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DIAGNOSTIC NEEDS FOR DIVERTOR AND EDGE PHYSICS

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1. INTRODUCTION

In order to determine the diagnostic needs for divertor and edge physics, it is important to understand the current state of divertor research and the goals driving that work. In future large tokamak reactors, such as that envisioned by ITER-FEAT, the divertor, and edge plasma, must meet several goals and criteria. The divertor must first be able to handle the power that crosses the separatrix into the Scrape-Off-Layer (SOL), by careful design of the target plates and use of plasma radiation to disperse the heat load. The divertor must also sufficiently pump helium that is produced by fusion reactions in order to avoid dilution of the central plasma. The boundary solution must also provide fueling for the main chamber to reach the high density that is required for optimal production of fusion energy. Finally the boundary solution must be designed to contain impurities that are produced at the plasma facing surfaces and prevent them from contaminating the main plasma.

Though a few other outstanding issues remain, some solutions to the individual requirements listed above have been demonstrated in a number of current experimental devices. A large number of divertor and edge diagnostics have been developed to demonstrate and study these solutions. A complete set of power balance measurements are routinely available in today's experiments. This includes IR cameras that measure the surface temperature, and thus infer the power flux, at the divertor target and other plasma facing components.[1] Bolometry can now determine the 2-D profile of radiated power that accounts for a major fraction of power balance.[2] Spectroscopy is routinely used to determine how the radiated power is split between different radiating species.[3,4] Numerous spectroscopic techniques are also used to determine impurity levels and profiles as well as other plasma parameters. Pressure gauges have been developed to operate in magnetic fields to measure the neutral pressure within divertor pumping structures.[5] Even measurements of the divertor target surface erosion have been made to project future divertor component lifetimes.[6]

Sophisticated 2-D modeling codes have also demonstrated a number of these solutions. These codes include most all the physical processes thought to be important and place them within the actual geometry of the experiments they are modeling.

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Qualitative, and some quantitative, agreement has been found between the modeling codes and solutions demonstrated in current experiments.[7] These codes are then used for predicting and designing the operation of future divertors which will be larger and must handle greater power densities. The codes attempt to answer a number of concerns for future divertor design. What level of impurities are needed in the divertor to produce sufficient radiation and reduce the target heat flux to a manageable level? What level of core contamination does this lead to? What upstream separatrix density is attained for a radiative divertor solution?

In order to have confidence in code prediction of future divertor operation a more careful comparison and analysis is needed of code modeling of existing experiments. This paper will describe the diagnostic measurements that are needed in today's experiments in order to build confidence in the results of physical modeling using computer codes.

2. PARALLEL TRANSPORT

The divertor and SOL plasma is essentially a complicated transport problem with many atomic and molecular processes which act as sources and sinks of energy and particles. Energy and particles are deposited in the main plasma by heating and fueling techniques. Both diffuse radially across the plasma until crossing the separatrix. Then parallel transport along the open field lines carries energy and particles towards the divertor. Radial transport broadens the deposition on the divertor targets and can lead to plasma flux to the main chamber walls. At the same time atomic processes also transport energy to the chamber walls through radiation and charge-exchange with neutrals. After the plasma flux recombines in the wall it is released into the plasma volume as a neutral fueling source. Impurities are also created by the plasma wall interaction and may enter the plasma. Modeling of such a system then requires sophisticated computer codes which can account for many coupled plasma processes in a realistic 2-D, and sometimes 3-D geometry. To have confidence in these models it is necessary to compare the experimentally measured processes driving the transport, as well as the resulting sources and sinks, with the modeling predictions.

We start with parallel energy transport which can be described by

$$q_{||} = -\kappa T_e^{5/2} \frac{dT_e}{ds} + n v_{||}\left(\frac{5}{2}\left(T_e + T_D\right) + \frac{1}{2} m_D v_{||}^2\right) \qquad (1)$$

where s is the parallel field line length, κ is the parallel electron thermal conductivity, T_e and T_D are the electron and deuterium ion temperatures respectively, n is the plasma density, m_D is the ion mass and $v_{||}$ is the parallel plasma fluid velocity. Under most parameter regimes the SOL and divertor parallel energy transport is dominated by the first term, electron thermal conduction. To account for heat flux, measurements of T_e in the divertor and SOL have been made for some time by a number of techniques. Langmuir probes mounted in plasma facing surfaces have measured T_e of plasma arriving at the target while insertable Langmuir probes have measured T_e at various upstream locations.[8] Other techniques include line ratios of impurity radiation and Thomson scattering. An example of a 2-D T_e profile of the divertor region is shown in Figure 1. This profile was obtained from a Thomson scattering diagnostic where a steady-state divertor plasma was swept across the diagnostic view locations.[9] These T_e measurements have been found to be in general agreement with parallel transport determined from the power balance measurements described in the introduction.[7] The remaining terms in

