rotation (E×B) profiles often result in transport reduction and often the formation of a clear internal transport barrier that leads to pressure peaking that are not compatible with high beta.

Discharges in DIII–D can have an internal transport barrier, with no edge transport barrier (L–mode NCS), a strong edge transport barrier, (H–mode NCS), and a self-regulating edge barrier with an internal transport barrier (ELMing H–mode NCS). These three cases have different challenges with respect to high beta and self-consistent solutions. The L–mode NCS has a very weak pressure gradient in the outer portion of the plasma. From Fig. 4, it is clear that very peaked pressure profiles that result from an ITB at small radius will result in a low stability limit. The key challenge for the L–mode NCS is to move the transport barrier to larger major radius to achieve higher beta and bootstrap alignment. The ELMing H–mode NCS and the H–mode NCS both lead to broader pressure profiles and the potential for high beta with an ITB. For the ELMing case, the repetitive ELMs provide a seed for neoclassical tearing modes, and the higher pressure gradient in the positive shear region make the neoclassical tearing modes unstable. For H–mode NCS, a clear strong barrier exists near the boundary, and the plasma is subject to low n kinks associated with the high edge pressure gradient and high edge current density. High beta, broad profiles, and strong shaping cause strong harmonic coupling and these edge driven modes are no longer localized to the edge. The key challenge for the H–mode NCS scenario is to understand the how to moderate the edge and avoid the edge instability or its strong coupling to the core.

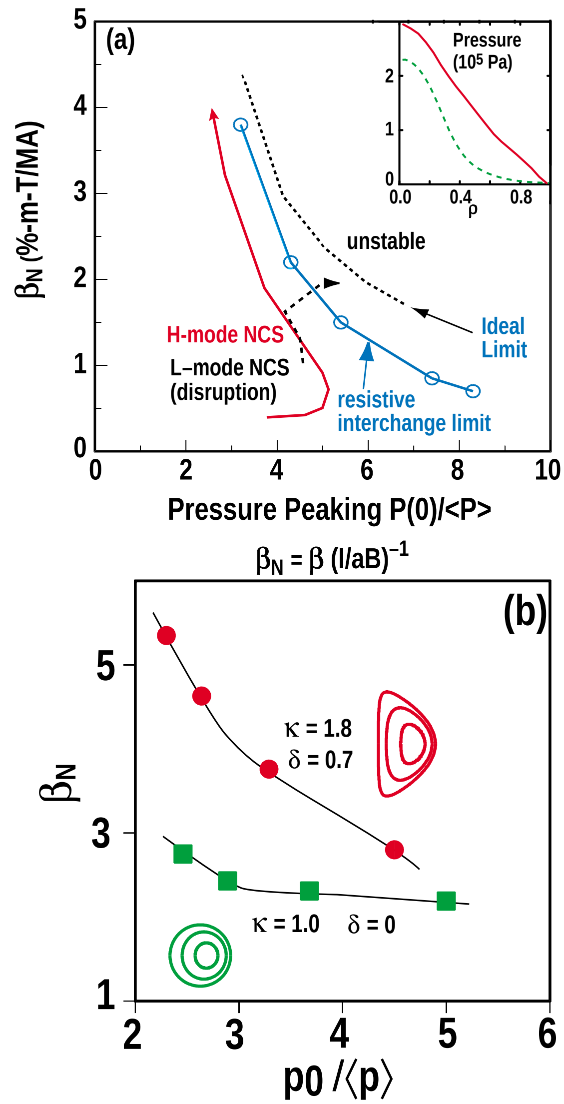


Fig. 4. Higher β is obtained with broad pressure profile (a) normalized beta vs. pressure peaking. Ideal and resistive limits are from generated equilibria similar to the experimental. Dashed trajectory is for an L–mode NCS discharge, solid trajectory is for H–mode NCS discharge (Lao 1996). Insets are exp. pressure profile just prior to disruption.(b) β_N vs. pressure peaking for D shaped and circular shaped equilibria, q_0 = 3.9, q_{min} = 2.1, q_{95} = 5.1, rw/a = 1.5 (Turnbull 1996).

A sound physics understanding of the reduced transport in the NCS discharges, and of the transport barrier formation, is developing based on sheared E×B flow stabilization of microturbulence. An extremely rich variety of physics effects provide for exciting and interesting fusion science research, as well as opportunities for control of the transport and transport barrier. The ability to vary the location of the internal transport barrier and to control the magnitude of the local pressure gradient can allow us to generate pressure profiles consistent with high beta stability and bootstrap alignment.

2.1.2. General High-l_i Considerations

A significant experimental basis for a high l_i high performance scenario exists. A number of tokamaks have observed experimentally that the maximum achievable beta increases with internal inductance, and the DIII–D experimental program has established the scaling relation $\beta_{max} = 4\ l_i * I/aB$.(Taylor 90 IAEA). This relation has been supported by a large number of experimental results from other tokamaks. It has also been shown to be consistent with theory and modeling results (Lao 1991), at least for a class of equilibria generally consistent with Ohmically driven current profiles. It has also been shown in a wide range of experiments that the energy confinement time increases with l_i. The increase in confinement has been shown to be a consequence of an increase in magnetic shear and a consequence of an increase in the sheared E×B flow shear. There exists a positive feedback mechanism between the two effects. These high confinement and the high beta results have to date been achieved transiently by ramping down the plasma current or by expanding the plasma size.

Self-consistency of the profiles in steady state does place limitations on the high l_i scenario. Maintaining q_0 slightly above unity and avoiding m/n = 1/1 sawtooth instability has been observed to be a necessary condition in achieving high performance in many tokamaks. We will make the most peaked current density profile possible (highest l_i) consistent with ballooning stability and resulting allowable pressure profiles in the following way. The current profile will consist of the driven seed current and the bootstrap current. The driven seed current will have a top-hat form and is located in the core, with the limitation that $q_0 > 1$. The total current density is equal to the maximum of the local current density and the local bootstrap current. The pressure gradient is limited to remain below the ballooning limit. The resultant current density profile is shown in Fig. 5, and the internal inductance is limited to $l_i \sim 1.1$. The maximum beta stable to ideal ballooning modes in such a case is $\beta_N < 4$ (form $\kappa = 1.8$, $\delta = 0.7$ equilibrium), and the maximum bootstrap fraction is limited to approximately 60% at $q \sim 7$. This scenario is an attractive advanced tokamak scenario, and we think the physics challenges are not very demanding. However, because of its bootstrap current limitations and implications on achievable steady state Q, the high l_i scenario is not our leading scenario.

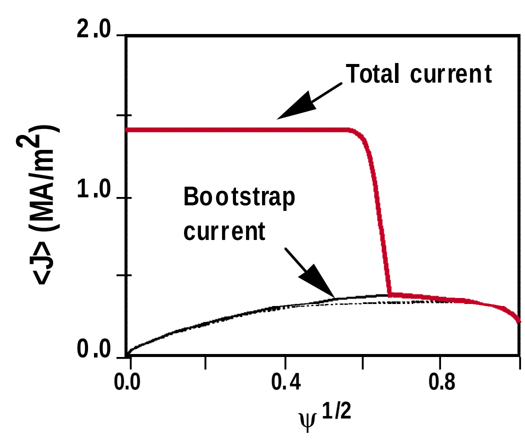


Fig. 5. Self-consistent current profile from high β, high l_i equilibrium β_N =4, l_i = 1.2, q_{95} = 8, q_0 = 1.05.

2.2. THE NEGATIVE CENTRAL SHEAR (NCS) SCENARIO IN DIII–D

The principal approach to the AT in DIII–D is the negative central shear regime. This regime has the best set of characteristics to take forward to a steady-state fusion reactor. The hollow current profile is compatible with the high confinement arising from a transport barrier since the off-axis bootstrap current produced by the transport barrier will produce most of the required off-axis current peak. The rest of the non-inductive current can be either on-axis for central q control or off-axis to supplement and align the bootstrap current peak with the required total current profile. The negative central shear q profile and the broad pressure profile resulting from a transport barrier and q_{min} being at large radius are compatible with high normalized beta. Wall stabilization is also needed owing to the closer proximity of the current peak to the plasma edge. This scenario can be made with either the L–mode or H–mode edge. Which is best for stability and confinement is an active subject of ongoing research.

There is considerable flexibility in this scenario in regard to how the plasma edge is managed and how the interior current and pressure profiles are controlled. It is not clear, for example, how much magnetic shear reversal is needed, even to the limit of zero shear. On three or four separate occasions in the last four years, different people from different viewpoints have constructed AT NCS scenarios for DIII–D using the ONETWO transport code, the stability codes GATO and BALOO, and the transport code CORSICA. We will summarize those scenarios below in order of increasing complexity of the transport modeling rather than in the chronological order in which they were done. They exhibit some different approaches and interests which provide pathways into the variations in experimental approach being currently pursued and we will comment on those pathways into the ongoing experimental program.

2.2.1. Scenarios Using Fixed Profiles

The purpose of this modeling exercise was to demonstrate the potential for intermediate advanced tokamak operation goals at intermediate values of plasma current and toroidal field with the view of a phased installation of a ten gyrotron system (nominal 1 MW/gyrotron source with 70%–80% delivered to the plasma) for ultimate operation at full field and current in DIII–D. For this purpose scenarios with 3, 6, and 10 gyrotrons were developed from B_T = 1.6 T to full B_T = 2 T and I_p ranging from 1.0 MA to 1.6 MA.

The starting point in each case was a stable MHD equilibrium with boundary consistent with the full RDP installation. Only the total pressure is important for the equilibrium, but the non-inductive current density calculations require the pressure to be separated into electron and ion density and temperature. This division is shown in Fig. 6 for the three gyrotron scenario with $\beta_N = 4$.

The density profile was chosen to be consistent with pumped ELMing H–mode discharges at higher q_{95}. These are more peaked than the canonical H–mode density profiles normally shown. Very little effort was directed to make an H–mode edge pedestal or consistency between the edge bootstrap current density and the total current density from the equilibrium because the purpose was to show whether the off-axis ECCD was sufficient in these conditions. The lack of predictive capability of the edge conditions during ELMs would make any detailed reconciliation baseless in any case. The level of the line-averaged density was limited by the empirical rule-of-thumb on DIII–D that the ELMing H–mode density can be varied from n (10^{19} m^{-3}) = 3 I (MA) to 6 I (MA). The actual density was maximized consistent with full non-inductive current. The impurity density profile was chosen arbitrarily to give a constant Z_{eff} = 1.5 across the plasma.

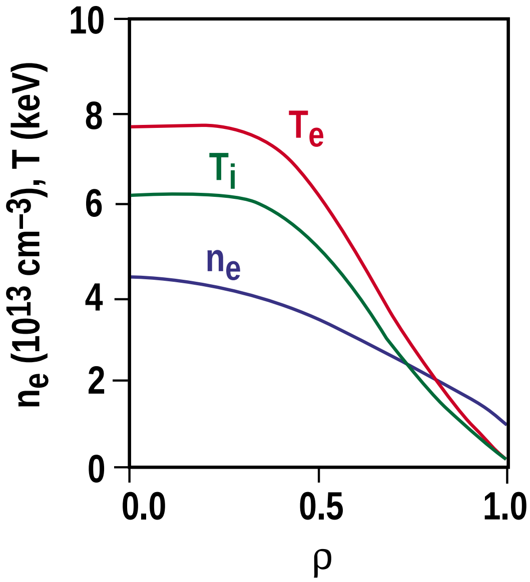


Fig. 6. NCS profiles of temperature and density.

The temperature profiles were chosen to be a constant fixed ratio across the entire plasma. The transport code was run in analysis mode to derive both the local transport coefficients and the global confinement relative to the ITER-89P scaling law. The local transport coefficients were checked to ensure the ion diffusivity was at or above neoclassical and near the electron diffusivity.

The same source calculations in the transport code also provide the non-inductive current densities due to NBI, bootstrap, and ECCD. The scenario was iterated to give zero net ohmic current, not zero ohmic current density at all radii. Again, the goal of our modeling at that time can be seen directly by examining Fig. 7 which shows the total current density from the equilibrium and the non-inductive current densities calculated using the profiles in Fig. 6. It is clear that 2.3 MW of EC power delivered to the plasma under these conditions supplies sufficient current at the half radius to maintain the off-axis current density in conjunction with the bootstrap current. A resistive evolution could have been done and would have resulted in a less reversed q profile, but the degree of negative central shear is not believed to be an essential feature of this scenario.

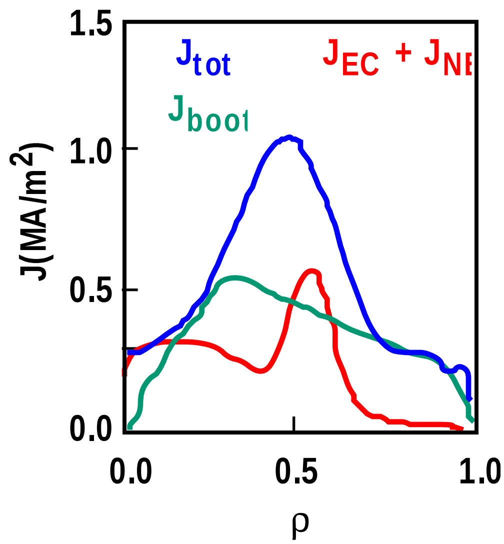


Fig. 7. Contributions to a hollow current profile.

Fixing the profiles is obviously equivalent to fixing the target β_N and H factor for the scenario. These calculations do represent first principles evaluations of where to place the RFCD and the efficiency of the RFCD. These calculations are of value in determining the RF and NBI power levels needed to make the target scenario in terms of current drive and assuming the target values of β_N and H. Scenarios at increasing plasma current and field were developed. Three scenarios are summarized in Table 1. We have labeled these scenarios by the number of gyrotron tubes we believe we will need to carry them out (note: two of the gyrotrons available in 2000 will have limited pulse length and so we have listed four gyrotrons being applied to the "three gyrotron" scenario) and also by the year in which we feel we can begin to attempt these scenarios. These scenario definitions have given focus to the effort in Thrust Area 2 in 1999 to develop transiently the plasma described in the year 2000 column. The plasma described in the year 2001 column is our principal target for the year 2001 demonstration of a sustained NCS AT mode, possibly with an integrated divertor solution.

Table 1
Parameters of NCS Scenarios Using Fixed Profiles.

	4 Tubes (2000)	6 Tubes (2001)
P_{EC} (MW)	2.3	4.5
P_{FW} (MW)	3.6	3.6
P_{NBI} (MW)	4.1	3.8
I_p (MA)	1.0	1.3
I_{Boot} (MA)	0.65	0.9
I_{ECCD} (MA)	0.15	0.2
B_T (T)	1.6	1.75
β_T (%)	4.0	6.3
β_N	4.0	5.3
H_{89P}	2.8	3.5
n (10^{20} m^{-3})	0.32	0.5
n/n_G	0.3	0.4
$T_i(0)$ (keV)	6	8
$T_e(0)$ (keV)	8	9

In order to obtain sufficient current drive efficiency, these scenarios use low densities, a low fraction of the Greenwald limit. These low densities are below where detached divertor plasmas are found, setting the challenge to either raise the scenario density or to the divertor program to develop ways of making radiative divertors compatible with these AT core plasmas. Density control at least is required from the divertor program to meet these scenarios.

