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DIAGNOSTICS FOR ADVANCED TOKAMAK RESEARCH

by
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ABSTRACT

Advanced tokamak research seeks to find the ultimate potential of the tokamak as a magnetic confinement system. Achieving this potential involves optimizing the plasma cross-sectional shape, current density, and pressure profiles for stability to MHD modes while simultaneously controlling the current density, pressure and radial electric field profiles to minimize the cross field transport of plasma energy. In its ultimate, steady-state incarnation, the advanced tokamak also requires pressure profiles that have been adjusted to achieve the maximum possible bootstrap current, subject to the constraints of MHD stability. This simultaneous, nonlinear optimization of shape, current, pressure, and electric field profiles to meet multiple goals is a grand challenge to plasma physics. To keep the plasma at peak performance, active feedback control will almost certainly be required. Diagnostic measurements play a crucial role in advanced tokamak research both for developing the scientific understanding underlying the optimization and for serving as sensors for real time feedback control. One outstanding example of this is the way motional Stark effect (MSE) measurements of the internal magnetic field revolutionized work on current profile shaping. Improved diagnostic measurements are essential in testing theories which must be validated in order to apply advanced tokamak results to next step devices.

I. INTRODUCTION

Advanced tokamak research seeks to find the ultimate potential of the tokamak as a magnetic confinement system.[1–3] One facet of this problem is increasing fusion power density by increasing the plasma pressure. Achieving this involves optimizing the plasma cross-sectional shape, current density and pressure profile shapes for stability to MHD modes. A second facet is reducing cross field energy transport in order to produce the required pressure in a smaller device. This requires controlling the shape, current density, pressure and radial electric field profiles. Transport control is intimately tied up with the pressure profile control needed for MHD stability. Finally, in its ultimate, steady-state incarnation, a third facet of the advanced tokamak optimization requires current and pressure profile shapes that have been adjusted to achieve the maximum possible self-generated (bootstrap) toroidal current, subject to the constraints of MHD stability.[4] To keep the plasma at the operating point, active feedback control will almost certainly be required. This simultaneous, nonlinear optimization of shape, current, pressure and electric field profiles to meet multiple goals is a grand challenge to plasma physics.

Diagnostic measurements play a crucial role in advanced tokamak research. One outstanding example of this is the way motional Stark effect (MSE) measurements of the internal magnetic field revolutionized work on current profile shaping.[5–8] Improved diagnostic measurements are essential in testing theories which must be validated in order to apply advanced tokamak results to next step devices. They are also crucial as sensors in the real time feedback loops.

In order to optimize MHD stability, the key issues to confront include (1) kink stability to resistive wall modes, (2) stability to neoclassical tearing modes and (3) stability of the edge localized modes (ELMs) in the H–mode. Although existing diagnostics are being used to attack these issues, improvements are needed for definitive experiments. For example, more accurate measurements of current and pressure profiles are needed to allow improved quantitative comparison with MHD stability theory as well as to provide the basic measurements to determine the effects of various profile control techniques. Improvements in the accuracy and spatial resolution of the current profile measurements are especially needed. For kink and tearing modes, better measurements of two and three dimensional internal structure of MHD modes are needed. To properly confront ELM stability, improved internal magnetic field measurements in the edge regions are needed to quantitatively assess the role of edge current profile in the stability of edge localized modes.

In order to optimize the pressure profile, control of heat and particle transport is needed. While significant progress has been made in the past five years in creating regions of the reduced transport in the plasma core, much remains to be done to understand transport, especially in the electron channel.[9,10] Improved two dimensional imaging of microturbulence and methods to look at short wavelength turbulence associated with electron transport are two key diagnostic areas here. Another important issue is the role of E×B shear in controlling turbulence and transport; this involves both the equilibrium electric field[9,10] and the fluctuating field associated with the poloidally and toroidally symmetric zonal flows.[11–13] While existing diagnostics have allowed us to investigate the role of the equilibrium electric field,[9,10] zonal flow measurements are still in their infancy.[14]

For current profile control and for steady-state operation, current drive is required. The ability of MSE measurements coupled with time-dependent MHD equilibrium analysis[15] to determine the internal loop voltage and, hence, the inductive and noninductive currents has led to a significant advance in our ability to assess the effectiveness of current drive techniques. The major diagnostic improvement needed here is a means of improving this analysis to achieve better spatial resolution. It is especially important to find high spatial resolution techniques to cope with the sensitivity to the radial electric field.[16–20]

II. BASICS OF ADVANCED TOKAMAKS

The advanced tokamak program seeks to significantly improve the tokamak as a magnetic fusion device by improving the plasma physics over that utilized for example, in the conventional ELMing H–mode plasma embodied in the ITER FDR design.[21] The ultimate goal of this research is a more compact, steady-state device with greater fusion power density. For purposes of the present diagnostic discussion, both conventional aspect ratio tokamaks such as DIII–D[22] or JET[23] as well as small aspect ratio devices such as MAST,[24] NSTX[25] or Pegasus[26] are included under the umbrella of advanced tokamaks. Indeed, most of the diagnostics that we will discuss are useful for other magnetic confinement devices (e.g. stellarators and reversed field pinches) as well as for tokamaks.

Improving MHD stability is the most pressing issue for approaching the advanced tokamak goal. In the range of plasma density and temperature where a tokamak-based fusion device is projected to operate, the fusion power density is proportional to the square of the plasma pressure.[27] Accordingly, in order to increase that power density, we must find ways to increase the plasma pressure. However, MHD stability sets strict limits on the pressure achievable in a given magnetic configuration, which is usually discussed in terms of the toroidal beta value $\beta_T = \langle p \rangle/(B_T^2/2\,\mu_o)$, where $\langle p \rangle$ is the volume averaged pressure and B_T is the toroidal field at the magnetic axis. The ideal MHD stability limits of tokamak plasmas are well understood;[28] indeed, the excellent agreement between experiment and ideal MHD stability is one of the success stories of fusion plasma physics. Accordingly, theory is a reliable guide in developing the means of extending the MHD stability limit that will be discussed presently.

