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DIAGNOSTIC DEVELOPMENT FOR THE DIII-D NATIONAL FUSION FACILITY

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R.L. BOIVIN for the DIII-D TEAM

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GENERAL ATOMICS

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The DIII-D National Fusion Facility has long been a center of innovation and development of diagnostics for magnetic fusion devices. The DIII-D device, a moderate size tokamak, with a high flexibility shaping coil set, neutral beam injection (NBI), electron cyclotron heating (ECH) and ion cyclotron heating (ICH), supports a very broad research program in fusion science, including critical aspects related to burning plasmas expected to be encountered in ITER. This scientific program is supported by a large set of diagnostics (approximately 50), which is the product of a highly collaborative program between universities, national laboratories and industry. Although many diagnostic systems are now routinely employed to measure the majority of plasma parameters, such as temperature, rotation, density and current profiles, there are many areas that are inherently difficult or prohibitively expensive to diagnose. Such areas include the measurements associated with energetic ion populations or with the characterization of plasma flows in the divertor/edge area. In addition, the study of burning plasmas will require the development of new and updated techniques, which need to be developed and tested in existing devices in relevant plasma conditions.

I. INTRODUCTION

The DIII-D National Fusion Facility is a leader in the development of the scientific basis for the optimization of the tokamak approach to fusion energy. Comprised of a national and international team of institutions, the collaborative program includes major elements of fusion science, namely stability, transport, wave-particle interaction and boundary physics. A major and fundamental aspect of this program is the integration of these elements with a goal of improving the tokamak concept. At the center of this integration effort resides the development of steady-state high performance scenarios, also known as the Advanced Tokamak. The development of these scenarios is particularly important for a study of burning plasma physics and the ultimate development of fusion energy systems. In particular, the DIII-D program is committed to resolving issues to ensure and to benefit from a successful ITER operation.

In its current five-year plan, the DIII-D research goals have been established to:

- To play a leading role in a national effort to understand the basic physics processes by which plasma turbulence produces cross-field transport and to use that knowledge to improve confinement,
- To clearly establish that the limit of plasma pressure in the tokamak can be substantially increased by the effects of a nearby conducting wall and plasma rotation, as predicted by theory,
- To understand flows and exchanges of fuel and impurity particles between the plasma and the material surfaces of the confining chamber for fuel and impurity control,
- To develop the basis for steady-state operation of a tokamak in which the large majority of the plasma current is self-generated,
- Proceeding from the above scientific building blocks, to implement sufficient plasma control (current profile, pressure profile, rotation, density) to assess the degree of achievable simultaneous optimization of the tokamak in the above four major areas of fusion science.

In order to achieve these goals, a significant effort is required to develop the best diagnostic set for unraveling the physics of fusion plasmas and to validate theoretical concepts and modeling, in support of next step experiments. At DIII-D, these diagnostics are the result of a sustained effort from many collaborating institutions, including universities, national laboratories and Industries. This development is also an excellent opportunity for students and post-docs to learn many aspects of fusion science and, in the longer term, can lead to a fruitful participation in new experiments such as ITER.

Over the last year, the facility has undergone significant upgrades[1] including, but not limited to, increased electron cyclotron (EC) power and pulse duration, rotation of one (out of 4) beamline (neutral beam heating) from co-current to counter current injection, density control for high triangularity double-null divertor plasma and many improved diagnostic systems. This major facility upgrade provides a renewed opportunity to address key issues for ITER success, in the scientific and in the diagnostic areas. It offers new possibilities to emphasize efforts in developing measurement techniques and related issues for burning plasma experiments.

The efforts in diagnostic development in DIII-D cover a wide range of research areas. They can be found associated to the following groups:

1. Development of diagnostic techniques for scientific research in support of ITER. Some key scientific issues that remain to be addressed for ITER require specific measurements, some of which have never been performed before. These systems are designed for existing experiments to answer open questions, and may not be fielded on ITER, although, if proved to be useful, robust and/or versatile, could be part of a future upgrade.