2.2.2 NCS Scenario Simulations Using Diffusion Coefficients Derived From Discharges

MHD stability studies of discharges with an internal transport barrier (ITB) show that the stability limit improves with increasing width and radius of the ITB based on a systematic scan of simulated equilibria with model q and pressure profiles. The scenario modeling described in the preceding section began with a total pressure profile consistent with an MHD stable equilibrium and rather arbitrarily divided the total pressure into electron and ion pressure which were then portioned to density and temperature. The scenario modeling described in this section is based on transport coefficients determined from an existing ITB discharge with an L–mode edge which are then scaled to different parameter regimes. To date the studies have been focused on using this approach to achieve the discharge conditions chosen by the method of the previous section. One of the limitations here is the absence of adequate existing shots close enough to the target conditions.

Time-dependent transport simulations were performed using the ONETWO and CORSICA transport codes. First, measured profiles from an ITB discharge with B_t = 2.1 T, I_p = 1.47 MA, β_N = 2 and H_{89P} = 2 were used to calculate thermal diffusivities $\chi_e(\rho)$ and $\chi_i(\rho)$. These calculated diffusivities, with the addition of the ion neoclassical diffusivity, were the baseline model diffusivities used in the time-dependent ONETWO simulations. Since the target parameters for the simulations are different than those of the ITB discharge, the transport coefficients are scaled based on the ITER89P scaling expression. CORSICA simulations have concentrated mostly on the sensitivity of results to other transport models.

In the process of performing the transport simulations, a number of iterations are carried out in order to optimize various choices. The ECH launching direction is optimized to align the electron cyclotron current drive (ECCD) profile with the off-axis bootstrap current profile, and to maximize the ECCD efficiency to overcome the dissipating ohmic current profile. We then evolve T_e, T_i, and current density with a fixed density profile for a period of 10 s. We iterate this transport simulation cycle three times, each with the starting profiles taken from those at the end of the previous cycle. In the last cycle, evolution of the MHD equilibrium is also performed. At the end of each cycle, we test the MHD stability with high-n (BALOO code) and low-n (GATO code) stability.

Figure 8 shows the summary of simulations for q(ρ), T_e(ρ), T_i(ρ), and individual components of the current density profile together with a tabulation of the parameters (Table 2), which can be compared with the case from the previous section. Off-axis (ρ ~ 0.43) ECCD with 3-MW absorbed ECH power in a beam-heated target plasma can sustain an enhanced confinement condition with a bootstrap current fraction of ~60%, normalized beta β_N ~ 2.7 and confinement enhancement factor H_{89P} ~ 2.2. We have not used symmetric fast wave heating yet, which is an option for reducing the impact of the Ohmic current density.

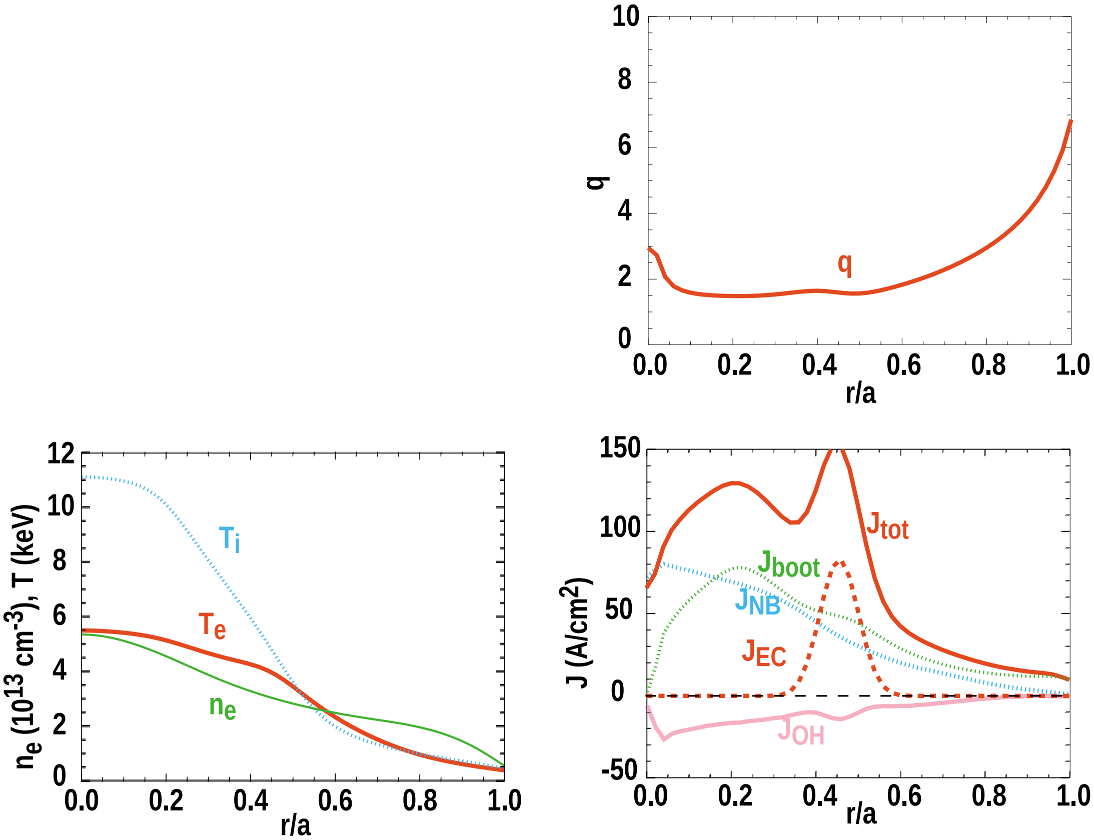


Fig. 8. NCS scenario using empirical transport coefficients.

Table 2
Parameters of L–Mode Edge NCS Scenarios Using Transport Simulations

	4 Tubes (2000)
P_{EC} (MW)	3.0
P_{FW} (MW)	0
P_{NBI} (MW)	6.2
I_P (MA)	1.11
I_{Boot} (MA)	0.59
I_{ECCD} (MA)	0.18
B_T (T)	1.6
β_T (%)	3.1
β_N (%)	2.7
H_{89P}	2.2
n (10^{20} m^{-3})	0.37
n/n_G	0.40
$T_i(0)$ (keV)	11.2
$T_e(0)$ (keV)	5.4

2.2.3 NCS Scenarios Using Models of Transport Barrier Formation

Scenarios using models of transport barrier formation have been produced twice. A table of numbers for these scenarios is given in Table 3. These scenarios were constructed by 1-D simulations using the ONETWO code. All of the scenarios lie in the range β_T 5%–11% at full field in DIII–D. They are at plasma currents of 1.6–2.2 MA and employ strong shaping (κ = 2.1, δ = 0.8). They employ a total of 15–20 MW of total heating and/or current drive power, made up of roughly equal contributions of NBI, ECH, and FW. They all employ fast wave heating to achieve a high core electron temperature. Except for Case 2, all these cases require at least 6 MW EC power delivered to the plasma. These considerations lead us to believe that a 10 gyrotron EC system will ultimately be needed to form the NCS plasmas at full field and current in DIII–D.

They all seek steady-state negative central shear current profiles for stability at high β_N. They also seek high bootstrap fractions and make up any difference in the plasma current and the bootstrap current by use of RF current drive. The resulting plasmas have rather low ρ_* and very low ν_*, but they match up well along dimensionless parameter scaling paths to future tokamak devices (Section 2.4 of the Five-Year Plan). The

Table 3
Parameters of DIII–D Scenarios

Case	1	2	3	4	5
β (%)	7.5	5.0	8.1	8.7	11.5
β_N	5.7	3.8	6.2	5.8	6.0
I_p (MA)	1.6	1.6	1.6	1.8	2.2
$I_{bootstrap}$	1.07	1.45	1.85	1.92	2.1
I_{ECCD}	0.35	0.50	0.03	0	0
I_{FWCD}	0	0	0	0	0
I_{NBCD}	0.25	0.11	0	0	0
I_{OH}	–0.07	–0.46	–0.28	–0.12	–0.10
q_{95}	6.5	5.0	5.0	5.3	3.6
q_0	3.8	2.3	2.5	3.9	3.4
q_{min}	2.6	3.3	2.1	2.3	
$T_i(0)$ keV	15	12.3	18.5	14.5	19
$T_e(0)$ keV	8.5	9.7	7.0	12.7	13
$n_e(0)$ 10^{20} m^{-3}		0.59	0.89	0.72	0.88
$\bar{n}$ 10^{20} m^{-3}	0.57	0.35	0.54	0.48	0.53
n_{edge} 10^{20} m^{-3}		0.23	0.23	0.23	0.21
$\bar{n}/n_G$	0.4	0.26	0.4	0.32	0.3
P (MW)	20	14	12	14	14
P_{NBI} (MW)	6.5	4.0	4.0	4.0	4.0
P_{EC} (MW)	7.0	6.0	4.0	6.0	6.0
P_{FW} (MW)	6.5	4.0	4.0	4.0	4.0
W (MJ)		1.25	1.3	4.6	6.0
τ_E (s)		0.21	0.29	0.28	0.4
H_{89P}	3.5	3.4	4.4	4.0	4.95
ρ_{*e} at $\bar{T}_e$	2.3×10^{-4}	2.5×10^{-4}	2.1×10^{-4}	2.8×10^{-4}	2.9×10^{-4}
ρ_{*i} at $\bar{T}_i$	1.3×10^{-2}	1.2×10^{-2}	1.5×10^{-2}	1.3×10^{-2}	1.5×10^{-2}
ν_{*e} at $\bar{T}_e$	1.2×10^{-2}	4.2×10^{-3}	8.0×10^{-3}	2.9×10^{-3}	2.1×10^{-3}
ν_{*i} at $\bar{T}_i$	2.7×10^{-3}	1.8×10^{-3}	0.8×10^{-3}	1.6×10^{-3}	0.7×10^{-3}

Case 1: SSC-VH [Turnbull PRL 74, 718 (1995)]
Case 2: β = 5%, P = 16 MW, n & v_ϕ transported
Case 3: β = 8%, P = 15.2 MW, n & v_ϕ transported
Case 4: β = 8%, P = 17 MW
Case 5: β = 11%, P = 17 MW

scenarios all have rather low densities and high temperatures. The combination of low density (well below the Greenwald limit) and high power will make it particularly challenging to obtain radiating, detached divertors in these scenarios.

For reference, the first scenario is that published by (Turnbull, 1995, see also St. John IAEA 1994 and Taylor EPS 1994). At that time, DIII–D had seen plasmas with hollow current profiles, very high central betas (calculated to be second stable) and had also seen in other discharges the VH–mode, a transport barrier formed around $\rho = 0.8$. The scenario described considered combining these two features into what was called then Second Stable Core VH–Mode (SSC-VH). The inverted q profile and suitably broad pressure profile was shown through stability calculations to give β_N of 5.7 assuming wall stabilization. Various transport models were used for the electrons including INTOR scaling, the Rebut-Lallia-Watkins model and the Hsieh model for electrons. Essentially the ion diffusivity was taken to be neoclassical near the core (the transport barrier model here was a small multiplier times neoclassical ion transport inside the radius of q_{min}) and rising to 5 times neoclassical near the edge. A combination of bootstrap current which peaked off axis and off-axis ECCD were used to sustain the hollow current profile. On-axis NBCD was used to control the central current density. Fast Wave heating sustained the core electron temperature. A limitation of this scenario was the use of a fixed density profile; no density transport was considered. The rather broad density profile used still contributed a significant bootstrap current. Steady-state solutions were found with the required current profiles and pressure profiles for the high values of β_N in this scenario.

For the Five-Year Plan, we constructed transport simulations using a full but complex model of E×B shear stabilization of turbulence to dynamically form the transport barrier in the simulation. The current profile was evolved to steady state verifying the compatibility of the transport barrier with the second stable core. The model is diagrammed in Fig. 2.3–1 on page 2.3–4 of the Five-Year Program Plan. The model calculates the turbulence shearing rate with no free parameters from the Hahm-Burrell formula based on the evolving density, temperature and rotation speed profiles. Then the local value of $\omega_{E\times B}$ is compared to a model of the turbulence growth rate. This prescription for E×B shear suppression is based on gyrofluid turbulence simulations of Waltz, 1994. The location where the transport barrier forms depends upon both the growth rate profile (which tends to rise from the center) and the source (heating, momentum and fueling) profiles. The barrier usually forms first at the edge due to the high power flux density and the strong density gradient (fueling). The H–mode edge can be suppressed (in the model) by a combination of high edge radiation and low recycling. We did so in order to concentrate on the core transport barrier properties and not on the more difficult to model edge L-H transition. A transport barrier then forms first in the core owing to the peaked heating and/or momentum sources. In the experiments, the reduction of the ITG mode growth rates due to hot ions and fast ion dilution have been found to aid internal transport barrier formation. The negative magnetic shear also eliminates MHD ballooning modes. Raising the power makes the internal transport barrier expand as the E×B shear pushes out against the rising growth rate. This model of transport barrier formation contains many feedback loops, since the radial electric field depends on all the profiles, and a very rich set of anticipated phenomena. It is also hard to run since both the model growth rate and $\omega_{E\times B}$ depend on local gradients.

The cases considered all model an L–mode edge. The transport barrier forms where the turbulence shearing rate from the radial electric field exceeds the local growth rate of the turbulence. When density transport is turned on, the strong local fueling source at the edge easily forms an edge transport barrier which can quickly lead to excessive edge pressure gradients. A large part of the NCS research thrust is aimed at controlling the edge pressure. To avoid this problem, we imposed a large edge growth rate to keep the edge in L–mode and fixed the edge density. The thrust to use an L–mode edge is one of DIII–D main AT thrusts but considering the high power flow through that edge, substantial mantle radiation or other means to suppress the L-H transition will have to be found. These are issues for future experimental and simulation work.

Despite the complexity of the model, the results in Table 3 represent another set of internally consistent numbers of target β_N and H factors with the required power levels and locations of current drive required to produce the necessary current profiles. The target β_N and H factors are large and represent ultimate goals for the DIII–D AT Program. Even with the high H factors, a 10 gyrotron system is needed and the total system power of 20 MW will be a challenge to the divertor power handling capability in long pulse.

For the near term scenarios, perhaps some of the qualitative features seen in these transport barrier modeling efforts are worth noting. The density transport equation was turned on in Cases 2 and 3 and a transport barrier was allowed to form in the density channel (Fig. 9). Density gradients are more effective than temperature gradients in creating bootstrap current and for that reason, we obtain more bootstrap current than in the original SSC-VH scenario. Also, we have moved the transport barrier further out in radius and that also increases the total bootstrap current. We find it rather easy (in fact too easy in these simulations) to obtain full bootstrap current. It appears that with central fueling from beams or pellets and a longer time for the density to accumulate, we should see strong transport barrier formation in the future through density gradient with accompanying large bootstrap fractions. This research area has only just started on DIII–D. We completed the central Thomson scattering system and it is now operational on DIII–D. With it we will finally be able to see what is happening in the density channel when transport barriers form in the plasma interior. The UCLA group installed an x–mode interferometer on DIII–D about a year ago so we could get an early glimpse of a density transport barrier. One such profile (Fig. 10) shows a spectacularly high density gradient, showing us what exciting phenomena may lie ahead in these studies. We have a plan in the 1999 campaign in the Thrust 7 on ITB control to use the inside launch pellet injection together with the counter beam injection to stimulate the formation of transport barriers in the density profile.