To achieve the plasma pressure needed for fusion, the energy confinement time τ_E must be large enough that the total power P_T flowing through the plasma can produce this pressure, since the definition of τ_E implies that $\langle p \rangle = (2/3)\ \tau_E\ P_T$. This power flow is due both to fusion power and to any auxiliary power needed, for example, for current drive. τ_E depends on a number of plasma parameters;[29] one significant dependence is a general increase with plasma size. Accordingly, if we can find ways to decrease the plasma energy loss, we can trade off this improvement in τ_E for smaller machine size.

Tokamak plasmas require a toroidal current I_p to maintain the magnetic configuration which confines the plasma. A portion of this current is the self-generated or bootstrap current which is inherent in the configuration. In order to achieve steady-state operation, external current drive must be used to produce the remainder. Since current drive requires injection of significant amounts of power into the plasma, maximizing bootstrap current can have a substantial payoff. Of course, the pressure profile which optimizes the bootstrap current must also give a current density profile which is consistent with the MHD stability requirements. The fraction of the current driven by bootstrap current is given by $f_{BS} = C_{BS}\, \varepsilon^{1/2}\, \beta_p$. Here, $\beta_p = \langle p \rangle / (\mu_o\ I_p^2/2\, \Gamma^2)$, $\varepsilon = a/R$ is the inverse aspect ratio, R is the major radius, a is the plasma half width, and Γ is the poloidal circumference. The coefficient C_{BS} depends on the details of plasma flux surface shape, current profile shape and the relative contribution of density and temperature gradients to the pressure gradient. The relation for f_{BS} suggests that increasing f_{BS} requires increasing β_p; however, β_p is limited by stability just as β_T is.

The MHD stability constraints which affect β_T and β_p are summarized in Fig. 1. In order to simultaneously increase fusion power density and bootstrap current, we need to simultaneously increase β_T and β_p. This means we need to work in the upper right hand corner of the space shown in Fig. 1. The hyperbolic boundary of that region is well approximated by

$$\beta_T\, \beta_p = 25\left(\frac{1+\kappa^2}{2}\right)\left(\frac{\beta_N}{100}\right)^2 \quad , \tag{1}$$

where $\beta_N = \beta_T/(I_p/aB_T)$ is the normalized β and κ is the vertical plasma elongation. Accordingly, the key to simultaneously increasing β_T and β_p is to increase β_N and κ.

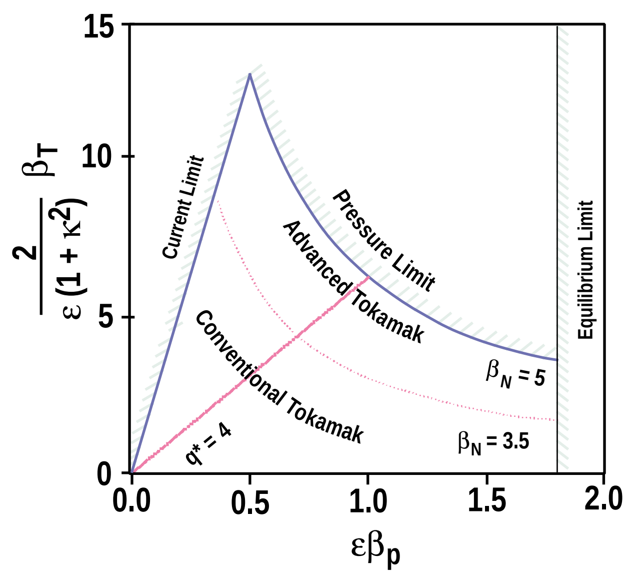

Fig. 1. Plot of the various stability boundaries which set limits on the performance of a tokamak plasma. Here, $\varepsilon = a/R$ is the inverse aspect ratio, a is the plasma half-width, R is the major radius, $\kappa = b/a$ is the plasma vertical elongation and b is the plasma half-height. As β_N increases, the hyperbolic boundary on the upper right moves up and to the right allowing a simultaneous increase in β_T and β_p.

Normalized β, β_N, can be increased by control of the shape of the pressure and current profiles, by changing the shape of the magnetic surfaces in the plasma and by the effect of a conducting wall close to the plasma. In general, broad pressure profiles have higher β_N than those with steep localized gradients.[30] Broad or even hollow current profiles are required for alignment with the bootstrap current and can provide access to the second stable regime for ballooning modes in the core region of the plasma.[31] Vertically elongated, somewhat triangular cross section plasmas have significantly higher β_N than circular cross section plasmas;[32] vertical elongation enters directly into Eq. (1) as well. Finally, in discharges with a broad current density profile, the β_N limit set by kink mode stability is increased by a close, conducting wall;[33] however finite resistivity of the wall makes the problem more complex and will almost certainly require feedback compensation of the resistive effects to reach the highest β_N values.[34] With the plasma shape and profile variations that appear possible,[30] factors of two to three increase in β_N can be achieved.

Although energy and particle transport in tokamaks is less well understood theoretically than ideal MHD stability, there has been significant progress in this area over the past 15 years. This has given us a number of tools which are being used to reduce the energy loss from tokamak plasmas. On the experimental side, there have been a number of observations of transport barriers or regions of reduced transport in tokamak plasmas. The first of these was the initial observation of the H–mode edge transport barrier in ASDEX.[35,36] As has been discussed previously, decorrelation and stabilization of turbulence by sheared $E\times B$ flow plays a significant role in most of the cases of transport reduction.[9,10] On the theoretical side, the emergence of the gyrofluid[37,38] and gyrokinetic[39,40] approaches to turbulence and transport modeling have achieved significant improvements in predictive modeling of the core plasma although the codes still require experimentally measured boundary conditions.