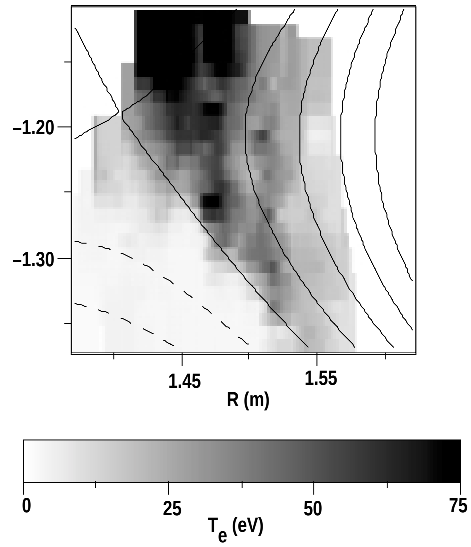

Figure 1. A 2-D profile of the electron temperature in the DIII–D divertor. The profile was taken with a Thomson scattering system by sweeping the divertor configuration across the vertical array of measurement locations.

Eq. (1) represent convection and become important at low T_e near the divertor target and sheath boundary. At the target plate some discrepancy has been found between the measured T_e and expected heat flux.[10] Additional measurements associated with the sheath, including T_D, will be needed to resolve this issue.

The parallel deuterium particle flux is mainly driven by pressure balance and results from particle sources and sinks. Proper modeling of the parallel deuterium flow requires an accurate measurement of the ionization source, or recycling. The neutral particles and impurities that enter the plasma result in most part from plasma flux to material surfaces. Computer modeling should be able to calculate most of these sources if the plasma flux to all surfaces is known. Since it is not, as will be described later, it is important to measure these sources. This is very much a 3-D problem. Most all today's magnetic fusion devices have an outer midplane wall that is very irregular in the toroidal direction. Plasma recycling can and is very much a strong toroidal function. Greater coverage of existing diagnostics, such as vacuum gauges, wall mounted Langmuir probes and D_α measurements would be a good start to address this issue.

Deuterium flux in the poloidal direction can also result from plasma drifts. Radial electric fields near the separatrix of H–mode plasmas, for exapmle, can make a significant contirbution to the total poloidal flow. In principle, most of the plasma drifts can be calculated by today's 2-D modeling codes. In practice, however, obtaining self-consistent solutions with all the drifts terms has proved difficult. Experimental measurements of the poloidal flow profile and the underlying electric fields, or potential, would be a great help in determining where and how they may affect the modeling solutions.

Some useful information on deuterium ion flows have been obtained from Mach probes.[11] These measurements have revealed and/or confirmed basic parameters of SOL and divertor operation. Such measurements, though, are limited to regions, or conditions, that allow probe access. Also, it is very important that both the parallel and perpendicular flow be measured. It is the poloidal flow that is important with the toroidal direction being the one of symmetry. Existing measurements indicate that the contributions to the total poloidal flow from the $E_r \times B$ drift and the parallel flow can be of comparable magnitude. One technique for obtaining simultaneous perpendicular and parallel flows is with a multi-factored Mach probe.[12] Obtaining 2-D profiles, though remains a challenge.

For the impurity ions, the primary forces that drive the parallel flow of an impurity, i, are given by:

$$F_i = -\frac{1}{n_i}\frac{d\left(n_i T_i\right)}{ds} + \frac{\left(v_D - v_i\right)}{\tau_s} + \alpha_e \frac{dT_e}{ds} + \beta_D \frac{dT_D}{ds} + \ldots \tag{2}$$

where the subscript D refers to the main deuterium ion parameters, τ_s is the impurity ion equilibration time and α_e and β_D are the electron and ion temperature respective gradient force coefficients.[13]

Impurity ion densities, temperatures, and flow velocities have been measured by a number of spectroscopic techniques. Doppler broadening and line shift of impurity emission can be used to obtain ion temperature and flow velocity.[14] A limitation of these techniques is that the measurement is spatially localized to the region where a specific impurity charge state exists in sufficient density. In hotter regions of the divertor or SOL the impurity ion will be ionized to a higher charge state.