2. Development, design and testing of identified systems for ITER. The design of many diagnostic systems for ITER has reached a certain degree of sophistication sufficient for the evaluation of their expected performance. In that process many issues have risen that require testing and specific R&D in the laboratory or on existing devices.

   These tests will aim at finding solutions to many problems and increase our confidence in the performance of these systems in a challenging fusion environment. Presently, many of these tests performed on DIII-D have focused on systems for which the US intends to deliver to ITER. The diagnostics for which the US is responsible include the electron cyclotron emission (ECE), the motional Stark effect (MSE), reflectometers, interferometers, the residual gas analyzer (RGA) and imaging systems. Others may be added as needs arise or in collaboration with other international partners.

3. Development of new and/or alternative techniques for ITER and other burning plasma experiments such as DEMO. The development of new diagnostic techniques has a long tradition at DIII-D. This effort is continuing, with a focus on identified gaps in ITER measurement needs. Significant efforts are also being applied to the development of alternative techniques to diagnostic systems that presently are not expected to meet ITER requirements. Finally, this work is very important for the success of a DEMO reactor which will require simple, robust and reliable diagnostics typical of what a commercial reactor may require.

4. Study of generic diagnostic issues encountered in the burning plasma environment. Many of the conventional diagnostics will be exposed to conditions imposed by the environment which may compromise their operation in the short or long term. Such issues include the effects of radiation (e.g. neutrons, gammas) on optical elements, insulators, conductors, etc. They also include the effects of long-term exposure to plasma deposition and/or erosion on the first optical component. In addition, constraints will be encountered on the reliability of the systems' calibration and their protection shutters.

5. Development of techniques for real-time control of the plasma characteristics through a state-of-the-art Plasma Control System, with the goal of steady-state operation. While historically, the wide majority of the experimental data is normally available offline, i.e. after the discharge is completed; an increasing number of measurements are needed in real-time to control the plasma and its fundamental characteristics, including precise and targeted internal profiles. These operations thus require not only good physics models to relate the measurements to the proper actuator, but also very robust measurements, with proper error handling and reliable calibration.

This paper will describe examples of diagnostic development along these five lines of research, in their respective section. In the summary, a short description of future avenues and open issues will be presented.

II. DIAGNOSTIC DEVELOPMENT IN SUPPORT FOR ITER SCIENCE

One of the most impressive recent improvements in diagnostic capability on DIII-D relates to developing understanding of turbulent transport and plasma stability. In these areas the development of the beam emission spectroscopy (BES) system[2] is particularly noteworthy. Using a wide collection optical system, coupled to high-efficiency detectors, this diagnostic measures the localized plasma density fluctuations through the line emission of excited neutral beam particles. This diagnostic presently consists of 32 channels typically assembled in array of 5x6 "pixels" (with 2 additional serving as common mode monitors). The fluctuations in emitted light (due to density fluctuations) show clearly the formation and evolution of small scale eddies (~1-2 cm), especially near the edge of the plasma, indicative of turbulent plasma transport. The recent upgrade of the system has improved significantly the signal to noise ratio and consequently enabled this type of measurement to be done much further inside the discharge, in some cases very near the core. The application of recently developed sophisticated time-delay techniques[3,4] on the fluctuating field has permitted the derivation of the mean flow and fluctuating velocity fields. This has proven to be important in the understanding of transport mechanisms and their suppression in high confinement regimes. In addition, this instrument has been increasingly used to study the Alfvén wave activity within the plasma due to the presence of fast ions, taking advantage of the localized nature of the measurement.