Another interesting but not fully understood results was that the ECH was very effective at moving the location of the transport barrier. To see such dynamics was a principal reason for using the complex E×B shear model. The ECH deposition profile is about as narrow as the gradient regions of the transport barrier, and so the ECH is a precision tool for barrier control. We found that ECH applied just outside the radius where a transport barrier was beginning to form would draw the transport barrier out to larger radius. Equally striking but not so positive was the effect of ECH when applied inside a formed transport barrier. The transport barrier was found to retreat to just inside of the ECH absorption layer. In the model this was due to the fact that the model growth rate increased with the electron temperature gradient but the E×B shear only depends on the ion temperature gradient. Thus, electron heating caused a loss of the E×B shear suppression. A retreat of an existing internal transport barrier with central ECH heating has been observed on DIII–D. Linear growth rate calculations suggest that the excitation of electron temperature gradient modes may be the cause.

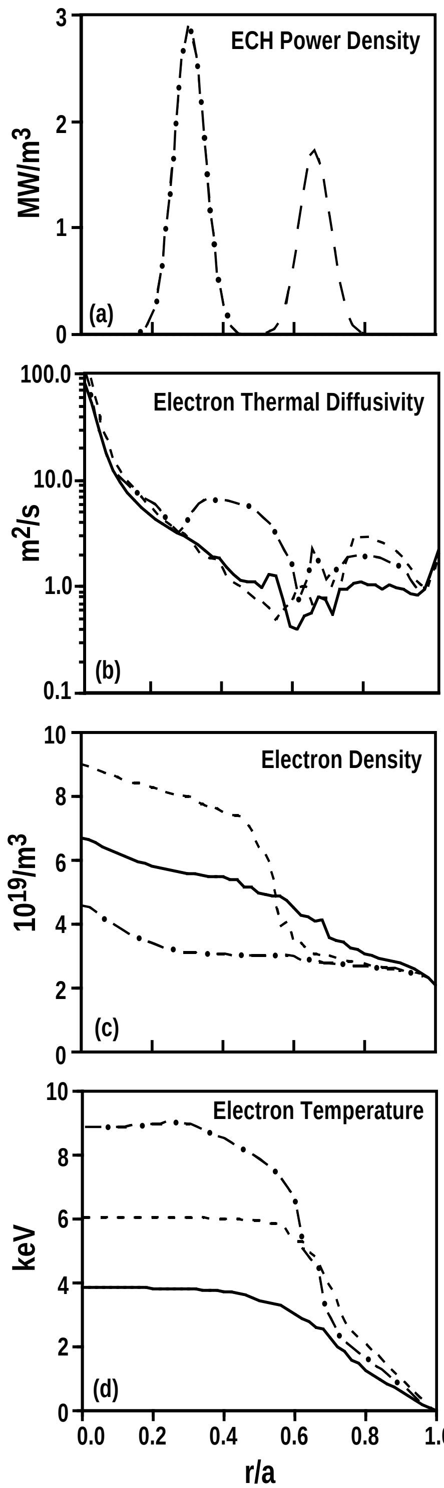


Fig. 9. ECH barrier control is illustrated by a three-case comparison: 6 MW NBI only (solid), 6 MW NBI + 6 MW ECH at r/a ~ 0.7 (dashed), 6 MW NBI + 6 MW ECH at r/a ~ 0.3 (dot dashed). Shown are profiles of (a) ECH power density, (b) model electron thermal diffusivity, (c) electron density, and (d) electron temperature.

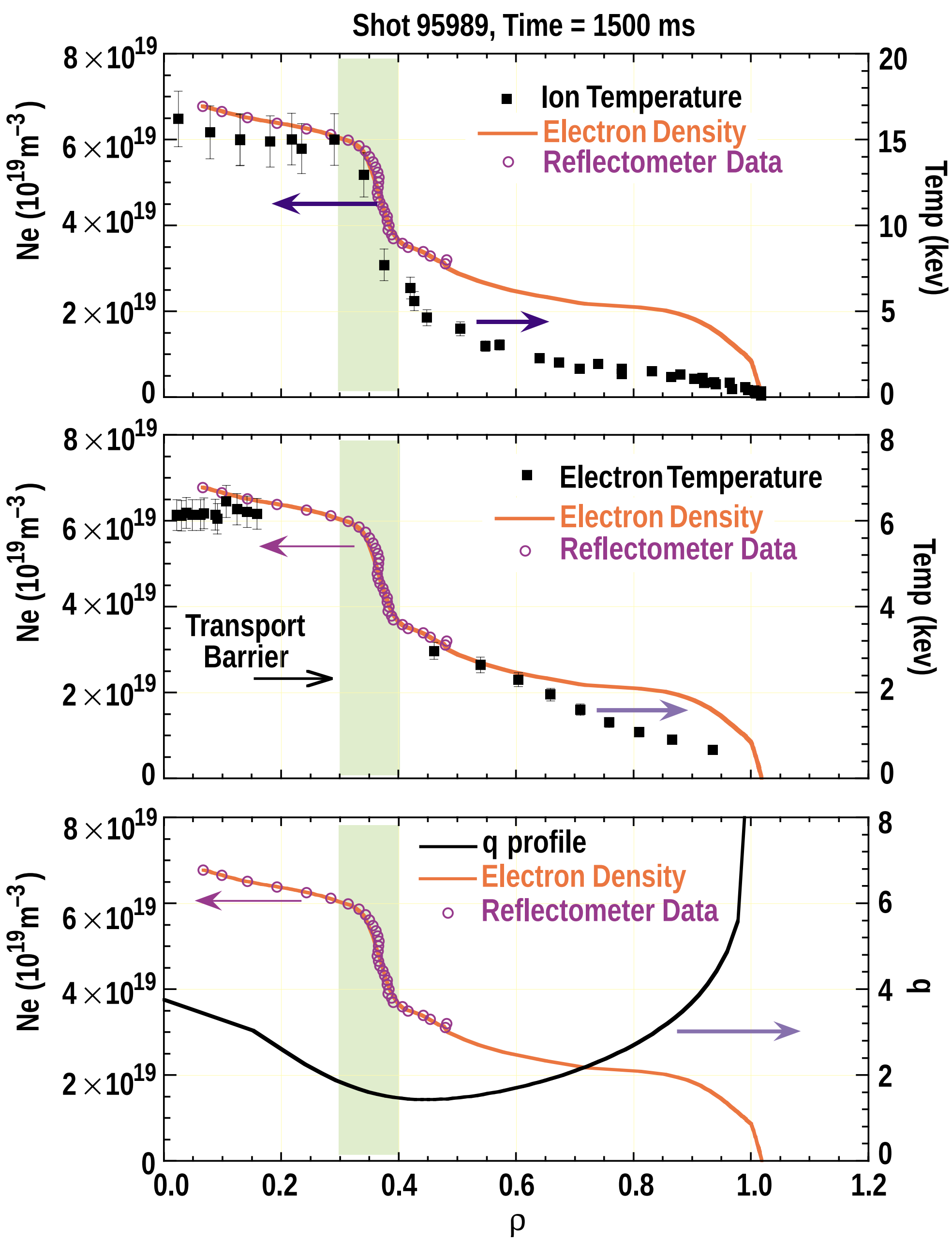


Fig. 10. Transport barrier in the density.

These modes are not expected to be stabilized by E×B shear since they have very large growth rates (and small wavelengths).

The physics picture mentioned above in which $\omega_{E\times B}$ pushes out radially against a radially rising turbulence growth rate suggests that the way to move the transport barrier out in radius is just to add power or momentum inside the formed transport barrier. However, because the transport coefficients are so low inside the barrier, increasing the power, especially localized power, can produce wild swings in local gradients affecting not only $\omega_{E\times B}$ but stability as well. The way through these complications is to increase power slowly. Using this approach, we were able in 1998 to make an internal transport barrier discharge that lasted the whole length of the neutral beam pulse (5 s). This was done at low current where the plasma was beset with Alfvén eigenmodes which had the beneficial effect of throwing out enough fast ions so that the central NBCD was weak and the central q stayed high. When such discharges were attempted at higher current, the Alfvén eigenmodes were eliminated but the central NBCD drove q_0 below one and instabilities terminated the high performance phase. In the 1999 campaign in Thrust 7 on ITB control, counter NBI will be used to keep q_0 high and this approach of gradually increasing power to expand the transport barrier radius may have even more success.

The various cases have varying assumptions about how low the transport rates become inside the transport barrier. In DIII–D we have already seen ion neoclassical transport rates all across the cross section so this assumption for the residual transport was made in all cases. But it is clear that similarly low levels for transport rates for electrons and particles in DIII–D are too good. Beta limits would be quickly exceeded. DIII–D does not presently see as much transport reduction in the electron and particle channels as in the ions and apparently will not require it to reach the scenarios shown. Parenthetically, we have done similar transport simulations for spherical tokamaks and there the higher beta values sustainable can make use of the very low transport rates for electrons and particles that may be achievable with a full exploitation of E×B shear stabilization.

These are some of the interesting phenomena we have seen in our initial exploration of the possibilities for AT physics in the plasma core. The simulations presented give a feeling for the parameter regimes achievable, the power levels in various systems to achieve them, the density and edge control that may be required. But the main value of such simulations is to open a wide vista of new phenomena that should open up as the auxiliary capabilities of DIII–D are developed toward the goal of long pulse sustainment of AT operating modes.

2.3. THE HIGH INTERNAL INDUCTANCE SCENARIO IN DIII–D

The high internal inductance (high ℓ_i) scenario is the second possible approach to AT performance being pursued in DIII–D research. This scenario is motivated by the well-established experimental observation that both the beta limit and the confinement multiplier increase approximately linearly with ℓ_i. This scenario has the advantages that the current and q profiles are monotonic, requiring less precise tailoring, that the current density and pressure gradient at the plasma edge are not so large as to require wall stabilization to reach $\beta_N \approx 4$, and that the required external current drive would be peaked at the axis which is more efficient and easier to implement than in the NCS scenario.

In order to achieve high values of β_N, relatively broad pressure profiles are required. This places the regions of high pressure gradient toward the discharge edge and, in discharges with large bootstrap current fraction, produces relatively broad current profiles. So, operation at high β_N and high bootstrap fraction tends to lower the self-consistent value of l_i.

The achievable value of β_N is expected to be consistent with the empirical scaling $\mathrm{Max}(\beta_N) \approx 4\, l_i$. Thus, a $\beta_N \approx 4$ operating point would have $l_i \approx 1$, a larger value than in the NCS scenario but smaller than the maximum values achieved in previous research on l_i scaling. Simulations have shown that bootstrap fraction in the range 50%–70% can be obtained self-consistently with $l_i \approx 1$. A sample equilibrium of this class is shown in Fig. 11. With strong shaping ($\delta \geq 0.7$) and flat J and q profiles in the center of the plasma, an optimized equilibrium can be found which is stable to n = 1 ideal modes and to n = ∞ ballooning at $\beta_N = 4$ without a conducting wall. This case has not as yet been examined for transport requirements although the bootstrap current profile is required to be consistent with the assumed pressure profile.

The achievable value of l_i for a given bootstrap fraction can be increased by reducing the value of the safety factor on axis. Reducing q_0 below 1 requires stabilization of the sawtooth instability. Previous work has indicated that sawtooth stabilization is possible with rf heating. A key question for the high l_i scenario is whether rf sawtooth stabilization can be done while maintaining the other requirements (high f_{BS} and β_N). Note that sawtooth stabilization should remove a primary source of perturbations which can initiate neoclassical tearing modes, which in turn may raise the beta limit. An example of this scenario is shown in Fig. 12. This example was developed with the same rules as the NCS example cited in Section 2.2.1, i.e., primarily to assess heating and current drive requirements.

Thus there are two distinct versions of the "high l_i" scenario, one requiring some current profile tailoring to maintain q_0 above 1 and sufficiently flat in the central region, and the other requiring effective stabilization of sawteeth.

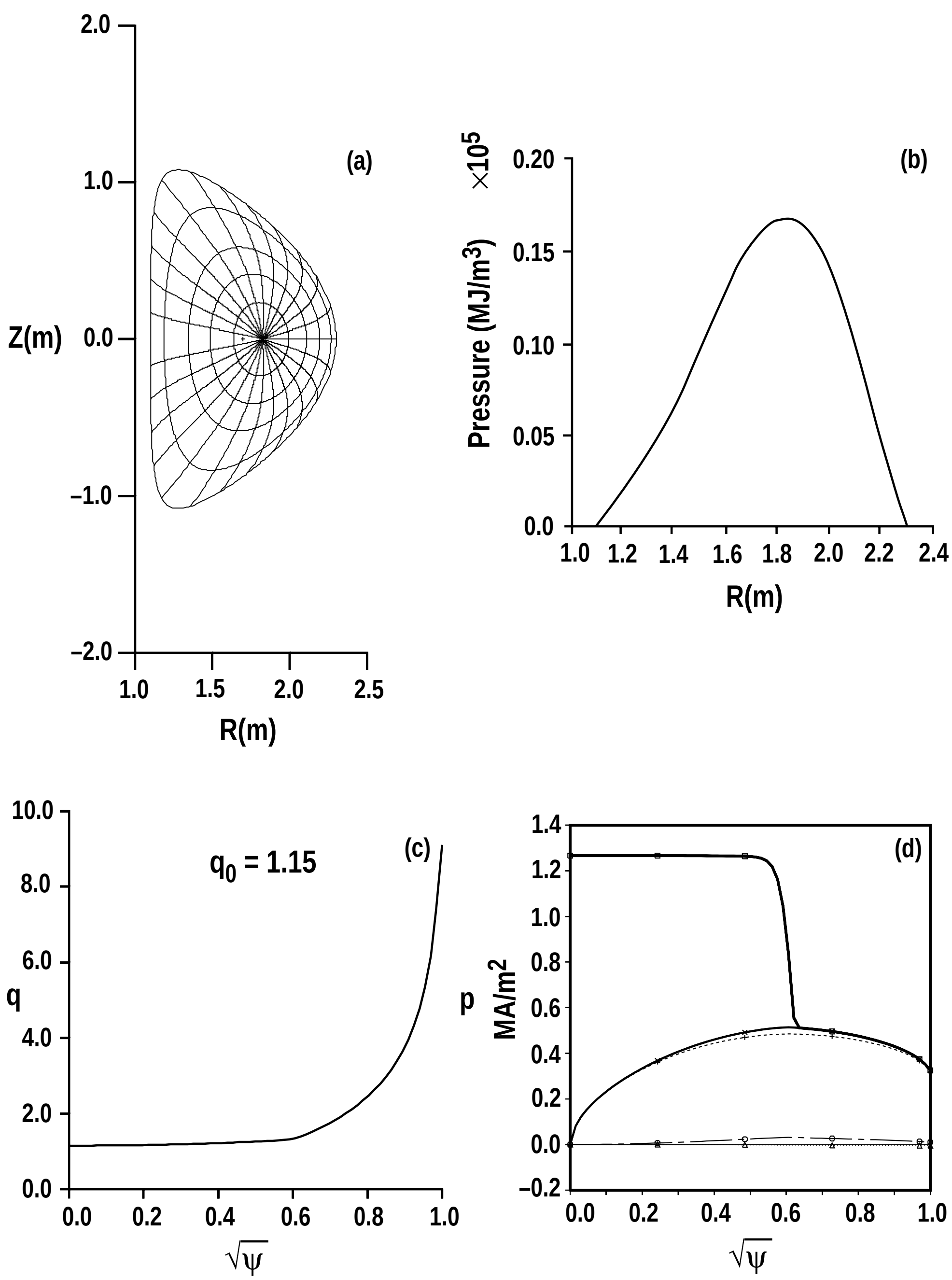


Fig. 11. Optimized ℓ_i, high β_N, edge aligned bootstrap equilibrium in the full-size DIII–D configuration with R = 1.7 m, a = 0.6 m, κ = 1.8, δ = 0.7, B = 1.9 T, I_p = 1.1 MA, q_0 = 1.15, ℓ_i = 0.92, $p_0/\langle p\rangle \sim 3.0$, β_N = 4.0, and f_{BS} = 70, q_{95} = 6.5. (a) Flux contours, (b) pressure profile across the midplane as a function of major radius, (c) q profile as a function of $\tilde{\psi}^{1/2}$, and (d) flux surface averaged toroidal current densities as function of $\tilde{\psi}^{1/2}$. Here, solid squares represent the total plasma current. The sum of bootstrap (dotted curve), diamagnetic (open circles), and Pfirsch-Schluter (open triangles) contributions is represented by crosses. parison: 6 MW NBI only (solid), 6 MW NBI + 6 MW ECH at r/a ~ 0.7 (dashed), 6 MW NBI + 6 MW ECH at r/a ~ 0.3 (dot dashed). Shown are profiles of (a) ECH power density, (b) model electron thermal diffusivity, (c) electron density, and (d) electron temperature.