The main ion properties, as seen in Eq. (2), are also very important in determining the parallel forces on an impurity ion. Though the impurity ion temperature and velocity can be measured by spectroscopy, the main ion may have very different properties. Because the parallel flow times can be shorter than the equilibration time, an impurity ion species can be created in one region and travel to another before it is heated or cooled to the local deuterium temperature. The flow velocity of the impurity may also be very different as can be seen from Eq. (2) where different forces act on the impurity and main deuterium ions.

One prospect for divertor ion temperature measurements is charge-exchange recombination (CER) spectroscopy. Main ions in the divertor would charge exchange with a diagnostic neutral beam. The resulting deuterium atom would be left in an excited atomic state that can be detected by spectroscopy. This is the same basic technique used for main plasma ion temperature measurements. An example of a main plasma CER diagnostic observing the SOL in DIII–D is shown in Figure 2. In this example the deuterium temperature and flow velocity can be different from impurity species that are often measured. Difficulties with this technique include discrimination of the signal from the high level of background recycling and spatial resolution due to the finite lifetime and flight of the measured neutral deuterium atom. A dedicated diagnostic neutral beam may also be expensive. However an exciting prospect for this diagnostic is the possibility of 2-D profiles using a planar beam with imaging measurement technologies.

Finally, modeling of impurity behavior requires measurement of the sources. Impurities are generated by plasma flux to the walls making distributed measurement of plasma wall fluxes, from Langmuir probes or D_α for example, important to this topic also. Measurements of neutral impurity emission can be a more direct measure of impurity generation, but requires some knowledge of the local plasma parameters. It is usually a more difficult spectroscopic measurement.

Impurities released at a surface due to a plasma flux may also vary depending on the physical or chemical structure of that surface. A number of techniques have been developed to study surfaces that have been exposed to plasma operation. However, many of these techniques require long exposure, or must remain in the machine under many

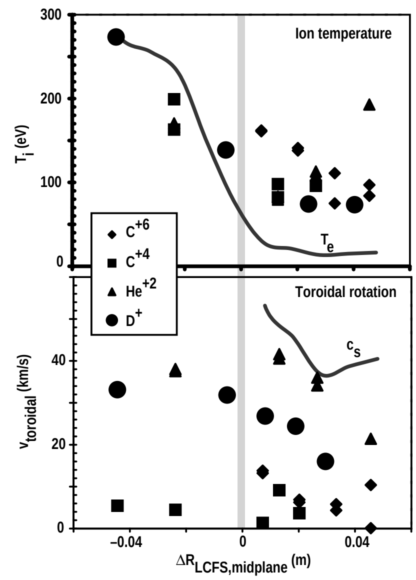

Figure 2. The ion temperature and toroidal rotation profile at the outer midplane as measured by the CER diagnostic on DIII-D for deuterium, He^{+2}, C^{+4} and C^{+6}.

different operational conditions before they can be removed and examined. Development is needed of diagnostics that can measure surface conditions *in situ* and real-time during plasma experiments.

3. PERPENDICULAR TRANSPORT

Perpendicular transport of energy and particles in the divertor, and the SOL in particular, is another issue that needs new measurements. Perpendicular transport is responsible for the width of the heat flux on the divertor target. It also controls the particle flux to all plasma facing components, including the main chamber walls. However, perpendicular transport is not well understood, or even well characterized. To describe perpendicular transport computer codes model the edge plasma with diffusion coefficients that do not vary radially, or poloidally. The coefficients in the codes are varied empirically to match experimental plasma profiles. How these coefficients might scale to future larger tokamaks is unknown. Also, experimental evidence and theoretical considerations suggest that radial transport can vary greatly from inboard to outboard as well as radially. The consequences of this uncertainty are unknown scaling of heat flux widths at the divertor target and unknown particle flux to the chamber walls. The particle flux to the wall is important for understanding main plasma fueling as well as driving SOL flow. The wall plasma flux is also the main source for impurity generation.

A number of diagnostic techniques are used to study SOL and divertor radial transport. Insertable Langmuir probes have measured correlated fluctuating levels of n_e, T_e, and ϕ to obtain localized values of convected and conducted radial fluxes.[15] Other techniques used to observe fluctuation driven transport in the core are now being trained on the SOL. Beam emission spectroscopy (BES) and microwave fluctuation reflectometry are two examples. The largest limitation to all these techniques is their spatial localization, with most having been employed only at the outer midplane. Extending these techniques to get more of a poloidal profile is needed. A 2-D image, encompassing the entire SOL region, would be ideal.