A major part of the challenge of creating an advanced tokamak plasma is integrating all the required parts together into one plasma. Figure 2 shows some of the key interactions schematically. The important point here is that this is a multifaceted problem with a number of internal feedback loops which affect the final plasma state.

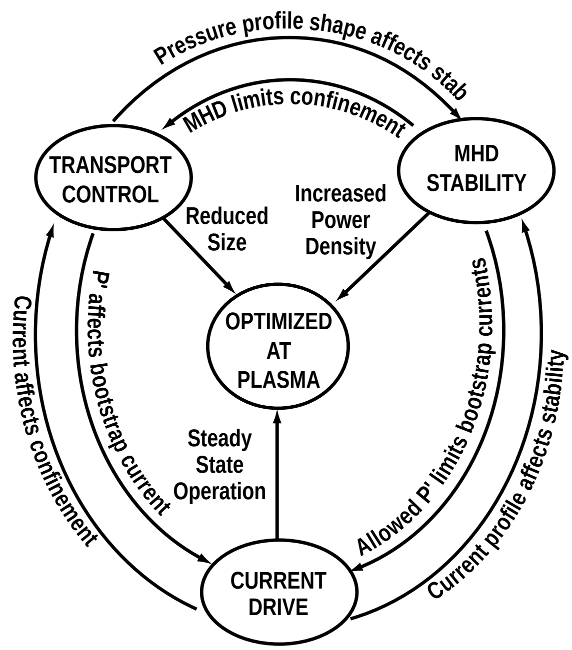

Fig. 2. A schematic diagram of the simultaneous improvements in MHD stability, transport and current drive needed for advanced tokamak operation. Also indicated are some of the interactions between stability, transport, and current drive which affect and constrain the overall optimization.

III. EXAMPLES OF KEY DIAGNOSTICS IN PRESENT-DAY ADVANCED TOKAMAK RESEARCH

As with other areas of physics research, fusion plasma physics research is impossible without quantitative diagnostic measurements. Indeed, new measurement capabilities have frequently produced new insights into plasma behavior. In addition, new measurements have often allowed experimentalists to operate the discharge in novel fashions. The key issues for advanced tokamak optimization are profile shape optimization and control, plasma shape optimization and active feedback control. In this section, several examples are given of how key diagnostics facilitate present-day advanced tokamak research in these areas.

Since profile control is one of the most difficult issues, we first highlight some of the techniques utilized to allow experimentalists to measure and manipulate the profiles of interest. The use of motional Stark effect polarimetry to measure the magnetic field line pitch angle internal to the plasma[5–8] is one of the best examples of how a new diagnostic facilitated the field of advanced tokamak research. Although it was not specifically developed with advanced tokamak research in mind, MSE became routinely available in the early 1990s just about the time that experimentalists were interested in exploring the effects of altering the current density profile.

The ultimate tool for creating a range of current density profiles is, of course, localized current drive (e.g. electron cyclotron current drive, lower hybrid current drive); however, a significant amount of preliminary investigation has been done by heating the plasma during the initial current ramp, thus slowing current diffusion and extending the period with broad or hollow current density profiles. An example of such a shot is shown in Figs. 3 and 4. Although this kind of current profile investigation could, in principle, have been done at any time in the past 20 years, without the ability of MSE measurements to determine the current profile, the experimentalists would have been flying blind. This inhibited work so much that even interesting preliminary results on hollow current density profiles[41,42] were not followed up as rapidly as they might have been.

The current density and MHD safety factor profiles shown in Fig. 4 illustrate another point about fusion diagnostics — the need to couple the diagnostic to larger analysis codes to fully utilize the data. The basic MSE measurement is the local magnetic field line pitch angle in the plasma, which is the ratio of the local poloidal and toroidal magnetic field. By itself, this is not enough to uniquely determine the current density profile. When this data is coupled in to the Grad-Shafranov equilibrium equation[43] and external magnetic measurements in a numerical