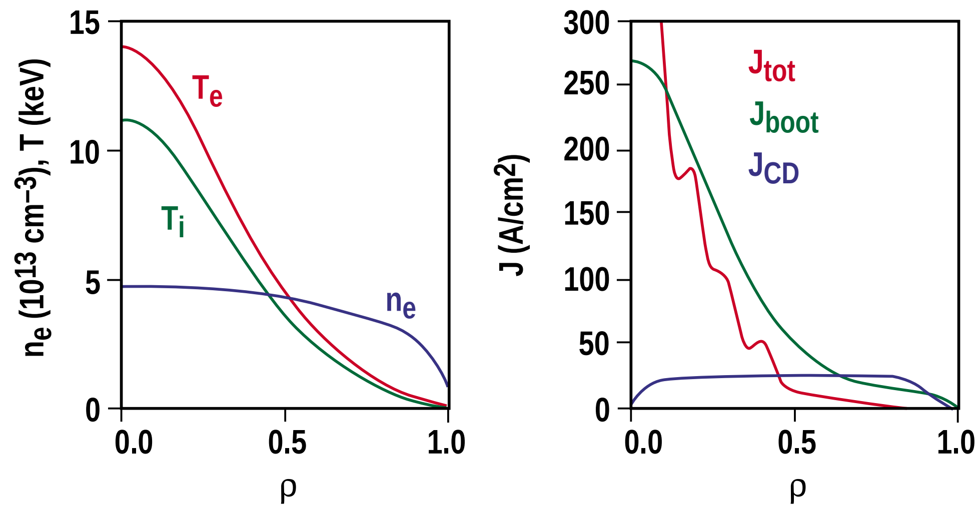


Fig. 12. A high l_i steady-state with $\beta_N H > 10$ can be sustained by a 3 MW ECH system. The parameters are: B = 1.9 T, l_i = 1.6, β_N = 2.5, I_{BS} = 0.41 MA, I = 1.0 MA, β_N = 4.0, P_{FW} = 3.6 MW, I_{FW} = 0.17 MA, q_0 = 0.55, H = 3.0, P_{EC} = 2.4 MW, I_{EC} = 0.14 MA, q_{95} = 5.6, n = 4.2×10^{19} m^{-3}, P_{NB} = 5.0 MW, I_{NB} = 0.27 MA.

One criticism of the high l_i scenarios has been that excessive external current drive power would be needed in a reactor. To examine this question explicitly, we have looked at spreadsheet modeling of three high l_i cases and compared them with ARIES-RS. Fixed parameters (with Aries-RS values in parentheses) are R = 5.0 (5.52) m, a = 1.8 (1.38) m, κ = 1.8 (1.7), δ = 0.7 (0.5), q_{95} = 6.5 (3.5), P_{fusion} = 2500 (2167) MW, and $n/n_{Greenwald}$ = 0.95 (1.78). Some of the results are summarized below.

	l_i = 1	l_i = 1.25	l_i = 1.5	ARIES-RS (l_i = 0.42)
β_N	4	5	6	4.84
q_0	1.15	0.85	0.55	2.78
B (T)	7.5	6.58	5.95	7.98
I (MA)	15	13.15	11.9	11.3
f_{bs}	0.61	0.67	0.60	0.88
P_{CD} (MW)	169	126	123	81
H_{89P}	2.1	2.44	2.65	2.35

The benefits of high l_i operation are clear. If satisfactory sawtooth suppression is possible and the increase in β_N can be demonstrated, the q_0 = 0.55 case has significantly lower magnetic field than ARIES-RS, with roughly the same total current. Although the driven current fraction increases by 233%, from 0.12 to 0.40, the required external power increases by only 52%. This is because the current is driven at the axis, where T_e is high and trapping is small making the current drive much more efficient. Further, because the current is driven at the axis, a less complex profile control system is needed.

The multi-year goal of research for the high l_i operating mode is to determine feasible scenarios for steady-state high l_i discharges in DIII–D consistent with the available tokamak resources. A goal would be to maintain elevated l_i values for twice the inductive decay time and confirm that the corresponding increase in confinement and stability is also maintained. The issues to be resolved over several years of work are:

1. Establish whether sawtooth stabilization is both possible and practical.
2. Establish the practical limits to β_N in the two high l_i scenarios without additional wall stabilization. Do the linear relationships between β_N and l_i, and between H and l_i extend to $q_0 < 1$ cases?
3. Establish the current drive requirements for steady-state sustainment of these two scenarios. How much current profile control is needed for the $q_0 > 1$ case?
4. Development of entirely self-consistent scenarios to find the optimum combination of current, density, and temperature profiles.
5. Select the $q_0 > 1$ or the $q_0 < 1$ approach.

Regrettably, due to the fierce competition for run time on DIII-D, we were not able to allocate any run time to a research thrust in this area for 1999. As it appears that it will also be difficult to find time in 2000, we have shown the start of this research path in 2001.

To outline the possible content of a future plan to pursue the high l_i scenario, we list here a simplified three-year view of the necessary research:

Goal: $\beta_N \bullet H_{89P} > 10$ with no inductive flux

<u>Year 1</u>

Demonstrate sawtooth stabilization for >1 s and validate the stabilization model. This includes modification and commissioning the ABB transmitters for operation at 60 MHz.

Develop the 3 MW ECH target scenario with $\beta_N \bullet H_{89P} > 10$ transiently. This includes FW coupling studies under the appropriate edge conditions and identification of core pressure limits.

Year 2

Demonstrate the 3 MW ECH integrated scenario. Develop a 6 MW ECH target scenario.

Year 3

Demonstrate the 6 MW ECH integrated scenario. Develop a 10 MW ECH target scenario.

2.4 A THREE YEAR VIEW OF THE DIII–D AT PROGRAM

The main aim of the next three years in the AT area is to develop the physics understanding as well as control tools and techniques to attempt an integrated, sustained demonstration of the Advanced Tokamak NCS scenario in 2001. The Advanced Tokamak hardware development plan needed to accomplish this is shown in Fig. 13.

The major hardware system needed is the increase in the number of installed 1 MW, 110 GHz gyrotrons from 3 to 6. The new gyrotrons, beginning with Unit 3, are equipped with diamond windows and can support 5–10 second operation in DIII–D. Experiments in 1999 will be conducted with three gyrotrons. Two of them are limited in pulse length to under 2 s. Consequently experiments in 1999 just begin to explore the use of ECH and ECCD. For the 2000 campaign, we expect to have a fourth gyrotron working. This will enable us to attempt the NCS scenario identified as four tubes (2000) in Table 1. Six gyrotrons will be available in the year 2001 so we can then begin experiments attempting the NCS at the higher parameters identified as six tubes (2001) in Table 1. For density control, the completion of the upper RDP private flux baffle and pump in 1999 is expected to give us the required density control for either high triangularity plasmas using the upper RDP pumps or for low triangularity plasmas using the lower pump. The completion of the upper RDP will allow resumption of the studies of optimizing the core/divertor plasma performance balance by being able to retain neutrals better in the divertor and by being able to implement the scheme of impurity retention in the divertor using copious flows in the scrape-off layer (puff-n-pump in local jargon). This effort to make a divertor compatible with an AT core may join the NCS scenario effort in 2001. The wall stabilization work begins this year with initial development of feedback control using a six-coil system but only one power supply driver. Additional power supplies will be available in 2000. An 18-coil resistive wall mode (RWM) feedback system should be in place for the 2002 campaign.

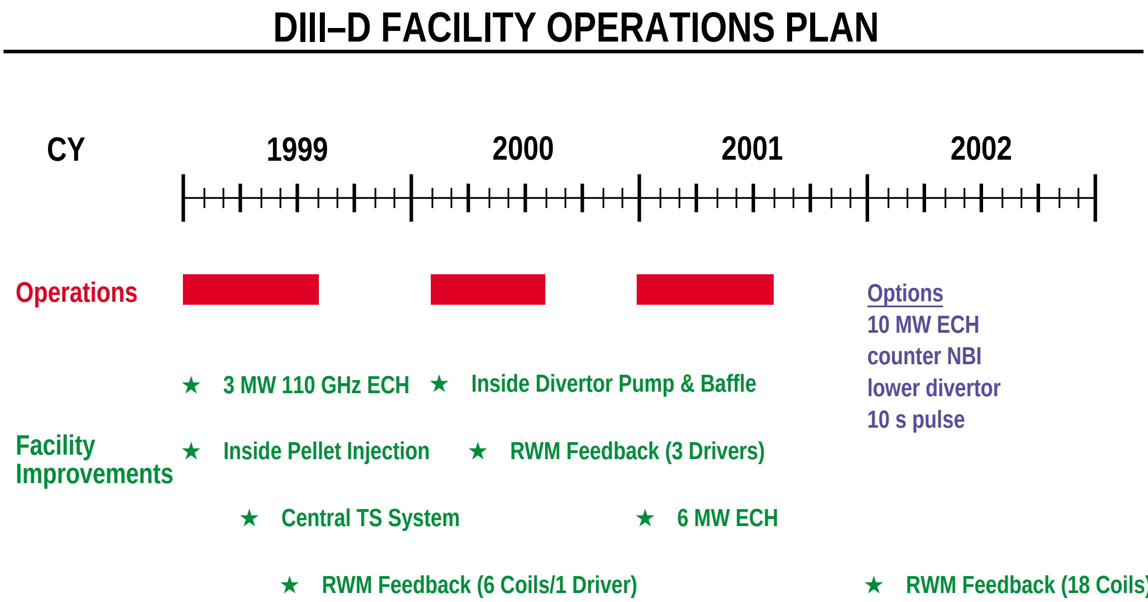


Fig. 13. DIII–D facility capabilities.

The research plan logic that is consistent with this DIII–D facility capability is shown in Fig. 14. The six thrust areas for this year are shown under 1999. We anticipate that this year's work on NCS plasma development and the work on internal transport barrier control will merge in 2000 into using four gyrotrons to begin exploring the AT scenario with $\beta_N \sim 4$, $H \sim 2.5$, $I_p = 1.0$ MA, $B_T = 1.6$ T. The emphasis will be on demonstrating the existence of the plasma in the targeted parameter regime and some modest extension of the high performance duration consistent with available EC power and what we have learned from the edge stability and neoclassical tearing mode work. This main line of attack will carry on into the year 2001 when six gyrotrons will be available to move the scenario up to $\beta_N \sim 5$, $H \sim 3$, $I_p = 1.3$ MA, $B_T = 1.8$ T and with an increased prospect for sustainment of the high performance phase. If this mainline is successful, we will want to carry it forward to full field and current operation in DIII–D in the year 2002–3 time frame. This will require expanding the 1 MW per unit system to 10 units or upgrading the 1 MW units to 1.5–2 MW units during 2002 and 2003. Other optional facility improvements for the period 2002 and beyond are listed in the figure.

The ITB work in 1999 is being carried out mainly with counter injection for the reasons discussed below in the synopsis of 1999 Research. If this line proves effective, then either more counter injection operation may be called for or in 2001 rf counter current drive on axis may be subtituted for the counter NBI. Among the various possible facility upgrade decisions to be made in 2001, a decision must be made about whether to operate with all counter beams or to turn one beamline around to the counter direction as is proposed as an option in the Five-Year Program Plan.

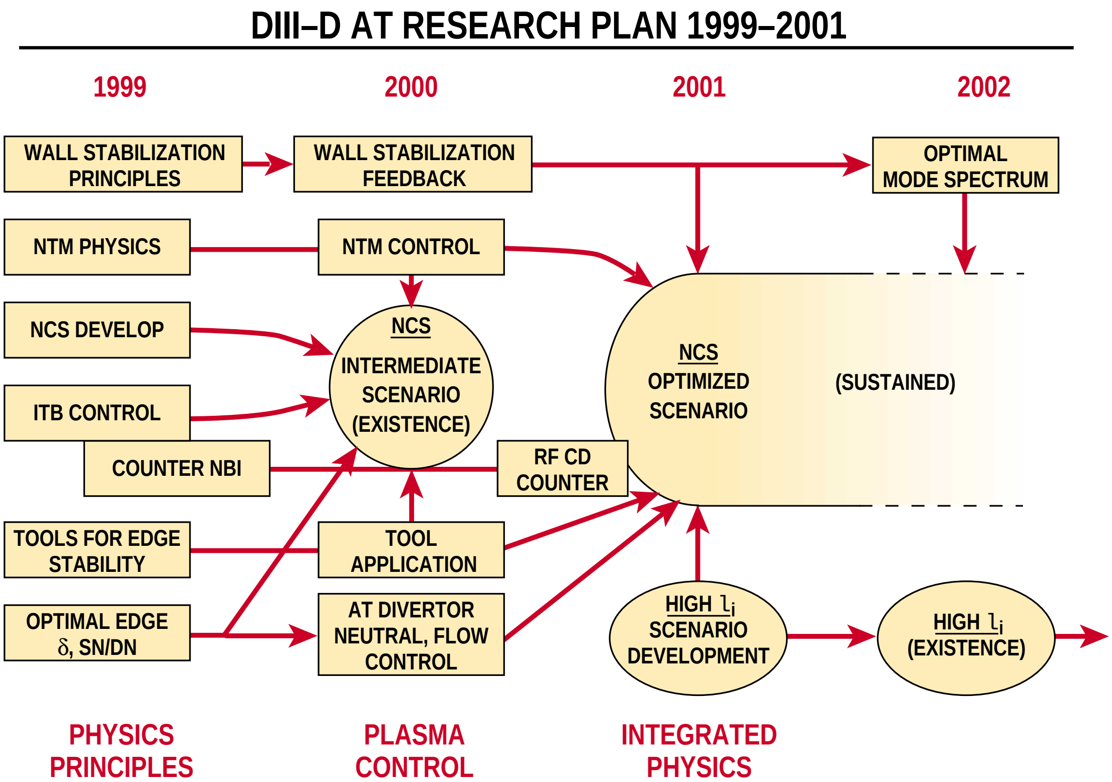


Fig. 14. DIII–D Research Plan 1999–2001.