This study of transport mechanisms has also benefited from significant development of scattering techniques to detect local fluctuations. By launching a known far-infrared (FIR) or microwave beam (with a known **k** vector), one can measure the plasma fluctuation spectrum by analyzing the scattered beam (based on energy and momentum/vector conservation).[5] These measurements have reached a broad spectrum of wavenumbers ranging from 0 to almost 40 cm^{-1}, which is believed to encompass most, if not all, of the turbulent mechanisms responsible for the anomalous transport. This system has been also instrumental in identifying and characterizing the Alfvénic activity across a much broader physical window.

In addition, the interferometer has been completely refurbished and upgraded, yielding information previously inaccessible, both in terms of fast events and of small amplitudes. The interferometer is a dual color (CO_2 and HeNe) system with vibration compensation measuring the line-integrated density. Recent upgrades include a fast, high precision digital fringe counting[6] capable of resolving very fast, low level fluctuations. Many other systems, such as the phase contrast imaging (PCI)[7] and reflectometers[8] have also been upgraded by increasing their coverage, and their frequency response.

As mentioned, all of these upgraded systems have had tremendous impacts on the characterization of Alfvénic activity and their effect on confinement. This undertaking has further supported a significant upgrade of the ECE diagnostic.[9] Measured blackbody radiation (typically in the microwave region — ~100 GHz) yields directly the local electron temperature for all cases when the plasma is optically thick, a condition found in most fusion plasmas. Over the last two years, three significant upgrades (on the radiometer system) have allowed new physics to be uncovered with this otherwise conventional diagnostic. First, all the microwave bandpass filters, which indirectly select the radial location, have been replaced by narrower ones, from 1 GHz previously to 400 MHz. Second, the receiving antenna has been redesigned to reduce the vertical and toroidal spot size in the plasma. Both improvements have led to increasing the localization of the emission by more than a factor of 2 across the full plasma profile. In addition, the data acquisition system has been upgraded to a faster and higher resolution system (14 bits versus the previous 12 bits), improving our ability to observe and isolate small fluctuation levels.

All these improvements, especially for the ECE, as illustrated in Fig. 1, and for the interferometer cases, have led to improved expectations of being able to detect and characterize the Alfvénic activity on ITER. Its large size would prevent, for example, the detection of these modes by conventional magnetic probes. Not only does this development improve our confidence in diagnosing these modes in ITER, but also provides the critical localization capability.

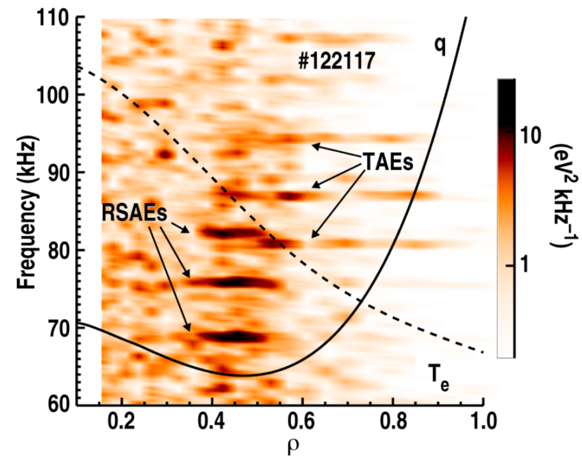

Fig. 1. Radial profile of the frequency spectrum of electron temperature fluctuations. modes observed correspond to toroidal Alfvén eigenmode (TAE) and reverse shear Alfvén eigenmode (RSAE).

III. DEVELOPMENT OF DIAGNOSTIC TECHNIQUES FOR ITER

Recent allocation of the credited diagnostic systems to the seven participating parties, has sparked a renewed interest in diagnostic development in the U.S. and abroad, reenergizing the design and development of these systems. At DIII-D, design and test studies have begun on both interferometers (tangential and divertor), while plans are being developed to extend this work to the reflectometer, ECE and MSE systems, and possibly others.