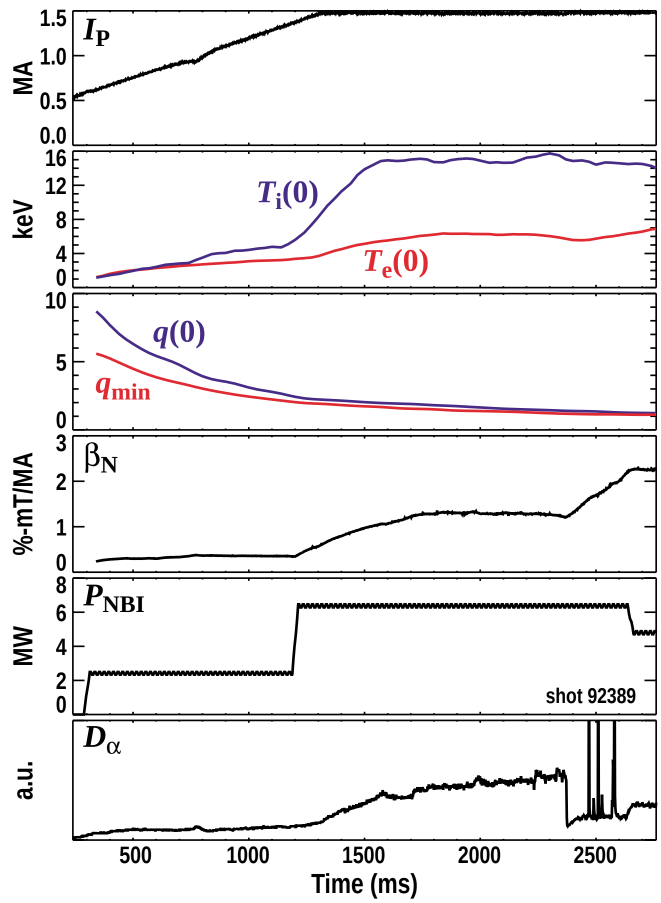

Fig. 3. Time history of a DIII–D discharge run to investigate the effects of hollow current profiles which are formed by heating the plasma early in the current ramp. (a) Plasma current, (b) central ion temperature from charge exchange spectroscopy and central electron temperature from Thomson scattering, (c) MHD safety factor on the magnetic axis q(0) and at the minimum value q_{min} (cf. Fig. 4), (d) normalized beta, (e) neutral beam heating power, (f) divertor D_α radiation. As indicated in (f), this plasma has an L–mode edge until about 2350 ms when the D_α drop shows a transition to H–mode.

MHD equilibrium code (e.g. EFIT[44]), the MSE data provides a significant constraint on the current density profile. Further, a time sequence of MHD equilibrium analysis with MSE input can be used to determine the inductive and noninductive current density profiles,[15] which allows testing of the theory of bootstrap and other non-inductive current sources. This work is one example of the need for time-dependent profile measurements.

Since current density profiles are a significant factor in MHD stability analysis, knowledge of that profile has permitted more accurate comparisons between theory and experiment. One example of this is the work on resistive interchange modes in plasmas with hollow current profiles.[45]

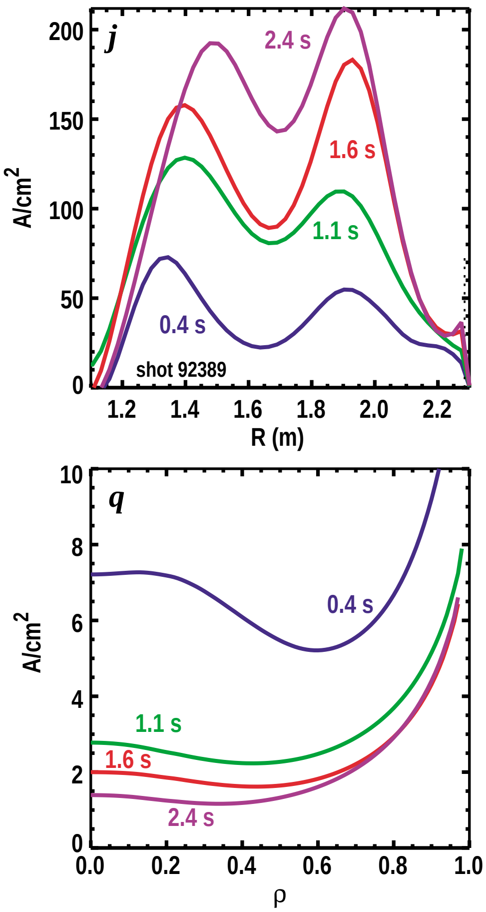

Fig. 4. Profiles of MHD safety factor q and current density at various time in the same shot shown in Fig. 3. The q value is plotted as a function of an effective minor radius, which is proportional to the square root of the toroidal flux inside the flux surface. The current density is plotted as a function of major radius across the midplane of the plasma showing the region both inside and outside the magnetic axis. As can be seen the hollow current profile persists throughout the time range shown in Fig. 3.

Another area where time-dependent profile measurements have been crucial to improved understanding is the area of plasma transport. Over the past 15 years, a whole suite of profile diagnostics has been developed that are capable of producing a complete time history of transport for a single tokamak discharge. This has greatly increased productivity, since the need for repeat shots for transport documentation has markedly decreased. These diagnostics include Thomson scattering and electron cyclotron emission measurements for electron temperature; Thomson scattering, interferometry and reflectometry for electron density profiles; and charge exchange spectroscopy for ion temperature, impurity density, plasma rotation and radial electric field. An example of time-dependent measurements of central electron and ion temperatures is shown in Fig 3; however, the important point is that the complete profiles are measured with the same time resolution as the central values. Typical time resolutions are a few to 20 ms and spatial resolutions are a few to several centimeters in the plasma core to a few millimeters at the plasma edge. Integrating all of these measurements with a time dependent transport code (e.g

TRANSP[46]) reveals the transport changes shown in Fig. 5. This figure shows the dynamics of the formation of a reduced transport region in the plasma core early in the shot, its expansion after the neutral beam power is increased and then the reduction in edge transport caused by entry in to H–mode. Tracing through the various profile changes that occur during these dynamics has been quite important in determining why transport improves.[9,10] Even more so than in the case of MSE and EFIT, these transport results show the importance of being able to integrate the results from a number of diagnostics into the overall analysis code.