The work on neoclassical tearing modes in 1999 is mainly to build a base of scientific understanding and perhaps make first attempts at shrinking island sizes or stabilizing the modes. The work will shift to suppression of the modes in 2000 and must begin to make a contribution to a stable, extended pulse NCS scenario. The ECCD is the principal tool to be used.

The work on wall stabilization is expected to proceed independently of the mainline NCS scenario work until 2001 owing to the large hardware buildup required and the complexity of getting the feedback arrangements working. The research work in 1999 will be done with a system gradually improved to six coils and three independent drivers. We expect a contribution from the six coil system to sustaining plasmas above the no-wall limit in the integrated NCS campaign in 2001. We expect the 18 coil system with its more optimal mode spectrum to make a contribution to stabilizing the plasmas in the NCS scenario in 2002.

The Thrust 1 on edge plasma stability is mainly exploring various tools to promote edge stability in 1999. In 2000, one or more tools must be carried forward to application in order that the year 2000 NCS scenario can last a significant amount of time. These techniques for edge stability will be augmented in 2000 by the upper RDP hardware which should increase the ability to use puff-n-pump techniques for promoting edge radiation and divertor radiation. This edge stability work will fold into the year 2001 NCS scenario work.

The work on Thrust 5 (Optimal Edge δ, Single-Null/Double-Null) in 1999 is related to Thrust 1 in that it seeks data to enhance our basic understanding of how to optimize the plasma edge, both just inside and just outside the separatrix. This work will make a contribution to the NCS scenario in 2000 in the area of edge stability and in the choice of triangularity and degree of double-null operation. But this thrust is expected to be mainly completed in 1999, with remaining work getting done in the relevant topical science area. It is anticipated that a new thrust in the divertor area in 2000 will focus on the use of the new upper RDP hardware for density control, neutral control, and plasma flow control in order to prepare to make a contribution to the NCS scenario in 2001.

Owing to the limited run time on DIII–D, we have not been able to allocate run time to the high ℓ_i thrust in 1998 or 1999. Although we intend to start some work in this area on limited topics in 2000, we show a thrust level effort beginning on the high ℓ_i scenario in 2001, focussing on development of the basic building blocks of the scenario. In 2002, we would begin the existence proof type work on the scenario. Consequently work on the high ℓ_i scenario will lag work on the NCS scenario by about 2 years.

Some additional three year view considerations on the thrusts that are expected to continue during that period are given below.

2.4.1. Negative Central Shear (NCS) Thrust

Three-Year Goal

Demonstrate normalized tokamak performance more than twice that of conventional ELMing H–mode with no inductive flux using the negative central shear core transport barrier approach for a duration limited only by hardware constraints.

Goal: β_N*H_{89P} > 10 in plasma with a core transport barrier with no inductive flux.

<u>Critical Path Items</u>

Tool Development.

Gyrotron commissioning (up to six gyrotrons operational)
EC laucher commissioning
Validation of ECCD in ELMing H–mode plasmas
Edge instability control
Particle (density) control
Feedback control with the PCS

Physics Issues.

Assessment of effects of q_{min} and $q_0 - q_{min}$ on stability and transport barriers.
Assessment of confinement when T_e approaches T_i.
Assessment of impurity accumulation.

Draft Three-Year Outline Plan

Issues to address in 1999.

- Develop targets transiently at the parameters from the scenarios and assess the impact of q_{min} and $q_0 - q_{min}$. Iterate with modeling. This would include direct reproduction of a plasma of this β_N*H_{89P} at any density and T_e/T_i, studies of particle and impurity control with the present baffle and pump, and demonstration of any core transport barrier plasma with high H and $T_e/T_i = 1$.
- Complete off-axis ECCD model validation.
- Commission the third gyrotron (first MW long pulse gyrotron) and new launcher.
- Validate ECCD in ELMing H–mode
- Develop the necessary sensors for PCS feedback control and begin handoff experiments from formation to sustainment phase.
- Assess present capability for particle control.
- Assess impact on confinement of T_e approaching T_i.

Issues to Address in 2000.

- Adapt preliminary Thrust 1 candidate edge instability control to transient targets.
- Apply new RDP hardware for particle control.
- Commission new gyrotrons and new launchers.
- Demonstrate PCS control.
- Demonstrate 3 MW ECH integrated solution
- Develop 6 MW ECH target scenario
- Commission 6 MW system and second pump

Issues to Address in 2001.

- Integrate particle, edge, and q profile control scenarios in a plasma consistent with available ECH power.
- Demonstrate 6 MW ECH integrated solution
- Develop 10 MW ECH target scenario

2.4.2 Neoclassical Tearing Mode Thrust

Two principal research lines are foreseen in a three year plan: (1) studies in H–mode with sawteeth present and (2) studies in an AT mode with raised q_{min} (Fig. 15).

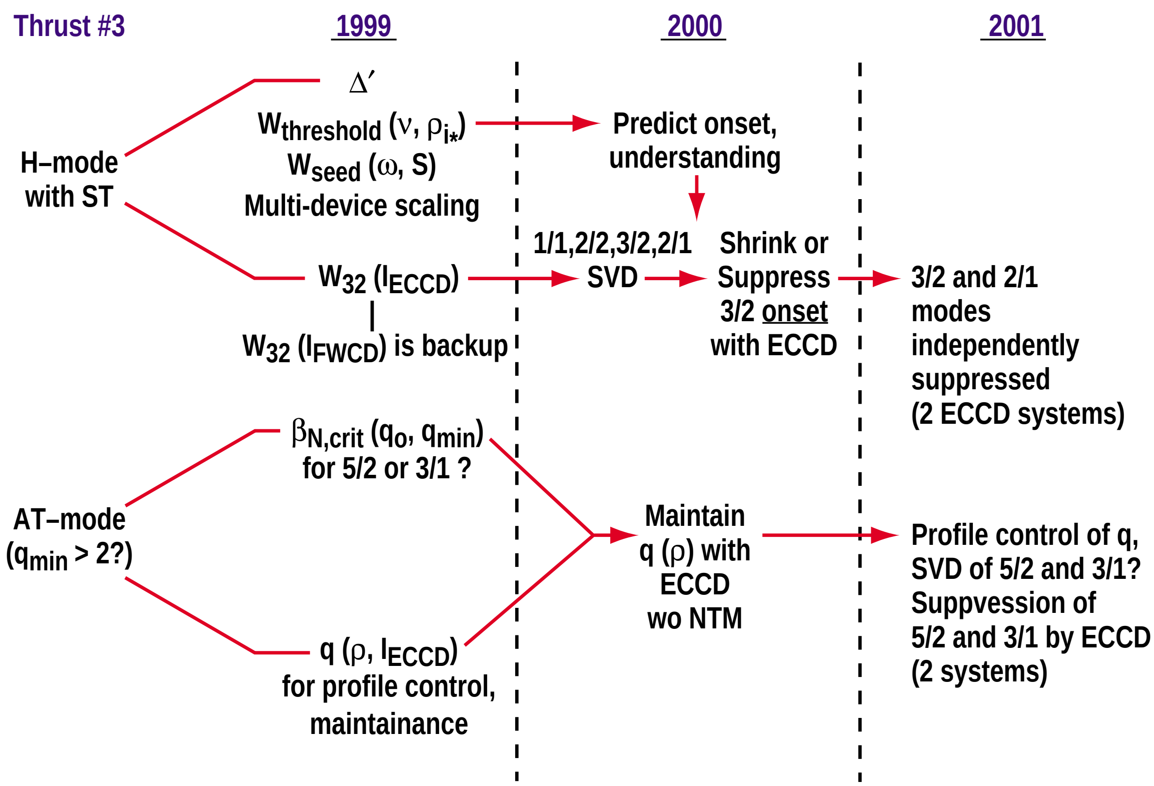


Fig. 15. NTM Research Plan.

H-mode With Sawteeth

1999

Explore standard tearing mode criteria.

Establish the NTM island width versus collisionality and ρ_*.

Establish the required seed island width versus rotation and shear.

Participate in multi-device scaling studies.

Modify the width of the 3/2 mode with ECCD.

2000

Be able to predict the onset of 1/1, 2/2, 3/2, and 2/1 modes.

Shrink the modes or prevent their onset with ECCD.

2001

Suppress the 3/2 and 2/1 modes separately using two ECCD systems.

AT Mode Line

1999

Establish the critical β_N (q_0, q_{min}) for 5/2 or 3/1 modes.

Study the variation of the q profile with ECCD.

2000

Maintain the q profile with ECCD.

2001

Control the q profile and suppress the 5/2 and 3/1 modes by 2 ECCD systems.

2.4.3 Wall Stabilization Thrust

Research Thrust 4 (Leader: G. Navratil)

Validate the model of wall stabilization and begin feedback stabilization experiments.
Overall Goal: Sustained operation at beta significantly above the no-wall limit.

1999: Validate Models of Rotational Stabilization and Initial Experiments on Feedback Control

Validate Models of Wall Stabilization in a Rotating Plasma.

- Develop physics understanding of rotational stabilization.
- Understand and develop approach to control slowing when $\beta > \beta$ (no wall).
- Dependence of RWM stability on q_{min}.
- Effect of outer region resonant surfaces and qualitative comparison to models.
- Extend wall stabilization to higher β_N regimes.

Initial Experiments on Feedback Control.

- Develop n=1 feedback control logic.
- Demonstrate improved stability with active non-axisymmetric coils, using one new 100 Hz bipolar power supply.
- Assess need for modified C–coil through 3D modeling and analysis of benchmark experiments.

Extend Lifetime of Plasma Operating Above the No-Wall Beta Limit.

2000: Validate Model for Active Control. Optimize control with six-element C–coil.

- Validate quantitative 3D model for n=1 feedback control.
- Extend the regime of improved stability with closed-loop feedback control, with higher power using three bipolar power supplies.
- Finalize design of upgraded external coil set for improved feedback control.

2001: Feedback With Improved C–coil

- Install upgraded coil set.
- Demonstrate sustained operation significantly above the no-wall beta limit in high performance AT plasmas.

Proposed Experimental Plan for 1999

1. Validate models of wall stabilization

   1.1 Physics of rotational stabilization (1 day co, 1 day counter)
   - — Vary angular momentum input
   - — Dependence on qmin and resonant surfaces
   - — Degradation in momentum confinement at high β_N.

   1.2 Develop improved target plasma (1 day)
   - — Develop path to wall stabilization in higher β_N AT regimes

2. Begin feedback stabilization experiments

   2.1 Closed loop control of resistive wall mode (3 days)
   - — Develop feedback control
   - — Demonstrate improved stability through feedback control
   - — Benchmark models of feedback control

   2.2 Closed loop control of locked modes (1 day)
   - — Test of “smart shell” on low density locked modes
   - — Extend stable operation at low density

3. TOPICAL SCIENCE AREAS — THREE YEAR VIEWS

3.1. STABILITY TOPICAL AREA GOALS (3 YEAR VIEW)

1. Advance the physics understanding of resistive wall mode stability, including the dependence on plasma rotation, wall distance, and feedback stabilization. *Develop sustained operation above the no-wall beta limit through passive or active stabilization of the resistive wall mode.*

2. Characterize the physics of edge-driven instabilities in plasmas with a large (H–mode) edge pressure gradient and associated bootstrap current. *Develop methods to avoid or reduce the impact of edge-driven instabilities through modification of the edge pressure gradient, collisionality, or shaping.*

3. Advance the physics understanding of non-ideal plasma instabilities including neoclassically driven tearing modes, sawteeth, and fast ion driven instabilities. *Develop sustained high beta operation free of sawteeth and neoclassical tearing modes, through profile control or active stabilization.*

4. Advance the understanding of disruption physics in advanced tokamak discharges and improve the viability of tokamak reactor concepts by avoiding and mitigating disruptions.
   Develop methods of mitigating runaway electron, halo currents, and disruption heat loads, and disruption prediction and avoidance using real-time identification of disruption boundaries.

3.2. CONFINEMENT AND TRANSPORT TOPICAL AREA GOALS (3 YEAR VIEW)

1. Improve the physics understanding and control of core transport barriers.
   — Detailed physics of barrier formation including generation of sheared E×B flow.
   — Techniques to control barrier location
   — More accurate comparisons with theory, including GKS theory for realistic toroidal geometry.

2. Understand the effects of electron heating on plasma rotation and transport, especially in discharges with core transport barriers.
   — Test ETG-mode hypothesis for explaining residual electron transport when ITG is stabilized.

3. Develop new tools to controlling E×B shear
   — Counter injection
   — Off-axis (vertical and tangential) pellet injection
   — RF creation of radial currents and sheared flows

4. Carry out innovative experiments to make quantitative tests of predictions of (theory based) transport models.

5. Utilize nondimensional scaling approach to define an attractive next-step device based on ELMing H-mode.
   — Determine dependence of local transport on various nondimensional parameters including T_e/T_i.

6. Investigate RI–modes, with emphasis on theory-experiment comparisons to elucidate the fundamental turbulence stabilization physics.

7. Improve comparison of experiment and theory of edge and divertor conditions needed to get H–mode.
   — Quantitatively test the new set of analytic theories developed in Europe.
   — Encourage detailed comparison of U.S. numerical work (e.g. Drake, Xu) with experimental results.
   — Determine if plasma parameters alone govern threshold or whether atomic physics (e.g. neutrals) is also important.

8. Search for means of lowering the L to H power threshold.
   — Can local pressure gradient changes induced by edge deposited pellets trigger the transition?

9. Study the H–mode edge pedestal and try to determine the key physics controlling the edge gradient and pedestal values.

3.3. DIVERTOR/EDGE PHYSICS TOPICAL AREA GOALS (3 YEAR VIEW)

The main function of the boundary plasma is to control particle and power flux at the interface between the core plasma and the material walls. The long range goal of the DIII–D divertor and scrape-off layer science program is to: (1) use state-of-the art 2-D diagnostics to identify the relevant physical processes, (2) model these processes with computational models (e.g. UEDGE), and (3) sufficiently understand the relevant physical processes in the edge plasma so that computational models can predict operation for new operating modes on existing machines and for new machine designs.