In addition to the work described in the previous section for the conventional interferometer, a prototype tangential interferometer and polarimeter system is being installed on DIII-D. In a geometry based on the one expected in ITER, the tangential system consists of an entry mirror (in our case made of polished copper), near the port and a retro-reflector at the far side of the tokamak, with a chord crossing near the magnetic axis, at the equatorial plane (midplane) elevation. The first test will be conducted at CO_2 wavelength (10 μm). This arrangement will be instrumental in testing the system against vibration, while gathering information on mitigating mirror coating (Sec. IV) and developing in-situ fast alignment feedback capability. This work will be easily extended to the divertor system, which presents similar technical difficulties with the added challenge of refraction due to very high plasma density in this region. Related details can also be found in Sec. IV in the discussion of fiber optic use inside the tokamak.

One of the fundamental measurements required in a burning plasma, which is not encountered in present day experiments, is the capability of measuring, in nearly real-time, the concentration of tritium at the core of the discharge. Fundamental to optimizing the burn, if not for direct control, one must be able to measure the ratio of deuterium to tritium. This is reminiscent of the need of measuring the hydrogen concentration in an ICRF-heated discharge (hydrogen minority heating), as found in existing experiments. Standard neutral particle or spectroscopic analyses are typically restricted to the edge of the plasma. A novel technique developed in DIII-D is based on the use of the ion-ion hybrid cutoff layer. A superheterodyne reflectometer can provide a direct and robust measurement of the composition of different charge to mass ratio ions (species mix). In the cold plasma dispersion relation approximation, the ion-ion hybrid cutoff frequency is uniquely determined by the density ratio and cyclotron frequencies of the two different species. A (fast wave) reflectometer diagnostic that uses an ICRF wave can be used to measure the ion species mix.[10] A proof of principle has been completed on DIII-D by tracking the phase of a 20 MHz wave launched from an antenna (usually, inner wall) to the cutoff layer and then back to the receiving antenna and was used to provide a direct measure of the hydrogen/deuterium species mix.[11]

IV. DEVELOPMENT OF ALTERNATE AND NEW TECHNIQUES

One of the important elements of scientific investigation in fusion plasmas has been the continuous development of new, innovative ways to measure key plasma parameters. In the case of alternate techniques they are not merely redundant systems, they can also illustrate and uncover some key physics element that could be missed with only one technique. This breadth in measurements brings a deeper understanding of the underlying physics and stimulate the development of new theoretical models. In the long run, this versatility in measurement techniques is beneficial in burning plasma experiments by reducing the risks associated with a challenging environment, with an ultimate goal of reducing their complexity and increasing their robustness.

In DIII-D, the need for a reliable and robust measurement of the current profile has led to a multi-prong approach. Recently, the MSE system has been upgraded with the addition of a 24 channel system[12] which views the recently rotated neutral beam line (from co- to counter-injection). By combining the first system, which views a co-injected beam with 45 chords and the new system, one can infer with great precision the local magnetic field pitch and the radial electric field, across the full profile. Figure 2 shows the new layout of the MSE diagnostic on DIII-D with both co and counter systems.

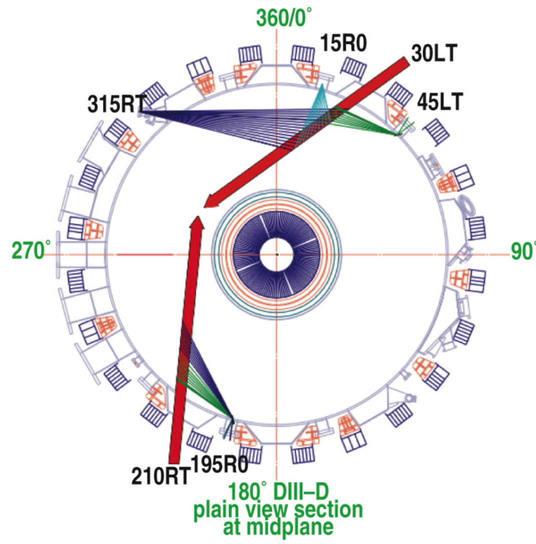

Fig. 2. Layout of the full motional Stark effect (MSE) diagnostic system. The combination of the co-viewing system (top 45 chords), and the counter-viewing system (lower 24 chords) allow the full inversion of the current and radial electric field profiles.