The radial particle flux driven by transport can also be inferred by radial profile measurements of n_e, T_e, and the ionization source profiles.[16] An example of this for DIII–D is shown in Figure 3. A short coming of this technique is that it relies on assumptions of toroidal and poloidal symmetry.

More complete coverage of plasma wall flux is needed. As described earlier the irregular walls in tokamaks today lead to strong toroidal asymmetries. Distributed measurements of plasma to surfaces and/or the resulting neutral density will help define these asymmetries. By knowing the total wall flux profile, coupled with ionization source measurements from spectroscopy, much can be learned about the level of radial transport. These are the same measurements that also provide particle source information for correctly modeling deuterium flow.

4. 2-D PROFILES

The divertor is very much a 2-D system with strong radial and poloidal gradients. The 2-D nature nature of the profiles exists in all the parameters that have been described as important to measure. In the case of the neutral density profile it is even a 3-D problem.

Imaging techniques are ideally suited for this task. Wide angle views of main chamber and divertor recycling have been made in the past. More quantitative work needs to be done with this kind of data in order to turn qualitative pictures into neutral density profiles on a regular basis. Quantitative analysis of visible image data presents some challenges. The emission rates of visible lines are often very strong functions of the plasma density and temperature. To calculate neutral or impurity densities based on this data requires some measurements of the density and temperature profiles. Reflections from other surfaces is also a concern at visible wavelengths. Measurements of lower order transitions $\lambda \lesssim$ 150 nm have less ambiguous interpretation, but imaging these lines is much more difficult because of the technology at shorter wavelengths. Some development in this area is the image of CIV emission taken from DIII-D and shown in Figure 4(a).[17] In this case the detection is made on a phosphor plate in vacuum that is then recorded with a visible camera. Imaging techniques for other parameters would aid in producing 2-D profiles that have been mentioned earlier.

Another difficulty with image data is spatial interpretation, or inversion. While it is easy to discern qualitative features in an image, a detailed quantitative comparison between an image and the results of 2-D modeling is much more difficult. For emission that is expected to have toroidal symmetry some analysis techniques have been developed. On DIII-D and JET,[18] toroidally viewing images have been inverted to produce a 2-D profile on one poloidal plane. An example of a raw and inverted image is shown in Figure 4(b). For signals that may not have toroidal symmetry, such as recycling D_α, or low charge state impurities, other assumptions would be required to invert such images.

A final difficulty with imaging is the sheer size of the data throughput. Though the technology exists for capturing and digitizing image data, more standardized techniques for computational processing and archiving the data are needed.

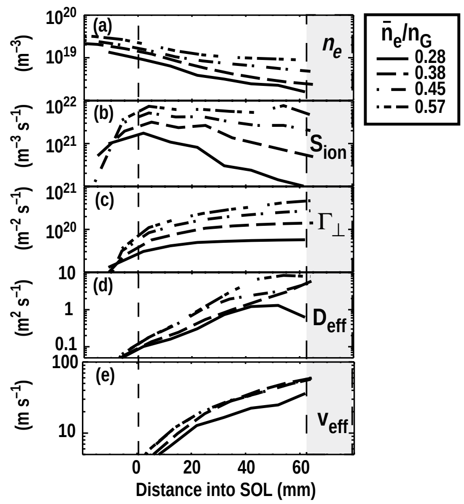

Figure 3. Midplane profiles on DIII–D of (a) electron density from Thomson scattering, (b) ionization source from spectroscopy, (c) the radial flux from particle balance calculations, (d) the effective diffusion coefficient if all flux assumed diffusive and (e) the effective radial fluid velocity if all flux assumed convective.

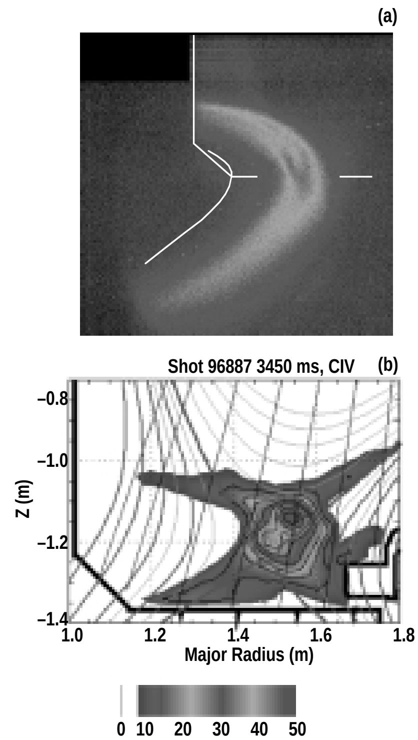

Figure 4. (a) A tangential image of CIV in the lower DIII–D divertor. (b) A inversion of the tangential image onto a poloidal plane.