We have identified and studied the radiative divertor or “detached” mode of operation which reduces the heat and particle flux in the divertor by deuterium puffing. Intrinsic carbon radiation is a key ingredient in this mode. We plan to extend the operating regime (i.e., operation at lower core n_e) for near-term AT operation by concentrating radiation in the divertor with injected impurities such as argon. The two tools to achieve this goal are so-called “puff and pump” techniques, (deuterium injection and pumping to provide a force on impurities towards the divertor) and divertor baffling (to better control neutrals). The baffling and pumping are also important ingredients in the control of density and impurities for the core plasma. We will also investigate the role of triangularity, single-null, and double-null on both divertor and AT conditions. Substantial progress has been made in the measurement (DiMES probe) and modeling (REDEP) of erosion and redeposition in the DIII-D divertor during detached operation. These studies will be continued during the next phase of impurity radiative divertor operation. They are also important in understanding the best means to control carbon radiation in an all-carbon machine like DIII–D.

3.3.1. 2000–2001 Work

At the start of the year 2000 we will have several new boundary research tools at our disposal which will significantly enhance our ability to conduct experiments in support of the DIII–D Advanced Tokamak and Boundary Physics program goals. The private flux cryopump-baffle system will be commissioned. The graphite armor tiles in the vicinity of the new divertor system will be improved so that they follow the field lines more closely, thereby reducing the number of sharp edges that can overheat. The new capabilities apply to single-null and biased double-null configurations while allowing a side by side comparison of the closed and open configurations.

With these modifications we will have the capability of independently pumping both legs of the divertor, reducing the core neutral source in single-null by an estimated factor of six relative to an open configuration, and sustaining AT plasmas up to the device volt-second limit before reaching the tile thermal limit. New research made possible by these modifications include:

1. Impurity control by the forced flow technique (“puff and pump”) may be extended to lower density plasmas, and perhaps even to ELM-free plasmas. If

successful, the technique will be used routinely to reduce carbon concentration in AT plasmas.

2. Research to expand the volume of radiative zone by convection.
3. Study feasibility of stable fully detached plasmas.
4. Investigate stability and confinement of density controlled rectangular cross-section plasmas.
5. Isolate the effect of neutrals on edge transport barrier and L-H transition.
6. Investigate heat flux control at densities compatible with AT scenarios using a combination of mantle and divertor radiation and application of convection.

The first order of business in the year 2000 is to evaluate the new system for these applications. The actual detailed experiments will be spread out through years 2000–2001. The commissioning work includes; evaluation of the result of improved wall armor on carbon content of the plasma, optimization and control of pumping configurations, preliminary evaluation of impurity reduction by bi-directional forced flow, and neutral density decrease due to the new baffle system. These preliminary rough measurements will be followed by more detailed focused experiments in years 2000 and 2001 as described below.

The outcome of the years 2000 and 2001 AT and divertor experiments will guide us towards the future course of the divertor effort. The options in the near future are:

1. Accept the version-2000 single-null/biased double-null configuration, perhaps with a number of refinements such an inner wall bump to increase the amount of baffling (but limit the shape flexibility).

2. Proceed to a full double-null divertor configuration.

3.3.2. Details of the Year 2000 and 2001 Experiments

1. Determine conditions necessary for divertor impurity enrichment. Make extensive use of the full upper RDP.
2. Develop radiative mantle discharges.
3. Develop heat flux reduction techniques at AT-like edge conditions.
4. Determine steps required to minimize carbon influx in high performance plasmas.
5. Demonstrate predictive capability of erosion/redeposition pattern in the DIII–D tokamak.

3.3.3. 2001 Work

1. Continue to develop radiative divertor and mantle solutions compatible with the density operation needed for the (near-term) AT core plasma scenarios.

2. Attempt an integrated divertor/core demonstration as an element of the NCS scenario work.

3. Prove viability of poloidal tokamak divertor concept by demonstrating control of erosion and co-deposition.

3.4. HEATING AND CURRENT DRIVE TOPICAL AREA GOALS (3 YEAR VIEW)

1. Establish predictive capability for ECCD, including dependencies on density, temperature, Z_{eff}, geometry, power, trapping, and dc electric field.

2. Advance the physics understanding of FWCD, including effects of frequency, $n_{||}$, competing edge losses, high harmonic absorption on beam ions and thermal ions, rf-induced resonant ion tranport, wave propagation, conservation of toroidal mode number.

3. Advance the understanding of NBCD, including the effects of fast particle modes and TAE modes. This would involve development of a model of fast ion transport.

4. Understand the effects of heating of electrons and/or ions on plasma rotation and transport, particularly transport barriers.

5. Develop long pulse discharges with full noninductive current drive, including discharges with very high bootstrap fraction as a step toward transformerless operation.

6. Develop routine electron heating using the ICRF system, through fast wave and/or second harmonic hydrogen minority heating (especially at high density where beam penetration is poor). Develop minority heating for sawtooth stabilization and minority or beam ion current drive.

4. SYNOPSIS OF THE 1999 DIII–D RESEARCH PLAN

The research campaign for 1999 has been organized into six research thrusts and a broader selection of experiments in four Topical Science Areas. Significant blocks of experimental time have been allocated to the research thrusts, since these activities are aimed directly at critical objectives for the DIII–D Program and for the tokamak research program generally. Additional experimental time in the topical areas maintains the breadth and scientific depth of the DIII–D Program. Below we convey the essential content of the various research thrust and topical science experiments and their goals and anticipated and hoped for results. The research described has been allocated to 58 run days out of a possible 73 run days in the 1999 campaign. Additional detailed information can be found on the Web locations:

http://fusion.gat.com/exp/1999_1_6
/http://fusion.gat.com/exp/1999_1_6/Outline1999.pdf
http://fusion.gat.com/exp/1999_1_6/TimeAllocations.pdf
http://fusion.gat.com/meetings/bs99/

The experiment plan was put together with input and prioritization by the 1999 Research Council. Based on the "DIII–D Five-Year Program Plan 1999–2003," August 1998, GA–A22950, the council developed a three-year plan from which 1999 research thrusts were identified. A call for ideas was issued and approximately 200 ideas were presented at a community "Brainstorming Meeting" of September 22–24, 1998 which was broadcast on the internet. Based on these inputs the Research Council revised the thrusts. A Thrust 7 was added to expand the extent and extend the time duration of internal transport barriers was added and Thrust 6 to investigate high internal inductance advanced tokamak modes was allocated no run time in 1999. The council also prioritized thrust effort levels as well as allocations for scientific topical areas.

The 1999 experiment plan, summarized in Table 4, consists of efforts in six thrust areas and four topical areas. Each of the ten efforts has a responsible leader and in some cases deputy leaders.

Table 4
1999 Run Time Allocation

Research Thrusts and Topical Physics	Days
1. Regulate the edge bootstrap current and/or the edge pressure gradient to extend the duration of AT modes. (Leader — M. Wade ORNL; Deputy — B. Rice, LLNL)	8
2. Preparation of an NCS AT plasma demonstration (Leader — T. Luce, GA; Deputy — P. Politzer, GA)	7
3. Validate neoclassical tearing model and begin stabilization with ECCD (Leader — R. La Haye, GA)	6
4. Validate the model of wall stabilization and begin feedback stabilization experiments (Leader — G. Navratil, Columbia U.)	6
5. Develop the basis for choosing single versus double null and the optimum triangularity of the outermost flux surface in future machine designs. (Leader — M. Fenstermacher, LLNL; Deputies — T. Osborne, GA, and T. Petrie, GA)	6
7. Expand the spatial extent and time duration of internal transport barriers. (Leader — C. Greenfield, GA)	8
Stability physics (Leader — E. Strait, GA)	3
Confinement and transport physics (Leader — K. Burrell, GA)	7
Divertor edge physics (Leader — S. Allen, LLNL)	4
Heating and current drive physics (Leader — R. Prater, GA)	3
Contengency for hardware problems and new experiments.	15
Total	73

4.1. RESEARCH THRUSTS FOR 1999

4.1.1. Research Thrusts 1 — Regulate the Edge Bootstrap Current and/or the Edge Pressure Gradient to Extend the Duration of AT Modes — 8 Days (Leader: M. Wade, Deputy: B. Rice)

This thrust is aimed squarely at solving the principal problem in carrying forward AT regimes, the termination of the AT high performance phase by instabilities that originate in the plasma edge. DIII–D has been able to produce transiently discharges with very high values of normalized beta and confinement factor H. Discharges have been produced with a product of $\beta_N H_{98y}$ up to 10, where we have used the more modern H–mode scaling ITER–98y as the baseline for the H-factor (Fig. 16). In terms of the more familiar H_{89P} L-mode standard, values of $\beta_N H_{89P}$ about 20 have been produced transiently. While these values show great promise for the tokamak in its AT regimes, sustaining these discharges has been more difficult. The figure below shows the current situation in regard to the trade-off of discharge performance and the duration of long performance. The basic task facing the DIII–D Program is to extend the duration of the Advanced Tokamak modes to in-principle steady state. The main obstacle standing in the way is instabilities that originate in the plasma edge. This thrust is aimed at finding ways to avoid or stabilize those instabilities.

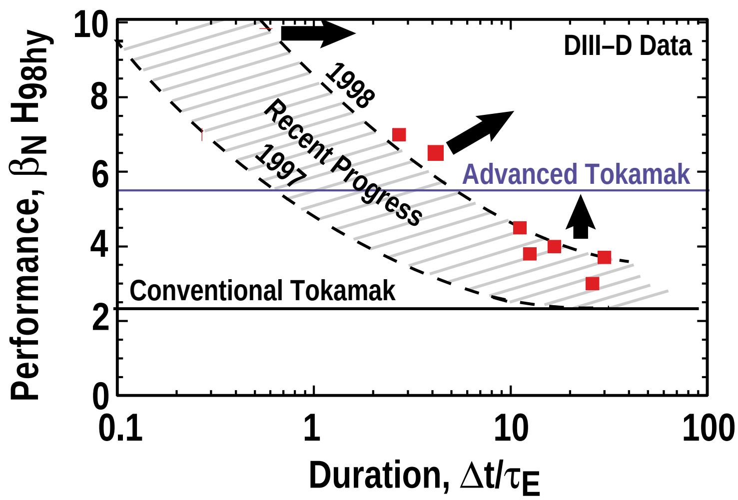


Fig. 16. Current status of trade-off of AT performance and duration.

Research on DIII–D in the last few years has resulted in a rather in-depth understanding of these edge instabilities. Basically, we believe one fundamental cause underlies the events that terminate the AT modes and that cause the familiar edge localized modes (ELMs) in H–mode. In plasmas with an H–mode edge, steep pressure gradients lead to significant bootstrap current being driven in the edge plasma. In shaped plasmas, this bootstrap current can open a radial zone of second stable access enabling the pressure gradient to increase above the calculated first ballooning limit. In such a way, a positive feedback loop is established between the edge pressure gradient and edge bootstrap current. Calculations and experimental results show that this increased current in the edge plasma reduces the mode number of the most unstable mode (to as low as 3) as the mode becomes more current gradient driven than pressure gradient driven. As a result, the radial extent of the unstable mode increases and the effect of the eventual instability on the plasma becomes more severe. While these instabilities serve as "soft" beta limits (i.e., they usually just cause a transition from an AT state to an ordinary ELMing H–mode state), the likelihood an AT mode being terminated by these edge instabilities increases with the radial extent of the second regime access zone. By closing access to second stability, this positive feedback loop between the edge pressure gradient and edge current can be stabilized, resulting in high-n ballooning modes of limited spatial extent. However, the increased edge pressure gradients allowed by second stable access also increase the edge pedestal height and the overall confinement of the whole plasma. So some compromise must be struck between having the high edge pressure gradients and bootstrap currents available in a second stable edge and incurring instabilities severe enough to terminate the AT phase. It is the purpose of this thrust to find that balance.

Two approaches will be explored this year. The first approach will suppress second stable access in the edge by choice of plasma shaping and will seek to build a high quality transport barrier inside the sufficiently stable rapidly ELMing first regime edge plasma. Previous experiments have already shown that second regime access can be eliminated by either low or high squareness shaping of the edge. There is also the suggestion this year that a localized bump in the outer flux surface at the midplane might also close second stable access. With one of these shaping approaches, the experiment will seek to develop a high quality transport barrier further inside the plasma.

The second approach will seek to retain the advantage in H-factor that results from second stable access but will seek to prevent or limit the consequences of the edge instabilities. Two viewpoints are possible. The one to be pursued as top priority will be to try to control the growth of the edge pressure gradient (and therefore the resulting bootstrap current) by such means as impurity mantle radiation and deuterium puffing in order to keep the pressure gradient from reaching the unstable limit. Another general approach to reducing the edge bootstrap current will be to increase edge collisionality. The second viewpoint is to use means such as edge ECH to decrease the edge collisionality to **increase** the ratio of bootstrap current to pressure gradient to maximally open the second stable window without filling it with an excessive pressure gradient. These research lines are higher risk but higher payoff since they seek to retain for the AT performance some fraction of the gain available from operating with edge pressure gradients above the first regime limit.

4.1.2. Research Thrust 2 — Preparation for an NCS AT Plasma Demonstration — 7 Days (Leader: T. Luce, Deputy: P. Politzer)

The negative central shear regime is the primary AT scenario being pursued by DIII–D in its long term development of the AT potential. The key to exhibiting this scenario in in-principle steady-state is the maintenance of a hollow current profile using ECCD to prevent diffusive dissipation of the off-axis current peak. Over the next three years, the EC power on DIII–D will be increased steadily from the present three-gyrotron system to a six-gyrotron (1 MW per unit) system. More importantly, the four newest gyrotrons will be equipped with diamond windows to enable longer (5 s) pulses. Two of the present gyrotrons have pulse length limitations of 1.3 and 2.0 s respectively. Set against this background of a gradual buildup in the necessary hardware, this thrust is aimed at a demonstration in 2001 of a sustained, high performance NCS scenario. The work this year is preparatory toward this goal. This thrust proposes to perform NCS target discharge scenario development using transient techniques, to commission the new EC system, and to validate ECCD for eventual use in the NCS scenario.

Modeling studies have allowed us to construct parameter regimes for the NCS scenarios. These parameter regimes are summarized in Table 5.

The scenario for the year 2000 is our near term goal using the anticipated four gyrotrons. Owing to the lower available power, this scenario is placed at lower B_T and I_p than the year 2001 scenario that anticipates using a six gyrotron system. We have an additional scenario (not presented here) for full field (B_T = 2 T and I_p = 2 MA) employing an eventual 10 gyrotrons. The values of β_N and H_{89P} that are the targets of these scenarios will be very significant intermediate accomplishments in our AT program. The main anticipated goal of the efforts in this thrust in CY99 is to form the plasma described in the year 2000 column above by the kinds of transient means (current

Table 5
NCS Scenarios Using ECCD

Gyrotrons Available (year)	4 Tubes (2000)	6 Tubes (2001)
P_{EC} (MW)	2.3 into plasma (4 source)	4.5 into plasma (6 source)
P_{FW} (MW)	3.6	3.6
P_{NBI} (MW)	4.1	3.8
I_p (MA)	1.0	1.3
I_{Boot} (MA)	0.65	0.9
I_{ECCD} (MA)	0.15	0.2
B_T (T)	1.6	1.75
β (%)	4.0	6.3
β_N	4.0	5.3
H_{89P}	2.8	3.5
n (10^{20} m^{-3})	0.32	0.5
n/n_G	0.3	0.4
T_i (0) (keV)	6	8
T_e (0) (keV)	8	9

ramping, early NBI, shaping control) that we have been using to form NCS and other AT plasmas. From those plasmas, we can obtain a benchmark of experimental data on stability and radial transport coefficients that will enable us to further hone our discharge simulations for the effort in 2000. A hoped for goal in 1999 is to get far enough to attempt to show extension of the time duration of this transiently formed scenario by applying the limited pulse duration EC power available. A positive result would show the promise of work planned in 2000 and 2001.