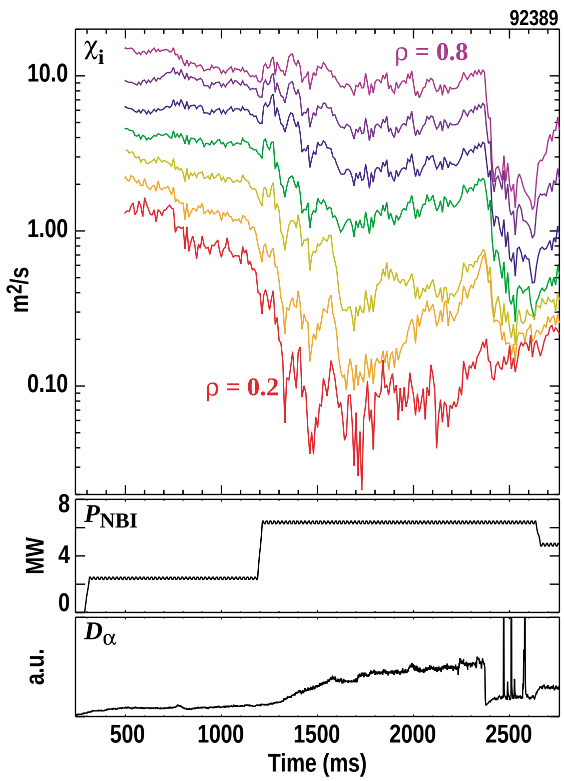

Fig. 5. (a) Ion thermal diffusivity from power balance analysis for a DIII–D discharge with core transport reduction plotted as a function of time for various normalized minor radii ρ. Reduction occurs in the very core even during the low power phase; this reduction region grows after the input power is increased at 1200 ms. Edge diffusivity decreases after the time of the L to H transition. (b) Beam power showing the time of power increase. (c) Divertor D_α measurement showing drop at the time of the L to H transition around 2350 ms. This shot is the same as used in Figs 3 and 4.

Although the effective thermal ion diffusivity can be substantially decreased in various regions of the plasma, as is shown in Fig. 5, most of the transport in the plasma is still anomalous in the sense that it substantially exceeds the predictions of transport driven only by particle collisions. Although the ion thermal diffusivity[47-50] and particle transport[51,52] has been reduced to neoclassical, collision-driven values in some cases, the angular momentum and electron thermal transport are still anomalous. This is believe to be due to additional transport driven by plasma turbulence. An important theme over the past decade in turbulence and transport has been the development of the model of E×B shear decorrelation and stabilization of turbulence.[9,10] In addition to being able to determine the local transport, experimental tests of this model required the ability to measure the E_r to determine the E×B shearing rate[53] and means of detecting the effect of changing E×B shear on the turbulence itself. The E_r measurement in the tokamak core is based on spectroscopic measurements of ion rotation and pressure gradients combined with the radial force balance equation.[9,10] The most accurate of these use charge exchange spectroscopy to determine local values of ion density, ion temperature, poloidal rotation and toroidal rotation.[54] A suite of density fluctuation diagnostics has been used to investigate the changes in core fluctuations which correlate with the transport reductions like those shown in Fig. 5. These include beam emission spectroscopy,[55] reflectometry[56–58] and far infra-red scattering.[59] An example of the changes in turbulence and E×B shear for the same data as in Figs. 3, 4 and 5 is shown in Ref. 60.

The model of the effects of E×B velocity shear was first developed to explain the decrease in transport seen at the formation of the H–mode edge transport barrier.[9,10] Indeed, the H–mode could well be called the first advanced tokamak mode and its edge transport barrier will almost certainly be a part of the ultimate advanced tokamak plasma. An example is shown in Figs. 6 and 7 of the use of charge exchange spectroscopy and reflectometry to demonstrate the connection between increased E×B shear, fluctuation reduction and increased edge gradients across the L to H transition. At the plasma edge, Langmuir probes provide very complete information on turbulence and transport.[61–63] For example, as is shown in Fig. 8, local measurements of the turbulence driven particle flux are possible with this diagnostic.

In all the work shown in Figs. 3–8, the ability to have complete sets of measurements, with both high temporal and spatial resolution, has been quite important in developing an understanding of what the plasma is doing. For example, results like those in Figs. 6 and 7 have been one of the key parts in establishing that changes in the E×B shearing rate can cause changes in turbulence and transport.[10] Results such as those in Fig. 8 have been key in demonstrating further that the actual turbulence driven transport does respond to the change in shearing rate.

The previous examples have stressed the role of profile diagnostics in making measurements which have led to improved understanding. Diagnostics also play a role in active feedback control. One example is the work on feedback stabilization of the resistive wall mode.[64,65] This mode exists because of finite resistivity of the plasma wall in cases where the plasma β_T is above the no-wall ideal limit.[34] The enabling technology here is the integration of the diagnostics, a series of saddle loops, into a feedback system which drives currents in a set of correction coils. As is illustrated in Fig. 9, various algorithms have been tried in the digital feedback system in order to minimize the effect of the resistive wall mode and extend the time that the plasma can remain above the no-wall limit. In the case shown, the plasma current is being continually ramped up so that the resistive wall mode is increasingly unstable. This accounts for the modest

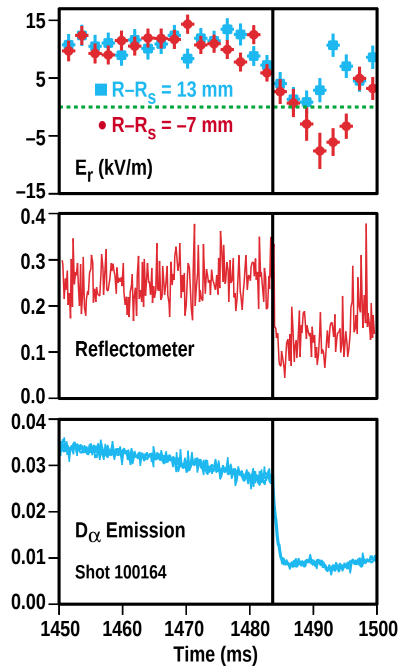

Fig. 6. Time history across an L-H transition in DIII–D showing changes in (a) radial electric field, (b) density fluctuations from reflectometry and (c) divertor D_α emission. Note that E_r begins to change slightly before the time of the transition as indicated by the drop in the reflectometer and D_α signals.