In addition, DIII-D pioneered the development of lithium beam polarimetry[13] to look at the formation and evolution of currents at the edge of the plasma. To avoid the ambiguity described above, where the total local electric field is measured (MSE), a diagnostic for measuring the magnetic field based on the Zeeman effect in the lithium 2S-2P resonance transition has been deployed. Because of the wide separation of the atomic levels in lithium, there is essentially no Stark mixing and the polarization and splitting of the resonance emission is strictly due to the local magnetic field. The neutralized lithium beam is launched radially from the equatorial plane with a total equivalent current of ~10 mA at 30 keV. The detection system consists of 32 channels located vertically below the beam, with a ~5 mm spatial resolution. By making the ratio of the circularly to the linearly polarized light, one can infer the local magnetic field pitch angle. This system has been instrumental in verifying the existence of edge currents driven by the bootstrap effect in presence of a large pressure gradient during high-confinement (H-mode) regimes.[14]

Another technique is being developed and tested on DIII-D which is based on the analysis of the "conventional" Stark components of the local D_α emission from a neutral beam. This analysis promises the evaluation of the local total magnetic field (|B|) and by the ratio of emission lines to obtain the local pitch angle.[15] This technique, while apparently similar to a conventional MSE approach, has the great advantage of not relying on polarimetric measurements, relying instead on spectroscopic technique alone. This implementation on DIII-D will allow a full comparison with the other two techniques, a critical test required before it can be considered as an alternative technique to MSE for ITER, which presents a particularly difficult implementation due to the multiple mirrors and induced Faraday rotation in windows.

We are also making plans to develop a new technique which relies on the use of the Fizeau effect, which presents the advantage of not depending on the use of a neutral beam. The Fizeau effect is a well-known phenomenon causing a small phase shift of an electromagnetic wave traveling through a moving dielectric medium. This phase shift is of the order of V_D/c of the normal phase shift for a stationary medium, where V_D is the drift speed of the medium and c is the velocity of light in vacuum. A practical problem with the measurement of the current density dependent term (the flow of the medium, i.e. electrons) is its small value compared to the phase shift for a stationary medium, since small changes in geometry or plasma density result in far more phase shift than that due to plasma motion. As a result, in tokamaks, with standard interferometers, measurement of the first order term is impractical. This difficulty is overcome with the interferometer arrangement where two legs of the measuring beam are run co-linearly on the same path, out of phase with each other. In this arrangement, the reciprocity theorem guarantees perfect cancellation of the zero order term (standard density measurement), while the contribution due to the motion of the medium is doubled. Initial tests have been performed in the Madison Symmetric Torus (MST),[16] and results are promising that such technique can be applied on DIII-D.

Another critical measurement need resides in the diagnosis of the confined fast ion population, especially of the fusion generated energetic alpha particles, whose confinement is crucial for ignition. Many techniques have been proposed, with varying degrees of expected performance in ITER. In DIII-D, we recently developed a new technique of measuring the fast ion population profile, which could have important application in ITER. In presence of a neutral beam, fast ions (alphas, deuterium beam ions, rf tail ions), present in the discharge, have a finite probability of being neutralized by charge-exchange. These newly neutralized fast ions are typically in an excited state, which will decay rapidly by emitting photons (such as the D_α near 656 nm, in the visible). By selecting the optimal view (in our case, directly above the neutral beam), we can isolate the Doppler broadening due to the fast ion energy only, eliminating the effects of the beam energy and the line emission fine structure. Shown in Fig. 3, is the measured fast ion profile using this technique. Shown are three time slices, varying in Alfvén wave activity, from low or no activity (top curve, t = 1.02 s), to moderate activity (middle curve, t = 0.78 s) and large activity (lower curve, t = 0.38 s). A clear broadening of the profile with increasing Alfvén activity, indicative of increasing expulsion of these fast ions form the core of the plasma.