The work in this area this year will commission the new diamond window gyrotron and build a new steerable EC launcher. A basic preparatory experiment will be a validation of ECCD in ELMing H–mode plasmas; the issue here is the degree of achievable power deposition localization in the presence of ELM effects on ray trajectories. This thrust will seek to the extent possible to use what is learned from Thrust #1 to limit the effects of edge instabilities. From the table, these scenarios involve rather low densities which require pumping. The basis for such low densities was set last year using either the lower (low triangularity plasma) divertor pump or the upper (high triangularity plasma) pump. The experiment will seek to assess the effects of q_{min} and $q_0 - q_{min}$ on stability and transport barriers, confinement when T_e approaches T_i, and issues about impurity accumulation which is always a concern at low densities.

4.1.3. Research Thrust 3 — Validate Neoclassical Tearing Model and Begin Stabilization With ECCD — 6 Days (Leader: R. La Haye)

After the edge instabilities that are the subject of Thrust 1, the next largest immediate stability concern for the AT work is the neoclassical tearing modes (NTMs). These modes have been seen to limit the performance in all our approaches to AT plasmas. Even in plasmas in which q_{min} has been raised above 2, NTMs have been observed. The purpose of this thrust is to gain further physics understanding of the neo-classical tearing modes and develop means of avoiding or stabilizing them.

This thrust has four highest priority tasks: use of unmodulated ECCD to stabilize NTMs, studies of the NTM critical β versus q-profile, studies of the ρ_{i*} and S scaling of the threshold β_N, and studies of classical tearing mode stability. In addition, one day is planned for the verification of ECCD in the configuration to be used for the NTM stabilization effort.

Two principal research lines are foreseen in a three year plan: (1) studies in H–mode with sawteeth present and (2) studies in an AT mode with raised q_{min}.

H-mode With Sawteeth

The diagnostic set now available on DIII–D, in particular the MSE diagnostic for measuring the current profile, affords a scientific opportunity not previously available to measure all the quantities involved in classical tearing mode theory to verify that theory. Some effort will be devoted to this basic science verification. Work in 1999 will continue on our ongoing collaboration with JET, ASDEX Upgrade, JT–60U, and Alcator C–Mod on the scaling of NTMs. The dependence of the NTM island width on collisionality and ρ_* will be studied. The relationship of the required seed island width and the plasma rotation and shear will be explored. An active attempt will be made to modify the width of the 3/2 mode with ECCD. This work will lead to efforts in 2000 toward predicting the onset of 1/1, 2/2, 3/2, and 2/1 modes and to shrink the modes or prevent their onset with ECCD. In the year 2001, we will be able to use two separate ECCD systems for suppressing the 3/2 and 2/1 modes simultaneously.

AT-mode Line

In 1999, we will establish the critical β_N as a function of (q_0, q_{min}) for 5/2 or 3/1 modes.

We will hope to find that the NTM problem becomes less severe with higher q_{min}. We will also begin to study the variation of the q profile that can be achieved with ECCD. In the year 2000, we expect to have enough long pulse ECCD power to maintain a more stable q profile. In the year 2001, we will seek to control the q profile and suppress the 5/2 and 3/1 modes by 2 ECCD systems.

Principal Goals for 1999

1. Show best regime for avoiding the NTM at high β_NH and duration. Develop input for ECCD control.

2. Show unmodulated ECCD shrinks NTM islands.

4.1.4. Research Thrust 4 — Validate the Model of Wall Stabilization and Begin Feedback Stabilization Experiments — 6 Days (Leader: G. Navratil)

The AT Program on DIII–D has shown from theory calculations that sustaining β_N greater than four requires stabilization by a nearby conducting wall. The two key elements of wall stabilization are the degree to which a conducting wall can look “superconducting” if the plasma rotates past the wall and the provision of suitable non-axisymmetric feedback to suppress the modes that grow locked to the wall. Recent theory work has suggested that even with a rotating plasma, a “resistive wall mode” can arise that is locked in position and does not rotate with the plasma. Experiments to date have provided support for both the existence of the resistive wall mode and transient evidence for the ability to operate plasmas above the no-wall beta limit as long as wall stabilization remains effective. Over the next three years, DIII–D plans to implement a set of non-axisymmetric coils to provide feedback stabilization of resistive wall modes. This thrust area has two main objectives:

1. Advance the physics understanding of resistive wall mode stability, including the dependence on plasma rotation, wall/plasma distance, and active feedback stabilization.
2. Develop sustained operation above the no-wall beta limit through passive or active stabilization of the resistive wall mode.

Specific goals for 1999 are:

1. Develop a physics understanding of rotational stabilization, including:
   - Reason for high (~6 kHz) threshold in some cases
   - Developing an approach to control rotational slowing when $\beta > \beta_{\text{no wall}}$.
   - The q_{min} dependence of RWM stability
   - The effect of resonant surfaces and qualitative comparison to rotation models.
   - Extend wall stabilization to higher β_N regimes
2. Demonstrate improved stability with closed loop active feedback control:
   - Developing the n=1 feedback control logic.
   - Demonstrate improved stability with active non-axisymmetric coils, using new bipolar power supply
   - Higher β and/or longer lifetime with $\beta > \beta_{\text{no-wall}}$
   - Benchmark models of feedback control
3. Develop quantitative 3-D model for active mode control using the VALEN Code
   - Assess need for modified C–coil through 3D modeling and analysis of benchmark experiments
   - Will provide the design basis for extended C–coil

The most important result we will reach for this year (and it is a stretch) is the demonstration of a time extension using active feedback control of a plasma operating above the no-wall beta limit. Code predictions indicate we could extend a $\beta_N = 2$ no-wall plasma to $\beta_N = 2.3$ with the existing six-element coil system and to $\beta_N = 2.6$ with the proposed 18-element coil system. Starting from a higher performance plasma, the corresponding improvement sequence would be $\beta_N = 3$ goes to 3.5 with six coils and 3.9 with the proposed 18-coil set. That demonstration will prove the principle of wall stabilization and will open up for exploration the extremely valuable terrain that lies above the Troyon limit ($\beta_N < 3$), up to perhaps $\beta_N \sim 6$.

4.1.5. Research Thrust 5 — Develop the Basis for Choosing Single Versus Double-Null and the Optimum Triangularity of the Outermost Flux Surface in Future Machine Designs — 6 Days (Leader: M. Fenstermacher, Deputies: T. Osborne and T. Petrie)

This thrust is related to Thrust 1 in that it focuses on the edge plasma. In contrast to Thrust 1 which will explore active interventions in the plasma edge, this thrust seeks to accumulate a large body of detailed systematic measurements aimed at building a deeper understanding of the physics of the coupled regions just inside the separatrix (the H–mode pedestal region) and the region just outside the separatrix (the SOL and divertor). More specifically, this thrust will implement a set of systematic data scans to obtain detailed edge pedestal, divertor information, and other plasma performance measures versus the triangularity and the distance between the separatrices of a double-null. This information is expected to answer key questions of future machine design related to the best shape of the outermost flux surface, focusing on an edge physics point of view.

H-mode Pedestal Studies

The interaction of ELM stability, L-H transition physics, and density limit physics and their relation to the H–mode pedestal parameters can be illustrated in a plot of the pedestal temperature versus pedestal density as first done by the ASDEX-U group. The phase space of a variety of tokamaks in these coordinates is similar and qualitatively illustrated in Fig. 17. Along the Type I ELM boundary, as the pedestal density is raised, the temperature is reduced keeping the pedestal pressure and H more or less fixed. Above a critical density: (1) the plasma can return directly to L–mode (typically at an ELM), (2) a MARFE can occur often followed by H-L or disruption, (3) Type III ELMs can occur with pressure gradient decreasing with increasing density; further increase in density in the Type III regime can lead to H-L or MARFE. A degradation in edge confinement at high density may be associated with enhanced turbulence approaching the H-L transition.

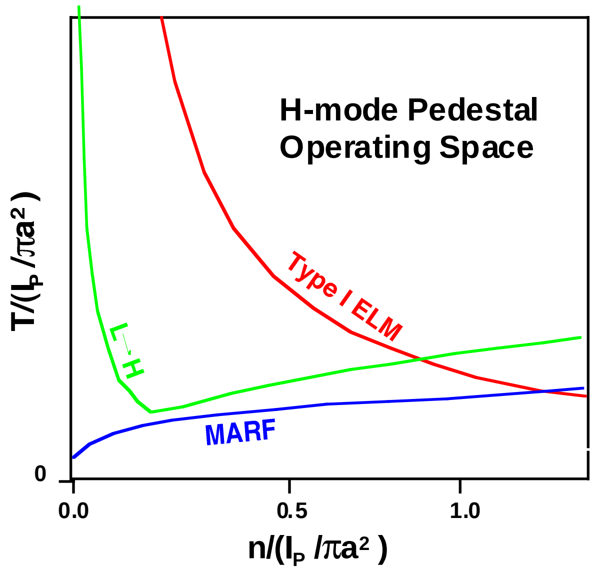


Fig. 17. Boundary curves for transition to different regimes in the H–mode edge.

In the area of H–mode pedestal physics, in this thrust we hope to accomplish the following:

1. **H–mode Transport Barrier Width**: (a) extend the range of density in the Type I regime to test the temperature dependence of the width, (b) test the Hinton-Staebler theory that the width is set by the extent of the edge particle source, (c) determine the dependence of the width on triangularity, (d) compare the widths from the edge profiles with widths determined from the fluctuation suppression zone, (d) determine the scaling with Type III ELMs.

2. **Type I ELM Stability**: (a) test Miller second stability access theory with discharges at low q and triangularity, (b) obtain data at different q, shape, and collisionality (edge bootstrap current) for comparison with low n kink-ballooning theory, (c) test Rogers, Drake, Zeiler theory relating stability to α_{crit}, (d) test high n peeling mode theory relative to low n kink-ballooning by varying plasma wall distance.

3. **Type III ELM Stability**: (a) determine the scaling of the onset conditions and critical pressure gradient for Type III ELMs and compare to theories.

4. **Confinement Degradation at High Density**: (a) determine if τ_E equivalent to the Type I ELM regime can be obtained with Type III ELMs at high density, (b) determine the effects of density and triangularity on ELM energy loss, (c) observe the degradation of energy confinement prior to H-L and compare to theories, (d) obtain data regarding the structure of the turbulence when the confinement degrades.

5. **H–L Transition**: (a) determine the threshold conditions for H-L back transition at high density and compare to theory, (b) determine the scaling of the low density L-H (Type IV) ELM boundary and compare to theories.
6. **MARFE Boundary**: (a) determine the scaling of the MARFE boundary and compare to theory.

SOL and Divertor Plasma Studies

In the same scans of triangularity, q, density, and distance between separatrices in which the pedestal is studied, the variation of the divertor plasmas obtained will also be studied.

Triangularity is expected to vary the amplitude of the ELMs introduced into the SOL and their effect on the divertor plasma and divertor targets. An important part of the optimization trade-off in the edge plasma is to balance the desire for a large edge pedestal for confinement with the restriction on ELM amplitude that can be tolerated from a divertor plate erosion point of view. This thrust should supply the information base to make such a trade-off.

The **distance between the separatrices (drsep)** of a double-null is a key parameter. Future machine designs have evolved to a position in which two X–points are in the chamber and data is needed to decide how close those separatrices should be. When they are many scrape-off layer widths apart, the plasma should be essentially single-null. The opposite limit, true double-null, is obtained when these two separatrices coincide. In our DIII–D experiments, drsep will be scanned from negative to positive by vertically shifting the plasma. When the sign of drsep changes, the direction of the grad-B drift at the last closed flux surface also changes. Hence we will also be studying the effect of the grad-B drift direction on the divertor and pedestal physics. Varying drsep is expected to vary the neutral densities that can result at the last closed flux surface and so perhaps influence H–mode properties. Divertor detachment thresholds are also expected to change. Perhaps most importantly, we have observed that true double-null plasmas are refueled in a qualitatively different manner than single-null plasmas. Owing to low power flow on the relatively isolated inboard side of a double-null, the inboard legs of a double-null are always detached and that region is transparent to neutral gas. Hence double-nulls refuel from the inner side, even though the gas source is the outer attached divertor leg. In single-nulls, the plasma refuels mainly through the X–point region or perhaps from the outboard side generally. We will investigate the qualitative difference that this refueling pattern makes in divertor and core plasma performance in double-null.

The **divertor depth, the distance from the X–point to the divertor target,** is another key parameter. We will investigate how much divertor volume needs to be used in single-null vs. double-null configurations to achieve comparable divertor performance in terms of heat flux reduction, neutral compression, impurity entrainment, etc. From a design point of view, if one used double-null plasmas, one would like the divertor on each end of the double-null to be one-half the depth of the single divertor. The question is whether the divertor physics supports this trade-off.

4.1.6. Research Thrust 6 — Preparation for a High ℓ_i AT Plasma Demonstration (Leader: J. Ferron)

This thrust proposed four main areas of effort: to maintain the high ℓ_i current profile and the associated improved confinement, sawtooth stabilization studies, investigation of $\beta_N > 5$, and production of model discharges with scenario target parameters. Regrettably, owing to the limited run time, we are unable to allocate any run time to this thrust this year.

4.1.7. Research Thrust 7 — Expand the Spatial Extent and Time Duration of Internal Transport Barriers — 8 Days (Leader: C. Greenfield)

The goal of Thrust 7 is to expand the operating space of discharges with an internal transport barrier.

— Expand the spatial extent of the ITB
— Sustain the ITB for full duration of NBI heating
— Extend the operating space toward $T_e/T_i \sim 1$.

New tools will be applied to the problem of transport barrier control.

Neutral beam counter-injection will increase q_0 and allow variation of the alignment of the grad-*p* and rotation terms in the radial force balance.

Pellet injection will provide on- and off-axis sources of particles. Using a strong density gradient to drive ITB formation may allow an ITB with $T_e \sim T_i$.

We will continue use of **AE reduction techniques* begun in 1998 in order to allow transport analysis.

Central Thomson scattering will be a key diagnostic to observe internal particle transport barriers and will be essential for pellet-fueled experiments where the ITB should appear primarily in the density profile.

ITB Expansion and Sustainment

Based on experiments from 1998 where ITBs were sustained for full NBI duration at low current and power.

- Alfvén eigenmode (*AE) fast-ion loss prevented successful transport analysis.
- The plasma had to be limited for a period during the early evolution to avoid H-modes which were perhaps triggered by *AE redistribution of fast-ions.