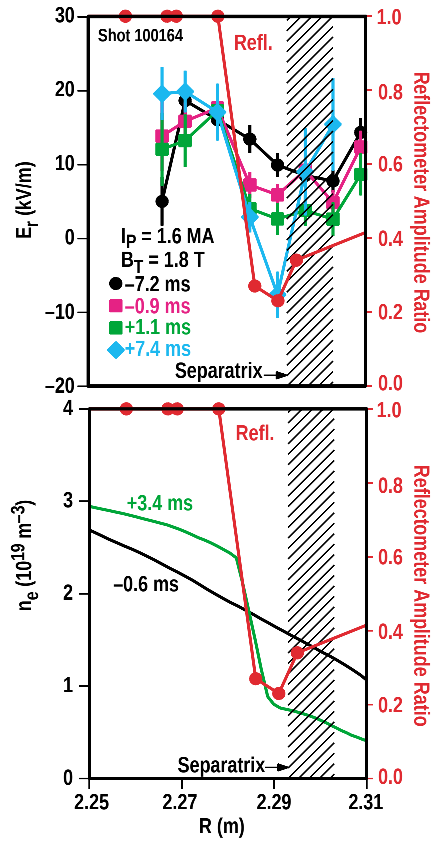

Fig. 7. E_r and density profile across the L to H transition for the same shot as in Fig. 6 showing the formation of the E_r well and the steepening of the edge density gradient characteristic of the H–mode.

improvement in discharge length with feedback. These results are, however, a clear illustration of the sort of feedback control mentioned earlier which will almost certainly be necessary to hold the plasma at the highest β_T values. Even in this case, current profile and, especially, rotation measurements, will play a significant role in the ultimate understanding the effect of feedback on the resistive wall mode.

One final point needs to be made about the diagnostics for advanced tokamak research. As is shown by Figs. 3, 4, and 5 and the examples in Ref. 60, working out the complete picture of what is happening in the plasma requires a suite of diagnostics which are all capable of working together on the same set of

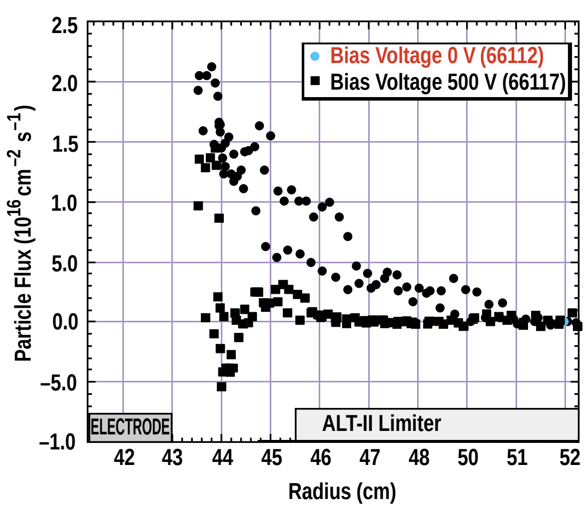

Fig. 8. Local particle flux in biased L–mode and H–mode in the TEXTOR tokamak showing the marked decrease in the local, turbulence-driven particle flux in the H–mode.

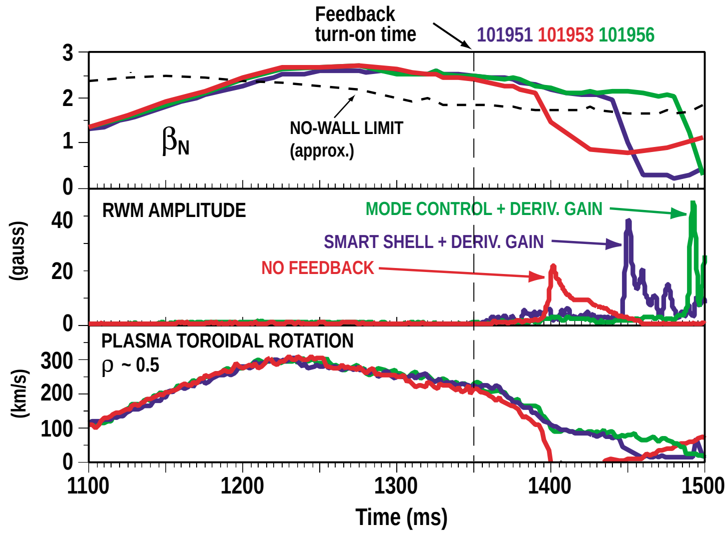

Fig. 9. Time history of three shots from a resistive wall stabilization feedback scan on DIII–D. (a) Normalized beta showing the value exceeding the no-wall limit at about 1200 ms. (b) Amplitude of the resistive wall mode as detected by saddle coils outside the DIII–D vacuum vessel. (c) Toroidal rotation of the carbon impurity from charge exchange spectroscopy. The three different curves show the effect of different feedback algorithms in stabilizing the resistive wall mode. In these shots, the plasma current is being ramped continuously, driving the resistive wall mode increasingly unstable. This accounts for the modest extension in discharge length with feedback over the length without.

discharges. In a very real sense, this is the diagnostic analog of the integration that is illustrated in Fig. 2. This requirement also provides a significant constraint on the development of new diagnostics, since they must be capable of operating in the advanced tokamak environment.

IV. FUTURE DIAGNOSTIC NEEDS FOR ADVANCED TOKAMAK RESEARCH

In order to work towards the integrated advanced tokamak scenario which is the ultimate potential of the tokamak, we need a number of diagnostic improvements. These are needed both for increased predictive understanding and, in some cases, as sensors in feedback control systems. In this section, we consider the quantities which need to be measured and, in some cases, discuss techniques which might be used to measure them. This list of techniques is by no means complete; indeed, significant diagnostic innovation will be required to routinely and reliably measure the quantities required.