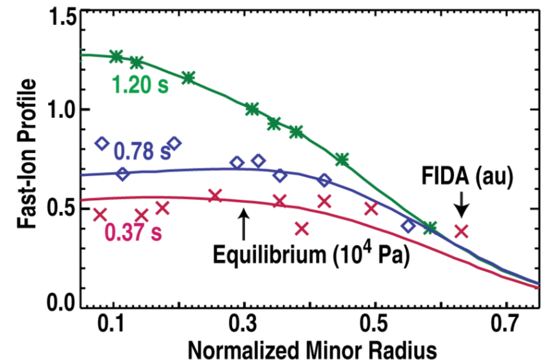

Fig. 3. Measured profile of fast ion population. Shown are three profiles taken at the beginning of a discharge, with high (lower curve, t = 0.38 s), moderate (middle curve, t = 0.74 s), and small or no Alfvén activity (upper curve, t = 1.02 s). Superimposed are the fast ion pressure profiles derived from the equilibrium measurements at the same times.

V. GENERIC ISSUES

While the existing generation of diagnostics may encounter implementation issues that are mostly local in nature (i.e. depending on the geometry, configuration or access in a given experiment), diagnostics slated to be used in ITER and other related Burning Plasma Experiments (even in the case of Inertial Confinement Fusion) will face environmental issues which can compromise their capability of fulfilling their goal.

In ITER, due to its expected high neutron radiation field, only mirrors can be used in optical systems as a first optical element; refractive elements (i.e. lenses) would be quickly damaged or luminesce. However, these mirrors (e.g. first mirrors), are subject to long term degradation through erosion and/or deposition from the plasma.

Many efforts have been expended in experiments around the world in order to understand and to quantify these effects. So far, though, the wide majority of these efforts have been limited to long term tests within an experiment, over the duration of a full experimental campaign, for example, which usually amounts to many months of exposure, over a large number of plasma discharges. Changes in local conditions, wall conditioning, and timing details are making these tests very hard to interpret. Recently a test on DIII-D has been performed in order to test one element of the underlying physics. Mirrors were exposed to DIII-D divertor plasmas using the

DiMES system[17,18] for a small number of repetitive shots (of the order of a day, or equivalently approx 100 s of direct exposure time). This particular test aimed at studying the effect of the mirror temperature on the amount of material deposition. It was found that in the heated condition (at 100˚-150˚C), the mirrors suffered little deposition,[19,20] as shown in Fig. 4. The test was recently repeated with improved temperature control[20] with similar results.

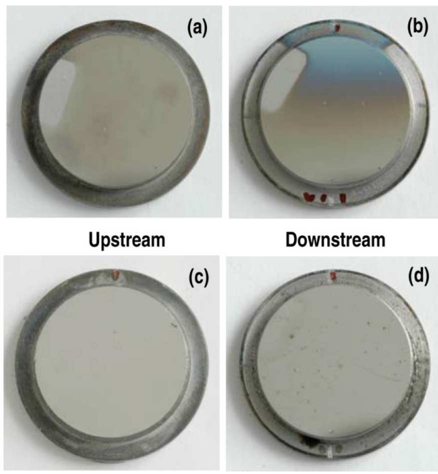

Fig. 4. Set of 4 molybdenum mirrors exposed to 22 identical DIII-D discharges. Mirrors were exposed using the DiMES exposure system in the lower divertor area. Mirrors (a) and (c) were exposed at room temperature, upstream and downstream respectively from the local field lines. Mirrors (b) and (d) were exposed to the same conditions but were heated at a temperature approaching 100˚C. The heated mirrors show no deposited coating, contrary to the non-heated ones.