The Performance of these discharges could not be reproduced after eliminating *AE activity by decreasing the NBI voltage and favoring more perpendicular sources.

- Central NBCD rapidly reduced q_0 to 1; the NCS phase could not be maintained.

Neutral beam *counter-injection* coupled with *AE elimination may allow access to improved performance.

- Central NBCD will drive q_0 higher rather than lower.
- Elimination of *AE induced L–H transitions may open up a previously inaccessible region of parameter space.

Increased neutral beam power should produce stronger negative central shear which should push out the radius of the minimum q and produce a transport barrier with a larger radius. The desired state is a JT–60U-like transport barrier at $\rho \sim 0.8$. The gradual increase of neutral beam power and plasma current is expected to allow production and sustainment of a high-performance state. The counter current drive effect should be replaced later in the DIII–D program (FWCD + ECCD), making this aspect of the experiment a demonstration of a possible future scenario.

Counter-injection also offers a rare opportunity to look at some of the E×B shear physics involved in the ITB.

- Pressure gradient and toroidal rotation terms in the radial force balance equation are in opposite directions in plasmas with co-injection. The strong pressure gradient can cancel the effect of rotational shear.

- Plasmas with counter-injection will have very different profiles of E_r and turbulence shearing rate $\omega_{E\times B}$ than those that we are accustomed to.

The possibility of different alignment of the force balance terms may make future counter-injection desirable. This experiment will also provide valuable data for a possible future decision on turning around one neutral beamline.

Internal Transport Barriers with T_e ~ T_i and Barriers in the Density Channel

We will make the first attempt at internal transport barriers with $T_e \sim T_i$ in DIII–D.

- Central pellet injection will build up a density gradient.
- The pressure gradient term in the radial force balance generates large $E \times B$ shear.
- With sufficiently large density gradient, ITB generation should be possible even with $T_e \sim T_i$.

Stand-Alone Experiment

We have chosen one stand-alone experiment that will not be part of the counter-injection work but is part of this thrust.

Expanding $\rho_{q_{min}}$ With Fast I_p Ramp and High Power NBI

This work continues 1998 experiments to expand $\rho_{q_{min}}$ by initiating a discharge with a rapid current ramp. The 1998 experiments were successful in producing a desirable current profile, but had severe MHD instabilities when reduction in the NBI power allowed rapid reduction of q_{min} through several low-order rational values. The proposed experiment will include steps to avoid this MHD.

- Longer early neutral beam pulse.
- Reduce or eliminate second current ramp.

This experiment is another route toward an ITB with large spatial extent.

4.2. PHYSICS TOPICAL AREAS

4.2.1. Stability — 3 Days (Leader: E. Strait)

Stability Topical Area Goals (3 Year View)

1. Advance the physics understanding of resistive wall mode stability, including the dependence on plasma rotation, wall distance, and feedback stabilization. *Develop sustained operation above the no-wall beta limit through passive or active stabilization of the resistive wall mode.*

2. Characterize the physics of edge-driven instabilities in plasmas with a large (H–mode) edge pressure gradient and associated bootstrap current. *Develop methods to avoid or reduce the impact of edge-driven instabilities through modification of the edge pressure gradient, collisionality, or shaping.*

3. Advance the physics understanding of non-ideal plasma instabilities including neoclassically driven tearing modes, sawteeth, and fast ion driven instabilities. *Develop sustained high beta operation free of neoclassical tearing modes, through profile control or active stabilization.*

4. Advance the understanding of disruption physics in advanced tokamak discharges.
Develop methods of mitigating halo currents and disruption heat loads, and disruption avoidance using real-time identification of disruption boundaries.

The first three major long term goals of this topical area are being implemented as Thrusts 1, 3, and 4 this year. In addition to this activity, this topical area will contain one experiment on basic sawtooth physics. Like the experiment on classical tearing modes in Thrust 3, this experiment will also make use of the excellent measurement capability to try to settle important basic questions in the stability of the plasma near the magnetic axis. The interior pressure profile and especially the q profile measurements afforded by the MSE system enable a detailed examination of Mercier stability and other criteria and a detailed investigation of the Kadomtsev reconnection vs. Taylor relaxation models for the sawtooth. The q profile evolution and energy flow in sawteeth will be examined, as well as the interaction of fast ions.

If any contingency time can be devoted to this topical area, the next experiments to be done would be to continue our successful disruption mitigation work. The work would use killer pellets and massive gas puffing to mitigate disruptions. The highest priority activity would be to use spectroscopic techniques developed last year to measure the electron temperature in the plasma during the disrupting phase. The work would also obtain essential data for prediction of halo currents, validate models of runaway generation, improve the physics understanding of gas puff penetration, and provide data toward a decision on whether liquid jets are desirable.

This topical area also has an experiment in the counter-injection campaign. The motivation for this experiment is that the resistive wall mode is observed to become unstable when the rotation in the outer part of the plasma decreases below a threshold value. The resistive wall mode in some cases rotates in the electron drift direction, suggesting that the relevant rotation may be the E×B flow, not the fluid flow.

With counter-injection, the contributions to the radial electric field from rotation and grad(p) have the same sign, eliminating the $E_r = 0$ region typically found near the edge of H–mode plasmas. This experiment will test theories of wall stabilization by varying the rotation profile and potentially increase the duration and maximum beta of wall-stabilized plasmas.

4.2.2. Confinement and Transport — 7 Days (Leader: K. Burrell)

This topical area has experiments under various headings:

In the area of **Core Transport Barrier Physics and Control,** the experiment is to explore a new use of ICH for control of the core plasma radial electric field. The idea is to create radial current flows by moving ions from one flux surface to another through ICH. This experiment follows recent lines of thought by C.S. Chang on the toroidal rotation generation in Alcator C–Mod. This experiment will probably be done with counter beam injection.

In the area of **H-mode Physics,** considerable work will occur in Thrust 5 on the edge pedestal. In addition an experiment is planned to investigate the importance of electron versus ion heat flux for the L-H transition, the H–mode critical temperature characterization, electrostatic Reynolds stress, and the dependence of pedestal parameters on neutral density.

In the area of **Nondimensional Transport** Studies, the experiment will be the experiment in the counter injection campaign to reproduce the ITER demonstration ρ^* scan with counter injection and to vary momentum input by replacing beams with EC + FW. The results would be compared to comparible co-NBI shots to determine the effect of flow shear on the ρ_* scaling work.

In the area of **Tests of Transport Models,** an experiment will be done to demonstrate the existence of a heat pinch with outside launch second-harmonic ECH and to determine if the heat pinch is dependent upon the sign of the magnetic shear as predicted. The inward transport effect seen with the 60 GHz system remains a severe challenge to the theoretical community. One remaining mechanism could explain the observed profiles without requiring inward transport: the conversion of the fraction of power which is not absorbed at the resonance to electron Bernstein waves at the upper hybrid layer. This mode conversion is not possible with second harmonic launch. The superior diagnostic set now available and the higher power densities possible with the 110 GHz ECH system could provide clear evidence of the mechanism responsible for the inward transport. Furthermore, the theoretical heat pinch model of coupled transport between Grad-J and Grad-T can be tested by comparing the non-diffusive electron transport for positive and negative shear plasmas.

A second **Tests of Transport Models** experiment is planned to provide tests of turbulence simulations, tests of transport models with modulated ECH, a test of turbulent transport simulations to validate predictive capabilities, and a demonstration of marginal stability in the electrons (L–mode part only).

In the area of **Impurity Effects on transport,** we will investigate the fundamental physics of the effect of impurity injection on turbulence and transport. Local measurements will be compared with GKS predictions.

We will look for ETG vs. ITG/TEM, and the role of E×B therein. These studies will use the impurities neon, argon and krypton.

In the area of **Fundamental Turbulence Studies,** we will attempt a definitive ITG mode identification by making a detailed comparison of theoretically predicted ITG mode signatures with experimental measurements. Active involvement of theorists working aon gyrokinetic codes is needed to get the best prediction of these key signatures.

4.2.3. Divertor/Edge Physics — 4 Days (Leader: S. Allen)

Divertor/Edge Physics Objectives FY99

A. **Physics of Flow in the Divertor/Edge Plasma and Its Use to Enrich Impurities in the Divertor.** The major effort in this topical area in 1999 will be devoted to studies of plasma flows in the SOL and divertor, continuing work that became a major topic of last year's campaign. The relative role of convection and conduction in carrying the heat through the divertor region will be revisited. Experiments will probe the relative importance of the grad-T terms and the friction force in driving impurities out of or into the divertor region. These studies will use forced flow or “puff-n-pump” techniques to try to get at the basic physics of impurity transport in the background fuel plasma. The much improved diagnostic capability to measure flows in the divertor region will be used to measure the flows driven by the puff-n-pump technique.

   Program: Continue the studies of parallel flow in the divertor, begin studies of poloidal flow, study the effects of geometry and biasing on flow. Use these flows to enhance impurity radiation in the divertor, reduce peak heat flux on the strike plate, and maintain a clean core plasma.

   Technique: continue the Doppler shift spectroscopy and the Mach probe measurements, investigate higher and lower x-point, use biasing of the ADP ring to alter flows. Use the newly developed 2D analysis tools to investigate the sources under various flow conditions. Continue “puff and pump” studies using new information learned.

B. **Develop an Understanding of the Relative Importance of the Various Possible Sources of Carbon to Core Contamination and Divertor Radiation.**

A smaller effort will go into a continued effort to find the source of carbon for the core plasma by using the DiMES surface station to make further surface erosion measurements. We have so far measured the erosion at the outer strike-point and in the private flux region and failed to find sufficient erosion in detached plasmas to explain the carbon content in the core plasma. There is good accountability of outer strike-point erosion and core carbon in attached plasmas, ELM-free plasmas.

Program: Continue the study of the basic processes leading to erosion and redeposition on the divertor walls and strike plates during partially detached operation.

Technique: Use spectroscopy and probe data, DiMES data, operation in helium plasmas, and modeling to develop a consistent model of carbon source strength and transport to the core plasma. Control sources if possible.

C. **Determine the Role of Edge q in the Degradation of Confinement at High Density.** DIII–D has observed good confinement as density approaches the Greenwald limit, while JET and ASDEX-Upgrade observed considerable confinement degradation at high density. It has been postulated that differences in the typical q_{95} may be responsible for the difference.

Program: Study edge plasma properties and core confinement as a function of line averaged density at several values of q_{95} ranging from about 3 to about 5.

Technique: Use the SOL data from Thomson and CER and the divertor diagnostics to evaluate the edge pressure pedestal and divertor MARFE formation as a function of density and q_{95}.

D. **Determine the Role of Divertor and Plasma Shape in Divertor and Core Plasma Operation (Thrust 5).** Determine the role of single-null/double-null and open vs. closed divertors on ELM behavior, particle and power control, edge plasma properties such as the pedestal width and height, and core performance. This can include items from the “handshake across the separatrix”, i.e., the influence of the edge plasma on the core parameters.

Experiments for 1999 in Priority Order

1. Understanding flows and the physics “near the X–point” 2.5 days.
   - Missing data for more complete flow picture — both attached and detached.
   - Can we affect the flow with either puff and pump or biasing?
   - E×B Drifts (forward and reversed toroidal field).

2. The source and control of carbon 0.5 days.
   — Where is there net erosion in detached plasmas (not observed yet)?
   — Is the divertor strike point, the private flux region, or the whole chamber wall the main carbon source?

3. Degradation of confinement compared to other machines at a high fraction of the Greenwald density (e.g., is $q >$ or $q < 3.5$ the important parameter?) 1 day.

4. Radiative divertor operation with decreased core density, 1 day contingency
   — Can we use puff and pump, closed divertor, and impurities to obtain heat flux reduction at reduced core density?

4.2.4. Heating and Current Drive — 3 Days (Leader: R. Prater)

The two highest priority activities in this topical area, the ECCD physics study and the combined counter-NBCD and counter-ECCD activities, are scheduled in Thrust 3 and the counter campaign respectively. The next priorities for this area are:

Optimization of $n_{parallel}$ FWCD
High bootstrap fraction discharges
ICRF at 2nd and 4th harmonic
Outer coil ramp-up
High harmonic FWCD

Three-year goals:

1. Establish predictive capability for ECCD, including dependencies on density, temperature, Z_{eff}, geometry, power, trapping, and dc electric field. Determine the effect of H–mode and ELMs (in Thrusts 2 and 3).

2. Complete the physics understanding of FWCD, including effects of frequency, $n_{\|}$, competing edge losses, high harmonic cyclotron absorption on beam ions and thermal ions, rf-induced resonant ion transport, wave propagation, conservation of toroidal mode number.

3. Complete the understanding of NBCD, including the effects of fast particle modes and TAE modes.

4. Develop long pulse discharges with full noninductive current drive, including discharges with very high bootstrap fraction as a step toward transformerless operation.

5. Develop routine electron heating using the ICRF system, through fast wave and/or second harmonic hydrogen minority heating (especially at high density where beam penetration is poor.)

6. Develop minority heating for sawtooth stabilization and minority or beam ion current drive.

Prioritized List of Experiments for 1999

1. **Combined Counter-NBCD and Counter-ECCD.** Essential to understand NBCD for development of scenarios. The actual profile of NBCD appears to be broader than predicted. Counter NBCD is needed to separate NBCD from bootstrap. Counter ECCD gives information on effect of dc electric field, improves the determination of driven current profile, and may lead to very high driven current.

2. **Optimization of $n_{||}$ FWCD.** Verify FWCD efficiency proportional to $n_{||}^{-2}$ and demonstrate expected strong antenna loading. Test FWCD at highest T_e using ECH.

3. **ICRF at 2nd, 3rd, and 4th Harmonics.** Need to compare absorption at 2nd, 3rd, and 4th harmonic of same species to determine the significance of residual thermal hydrogen on 4 th harmonic deuterium scenarios. Need data on absorption by beam ions to test computational models.

If contingency time becomes available, the first experiment to be added to the list is:

4. **High Bootstrap Fraction Discharges.** Develop understanding of bootstrap current profile alignment and evolution (stability) of fully noninductive plasmas.

4.3. COUNTER-INJECTION CAMPAIGN

DIII–D plans a two week counter-injection campaign in 1999. While this is not a thrust on its own, because the counter-injection operation is a large undertaking, we keep visible the entire package of work that will be done in the counter campaign. The main user of counter-injection is in Thrust 7 for transport barrier control as described above. In addition to that work, there are four other one day experiments seeking counter-injection time. Those experiments are described in more detail in the thrusts or topical areas from which they originate. Those experiments are:

1. Two experiments from the Confinement Topical. The first is to investigate whether the core of DIII–D internal transport barrier discharges behaves like the

predictions of self-organized criticality by enabling a radial location of zero electric field. The second is to examine the possible effect of ICRH-induced radial current on plasma E_r and rotation.

2. An experiment to try to sort out the role of plasma rotation and the resulting electric field effects in our dimensionless parameter scaling studies.

3. An experiment from the wall stabilization Thrust 4 to use counter injection to probe the role of toroidal rotation in wall stabilization.

4. An experiment from the heating and current drive topical area to better pin down local ECCD by the use of counter current drive comparison shots.