In considering future diagnostic needs for advanced tokamak research, diagnostics for MHD are the most fundamental, since discharges which violate the MHD stability criteria either disrupt or have significantly reduced confinement. Since ideal MHD works well under conditions where it applicable,[28] most of the cutting edge research in the MHD area is on non-ideal effects. These include the effect of neoclassical equilibrium currents on stability (neoclassical tearing modes[66]), the effect of the previously mentioned resistive wall,[34] and the effect of multiple coupled modes contributing to edge localized modes (ELMs) in the plasma edge.[67]

To improve our understanding of nonideal MHD stability, in addition to enhancing the capability of existing profile diagnostics, we need

1. Techniques to measure the internal structure of rotating and non-rotating MHD modes with toroidal mode number n in the range of 1 to 5. These are the key modes for kink and resistive wall mode work. Two and three dimensional reconstructions of mode structure are needed.
2. Techniques to determine the mode structure of the moderate to high toroidal mode number modes (n = 10–30) which contribute to ELMs.
3. Methods of accurately measuring the edge current density in the outer 20% of the plasma. This is crucial determining whether the edge plasma has access to the second-stable regime for ballooning modes.[67]
4. Measurements of the fast ion pressure profile in the plasma core. This is needed to find the contribution of the fast ion pressure to the total pressure for equilibrium and stability calculations as well as for determining the contribution to fast-ion-driven instabilities.

5. Measurement of the local B_T internal to the plasma, the *ff′* term in the equilibrium toroidal current density.[27] Knowledge of this term will help constrain MHD equilibrium calculations and the stability calculations based on those equilibria.

The low n internal mode structure might be determined by using multiple ECE systems to image the plasma at several poloidal or toroidal locations. Preliminary work in this area has been done on TFTR using two ECE systems.[68] Another possibility would be to use beam emission spectroscopy or tangential soft x-ray cameras to produce two dimensional images at several toroidal locations.

The edge current density determination near the plasma edge can be done by accurate magnetic field line pitch angle measurements. This can either be done using MSE polarimetry or by using Zeeman polarimetry based on a lithium neutral beam.[69] One might think that the existing MSE systems could immediately be used for this measurement. However, there is a problem owing to the large E_r that exists in the edge of H–mode plasmas.[16] Because the MSE measurement is also sensitive to that local E_r in the plasma,[17,18] either two different views[19] or measurements at the full and half-energy[20] of the neutral beam are required to separate the effects of magnetic field and electric field. Owing to the extremely accurate spatial resolution required for the edge measurement, having two views would require two neutral beams, one injected in the direction of the plasma current and the other opposite to it. The lithium beam is not sensitive to the E_r and the beam penetration is sufficient to cover the 5 to 10 cm wide region of interest in the plasma edge.[69]

The measurement of *ff′* can be done either by spectroscopically measuring the wavelength splitting of the motional Stark components or by simultaneous O–mode and X–mode reflectometry.[70] An example of the wavelength splitting of the D_β emission in a plasma used for resistive wall mode studies is shown in Fig. 10. It is clear that the splitting changes enough during the shot to be readily measured.

The theory for transport in tokamaks is not as far advanced as ideal MHD. Accordingly, while specific diagnostics are targeted at testing specific aspects theoretical models, there is also a need for diagnostics which can test the fundamental paradigm on which those models are based. Since the fundamental assumption is that anomalous transport is driven by turbulence, the diagnostic needs heavily emphasize turbulence measurements.

To improve our understanding of transport, we need

1. Techniques for two dimensional turbulence visualization. In the studies of the dynamics of neutral fluids (e.g. water), the ability to visualize the flows has lead to fundamental advances in understanding; being able to do the same for plasmas would almost certainly pay similar dividends.

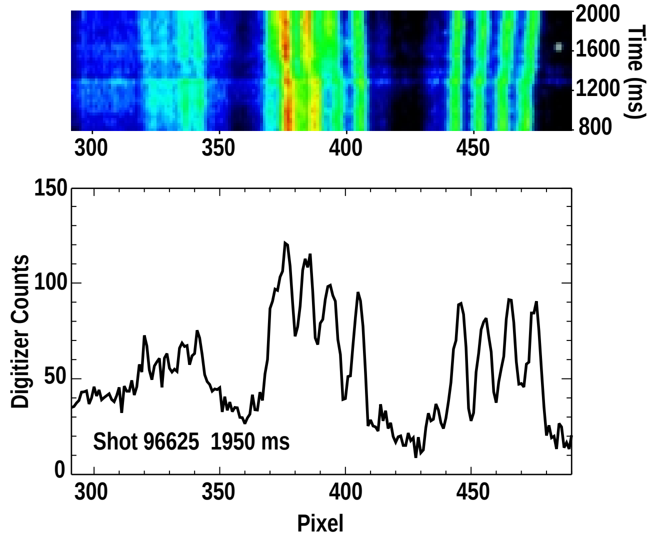

Fig. 10. Stark splitting of Doppler-shifted D_β from DIII–D shot 96625. The lower box shows a spectrum as a function of wavelength at one particular time during this shot. The dominant features here are the eight strongest peaks from the Stark-split D_β from the full energy component of the neutral beam. Two weaker Stark components lie between these eight. Off to the left is the longer wavelength group of the Stark components from the half-energy component of the neutral beam. The shorter wavelength components of the half-energy group are obscured by the left-most four peaks of the full energy set of Stark peaks. The upper box shows this spectrum as a function of time in the discharge. The splitting changes, indicating that the measurement can resolve changes in |B|.

2. Methods of determining the nature of the basic modes included in the theories. Transport theories are couched in terms of various micro-turbulence modes such as the ion temperature gradient mode, trapped electron mode, electron temperature gradient mode, etc. We need to devise definitive tests to determine if these modes are actually present and if their properties really match those given by theory.
3. Techniques for determining what drives electron transport, especially in plasmas where E×B shear has reduced ion thermal diffusivity to neoclassical levels. One part of this question is whether magnetic fluctuations play any role in tokamak transport.
4. Direct measurements of zonal flows.[11–13]
5. Techniques to determine whether large events (e.g. avalanches[71,72]) play a significant role in tokamak transport. This is especially important to determine in plasmas in reduced transport regions.
6. Techniques to demonstrate quantitatively that fluctuation-driven transport is big enough to play a role in the plasma core.