To further quantify and understand the underlying physics and mechanisms of deposition and particle transport, a series of quartz microbalances[21] has been installed in the lower divertor in DIII-D behind tiles located near the inner wall, a location known for its relatively strong erosion and redeposition. At each location, two quartz crystals are set to resonate, one with an open view to the environment to collect particles, a second with no aperture, used as a temperature compensation reference. One can derive the amount of deposited material by measuring in real-time the small changes in resonant frequencies of the crystals. In this application, the balances (QMBs) can contribute to the physics of particle transport in the boundary area, as well as addressing the issues associated with mirror coating in ITER-like environments.

While further diagnosis of the coating process is continuing, the mitigation technique by heating mirrors has been applied to a new diagnostic, a viewing telescope, installed below the lower divertor shelf. This telescope consists of seven views, fanning from underneath the new shelf installed in 2005-2006. These views are directly coupled to fiber optic cables through a front lens. The telescope itself is made of a single piece of stainless steel mounted to the tokamak floor with 2 heaters and 2 RTDs (1 set being a spare), and is maintained at 150˚C between machine bakes. Long-term results of this application are not yet known, although no effects were immediately visible after 6 months of operations in 2006.

This application served also as a test of new fiber optic technology to be used in high-vacuum, high temperature (up to ~400˚C) conditions. Until recently, only low temperature (<150˚C) assemblies were available. This telescope uses directly coupled fibers which are fed through the vacuum interface without the need for relaying lenses or similar setup. This type of development can help tremendously in fielding diagnostics in hard to reach areas (such as divertors) without having to rely on conventional, more complicated optical trains based on lenses and mirrors. It reduces the needs for maintaining alignment and potential degradation of multiple optical elements. However, these fibers cannot sustain high radiation fluxes at the moment, and are thus relegated to the bioshield and beyond in ITER. Hydrogen doped fibers show promises in that regard but will require substantial testing and development. The use of holey[22] fibers, used in the infra-red (IR) may reduce these effects, and we plan to test these fibers in the near future. This test would be particularly relevant in the development of a divertor interferometer described in Sec. II.

VI. DIAGNOSTICS USED IN PLASMA CONTROL SYSTEM

The ultimate goal of fusion research resides in a controlled burning fusion plasma which can provide sufficient gain, steady-state conditions in stable discharges. While rudimentary plasma control has been available for many years, such as density, shape and total plasma current control, advanced plasma control is a fast growing aspect of fusion research at DIII-D. One of the key elements in that aspect is to develop the proper physics model that relate the “raw” measurement (i.e. the sensor) to the effective response instrument (i.e. the actuator). An important factor in this development at DIII-D is to effectively control profiles (e.g. current, temperature, rotation) in order to optimize the stability and confinement (for example) of the discharge, a fundamental aspect of the Advanced Tokamak line of research.

In DIII-D, many aspects of this control system are being developed, but three elements merit special attention. The first one has focused on the real-time measurement of the current profile using the MSE technique. Of particular interest is the control of the location and amplitude of the local minimum q value (safety factor), hence the plasma current profile, a key element in high performance discharges.[23] This measurement, in turn, controls the amount of neutral beam injection (NBI) and/or electron cyclotron current drive (ECCD) necessary to achieve the target.

The second case involves a co-dependent set of plasma parameters to be controlled simultaneously. It has been demonstrated that sufficient plasma rotation is beneficial in terms of stability and confinement. However, its fine control, through an optimal profile distribution, and the need to also control the stored energy, requires that multiple measurements be incorporated into a control scheme to actuate multiple energy and rotation external sources. In DIII-D, this scheme was recently developed and tested, using a real-time measurement of the rotation profile, through charge-exchange recombination (CER) measurement coupled to a diamagnetic measurement of the stored energy. As shown in Fig. 5, control of the rotation and stored energy was achieved independently of each other, using in real-time the proper mix of co-current and counter-current injection of neutral beam power.