5. SPECIAL ISSUES

One of the largest concerns in the design of a future fusion device, such as ITER-FEAT, is the issue of tritium retention.[19] Experimentally it has been found on JET that a large fraction of the tritium that has been injected during DT experiments has remained bound up in the walls of the vessel.[20] The level of tritium retention is such that it would limit the operation of a future burning plasma experiment. It appears that most of the retained tritium is trapped in layers of redeposited carbon. Determining the location of carbon erosion as well as its transport in the plasma to the location of redeposition will be required to design a divertor and first wall that minimizes tritium retention. These are all issues that would benefit from additional diagnostic development work as described above.

Another serious issue for the design of ITER-FEAT are the transient heat pulses that arrive at the divertor due to ELMs.[21] ELMs result from a periodic relaxation of the main plasma boundary during H-mode operation. A 1.0 $\mathrm{MJm^{-2}}$ pulse of energy could be deposited on the divertor target in as little as 100 µs. Though the heat pulse originates in the main plasma, understanding how the SOL and divertor plasma responds to such a transient in transport is important in designing future large tokamaks. The issues involved in ELM transport are many of those described above. The difficulty arises that measurements of ELM events must be made on a fast time scale. The ELM heat pulse typically lasts 100–500 µs. Measurements of temperature and density in the divertor must be made on a faster time scale in order to study ELM evolution. Another complication is that the ELM can change the parameters, or signal levels, by more than an order of magnitude during the pulse. This requires diagnostics with an extended dynamic range, or measurements must be dedicated to ELM observation only.

6. CONCLUSIONS

In order to advance our understanding of SOL and divertor plasmas, basic measurements of plasma properties are still needed. Two areas stand out in particular. First, measurements of the main ion properties are needed, in particular the temperature and flow velocity profiles. The other is characterization of the fluctuation driven radial transport, and its spatial profile. Much the same as progress has been made in understanding core radial transport, the same can be done for the SOL and divertor. From a diagnostic point of view, though, it is more difficult because it is an inherently 2-D and in some cases 3-D problem. Advances in imaging techniques to obtain 2-D profiles of a number of plasma parameters can also play an important role in advancing divertor physics.

7. ACKNOWLEDGMENT

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8. REFERENCES

1. C.J. Lasnier, et al., Nucl. Fusion **38** 1225 (1998).
2. J.C. Fuchs, et al., Proc. 21st EPS Conf. On Control. Fusion and Plasma Physics, Montpellier (France).
3. H. Kubo, et al., Nucl. Fusion **33** 1427 (1993).
4. R.D. Wood, et al., 23rd European Conf. on Control. Fusion and Plasma Physics, Kiev, Ukraine, Vol. 20C, Part II (1996) 763.
5. G. Haas, et al., J. Nucl. Mater. **121-151** (1984).
6. D.G. Whyte, et al., J. Nucl. Mater. 266-269, 67 (1999).
7. G.D. Porter, et al., Phys. Plasmas **3** 1967 (1996).
8. G.F. Mathews, et al., Plas. Phys. Contr. Fusion **36** 1595 (1994).
9. S. L. Allen, et al., J. Nucl. Mater. **241-243** 595 (1997).
10. A.H. Futch, et al., J. Nucl. Mater. **196-198** 860 (1992).
11 J. A. Boedo, et al., J. Nucl. Mater. **266-269** 783 (1999).
12. C.S. MacLachy, et al., Rev. Sci. Instrum. **63** 3923 (1992).
13. J. Neuhauser, et al., Nucl. Fusion **24**, 39 (1984).
14. R. C. Isler, et al., Phys of Plasmas **6** 541 (1999).
15. D. L. Rudakov, et al., Rev. of Sci. Instr. **72** 453 (2001).
16. B. LaBombard, et al., Phys of Plasmas **8** 2107 (2001).
17. D. G. Nilson, et al., Rev. of Sci. Instr. **70** 738 (1999).
18. K. Itami, et al., J. Nucl. Mater. **290-293** 633 (2001).
19. G. Federici, et al., J. Nucl. Mater. **266-269** 14 (1999).
20. A. T. Peacock, et al., J. Nucl. Mater. **266-269** 423 (1999).
21. G. Janeschitz, et al., J. Nucl. Mater. **290-293** 1 (2001).