To judge by the number of papers at this meeting,[73–76] the challenge of two dimensional imaging of turbulence is being taken up by a number of experimentalists. The beam emission spectroscopy work[73] has already produced first results across the L to H transition at the plasma edge while the other techniques are in the design study phase.

There are several examples of measurements which test the basic modes included in transport theories. Some preliminary work on testing the ion temperature gradient mode predictions has been done using high speed spectroscopy.[77] Work on electron temperature gradient modes and on electron transport modes in general can be done by high-k scattering ($k \gtrsim 10\ cm^{-1}$) based either on microwave scattering[78] or on far infra red scattering.[79]

Zonal flow measurement techniques have been discussed by several authors[14,80] and some preliminary results have been obtained using bicoherence analysis.[80] A key issue here is isolating the poloidally and toroidally symmetric portion of the turbulence characteristic of the zonal flows from the bath of fluctuations that exists at any given point in the plasma.

The greatest challenge for core fluctuation diagnostics is to create the complete measurement set needed to determine the fluctuation driven particle and heat flux. This requires measuring $\tilde{n}$, $\tilde{v}_r$ and $\tilde{T}$ simultaneously so that the fluxes can be determined from the time average of the various cross products. This requires measuring both the amplitude and the phase of the various oscillations as a function of time. The initial measurements are likely to be point measurements at best. The second part of the challenge is to find a way to determine what the poloidal average over the flux surface would be. This requires either poloidally distributed measurements or some powerful theoretical arguments which would allow the average to be computed from the point measurement.

The final portion of the advanced tokamak triad is current drive. Perhaps because localized current drive tools are just becoming widely available, the list of desired measurements is actually quite short. First, there is a need for improved analysis of the internal loop voltage for better determination of the noninductive current. This involves improved spatial resolution and accuracy of MSE measurement, including a way to allow for the E_r sensitivity of the measurement without compromising the spatial resolution. Second, there is a need for more information of the distortion in the electron distribution function caused by the current drive itself. A possible technique here involves the use of a tangential viewing soft x-ray camera.

MSE measurements have played and will continue to play a role in the stability and current drive areas. MSE measurements are particularly challenging in tight aspect ratio advanced tokamaks [24–26] where the toroidal field is a fraction of a Tesla. This reduces the wavelength separation of the motional Stark components by about an order of magnitude from that in machines with more conventional aspect ratios. Special techniques (e.g. Lyot filters) will be needed to separate the various Stark components. In addition, the MSE measurement will require installation of a diagnostic beam on machines which are not equipped with neutral beam heating.

Although this discussion has emphasized diagnostics needed to advance predictive understanding, we should not lose sight of the need for diagnostics to be built into plasma control systems. This imposes significant extra demands on the diagnostic because we need to process the acquired data in near real time and provide it to the plasma control system so that a complete feedback loop can be created. Examples of quantities needed for feedback control range from simple scalars to complete spatial profiles. Scalars include β_T and β_p while profiles include the q profile, plasma rotation profiles (for resistive wall mode control) and electron density and temperature profiles (for current drive control).

Most of this discussion of diagnostics has focused on hardware, as is appropriate for the High Temperature Plasma Diagnostics meeting. However, software issues are becoming increasingly important. As the gyrofluid and gyrokinetic models mature, there is increasing need to test their detailed predictions against experiment. However, too often the code results are couched in terms of quantities which are not measured or which are difficult to measure (e.g. local fluctuating electrostatic potential). It is important for diagnosticians and modelers to work together so that software versions of various diagnostics can be built into the modeling codes. In this way, the comparison of the measurements with the code predictions can be made in a much more definitive fashion. For example, if there is a spatial or temporal averaging that takes place in the actual measurement, the software analog of the diagnostic could have the same features built into it. While diagnosticians will continually work on creating improved measurements, we need to engage the modelers so that, as best possible, they predict the quantities that we can actually measure.

V. ISSUES IN REACTOR-SCALE PLASMAS

If we are successful in present-day devices in making a significant advance towards the advanced tokamak goal, we will then have to confront the results of our own success. We will have developed the capability for creating a high power density fusion device with all of the associated engineering problems. An enormous issue here is the divertor. Although the divertor is not the dominant issue in present-day devices, it will be a major engineering challenge at the power densities which an advanced tokamak reactor could reach. How to minimize the heat load on the divertor components while still maintaining the attractive feature of the steady-state, advanced tokamak core is a formidable problem.

Reactor level issues affect the core plasma diagnostics in a number of practical ways. The problems of implementing diagnostics on a reactor grade plasma are formidable.[81] There are the obvious ones of lifetime of diagnostic components, radiation-induced noise, long-pulse or steady-state operation, maintenance of alignment and calibration. Less obvious ones are the need to interface to machine systems such as blanket modules and cryostats, the need to satisfy stringent requirements on vacuum integrity and tritium containment, and to be compatible with remote handling and compliant with safety requirements. In addition, port space is likely to be quite limited. The consequence of all this is that there is a need to develop relatively simple and rugged diagnostics. This need for rugged, simple diagnostics is especially acute for those which will be part of active feedback loops controlling the plasma.

The divertor diagnostic issues in a reactor-grade plasma have special, additional difficulties.[81] The access is poor while the requirements are demanding and the problems of lifetime of diagnostic components are even worse in the divertor than at other locations around the device. In addition, the knowledge base and experience are far less developed.

Even if we achieve the advanced tokamak plasmas consistent with Figs. 1 and 2, these issues demonstrate a need for continuing, long-term work on high-temperature plasma diagnostics.

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