The third case involves the development of a localized measurement of the phase of a neoclassical tearing mode (NTM). Recent results in DIII-D and elsewhere have demonstrated the possibility of eliminating and preventing the presence of this mode by driving local currents through ECCD at the radial location of the mode. It was quickly determined that such a technique may be prohibitively difficult and expensive in ITER. However, by modulating the ECCD with the proper phase with respect to the mode, one can lower the power requirements and optimize its efficiency. Work is now ongoing in using very narrow microwave filters that can determine the local electron temperature response to this mode, as seen by the ECE diagnostic, and in using this information to synchronize the injection of current drive.

VII. OPPORTUNITIES AND SUMMARY

In spite of a sustained effort in the development of new diagnostic techniques and related technology, a significant number of measurement needs remain untapped due to especially challenging requirements and/or high implementation costs.

In DIII-D, such opportunities for the development of new and creative techniques exist, i.e., for the localized measurement of flows in the divertor and boundary areas. These flows represent the largest uncertainty in properly modeling plasmas in that region, an uncertainty which propagates into the expected fuel and impurity transport in ITER divertor. One attempt to achieve these measurements includes the fielding of the fiber optic telescope described in Sec. IV, but difficulties remain in the localization of the emission.

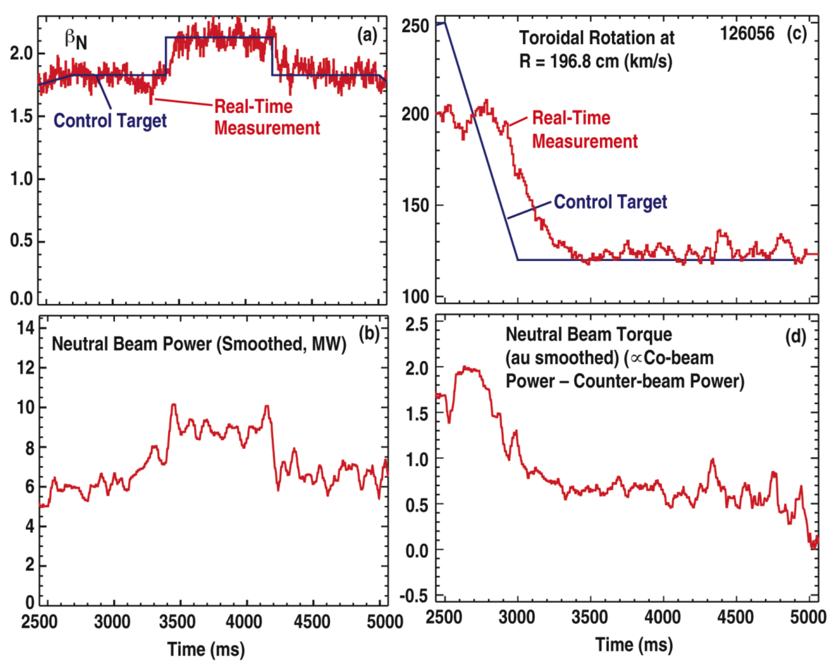

Fig. 5. Time evolution of the controlled normalized stored energy (β_N) (a), neutral beam power (b), toroidal rotation (c) and neutral beam torque (d), using a combination of co-current and counter neutral beam injection. The rotation is measured by the charge-exchange recombination (CER) system, at r/a ~ 0.3.

Another area of inherent difficulty lies in the diagnosis of the dynamics of the (fuel) neutrals near the edge of the plasma, on both sides of the separatrix. Such proper accounting requires a nearly 2-D measurement (around the poloidal cross-section), and in some cases, could require a full 3-D reconstruction, thereby adding the toroidal dimension, which is usually ignorable.

Deviation from the expected axisymmetric geometry in tokamaks, is increasingly looked upon for its importance in stability, confinement and others. That challenge is more pronounced in the magnetic configuration (stability), in the dynamics of particles, both fuel and impurities and the interaction of the plasma with the first wall.

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