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An Overview of Requirements and Status

Kenneth M. Young

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FS&T Chapter 1

Plasma Measurements: An Overview of Requirements and Status

KENNETH M. YOUNG*

Princeton Plasma Physics Laboratory, P.O. Box 451, Princeton, NJ, U.S.A.

Abstract

This paper introduces this special issue on plasma diagnostics for magnetic fusion devices. Its primary purpose is to relate the measurements of plasma parameters to the physics challenges to be faced on operating and planned devices, and also to identify the diagnostic techniques that are used to make these measurements. The specific physics involved in the application of the techniques will be addressed in subsequent chapters. This chapter is biased toward measurements for tokamaks because of their proximity to the burning plasma frontier, and to set the scene for the development work associated with ITER. Hence, there is some emphasis on measurements for alpha-physics studies and the needs for plasma measurements as input to actuators to control the plasma, both for optimizing the device performance and for protection of the surrounding material. The very different approach to the engineering of diagnostics for a burning plasma is considered, emphasizing the needs for new calibration ideas, reliability and hardness against, and compatibility with, radiation. New ideas take a long time to be converted into "work-horse" sophisticated diagnostics so that investment in new developments is essential for ITER, particularly for the measurement of alpha-particles

*e-mail: kyoung@pppl.gov

1. Introduction:

Measurement of plasma parameters has always been key to advancing our understanding of the plasma performance and hence to the progress towards our building and operating a burning plasma device. In magnetic fusion, ever more detail in the measurements is being called for, as the interpretive power of the theory and modeling increases and as technological advances are made in the enabling hardware. This special edition of *Fusion Science and Technology* is devoted to describing the diagnostic techniques that have been developed and are now being applied to make these measurements.

Understanding the behavior of a high-temperature ($\geq 10^7$°K (~ 1 keV)), relatively low-density plasma ($\sim 10^{19} - 10^{21}$ m^{-3}), confined in a strong magnetic field ($\geq$ 1 T), calls for measurement of the spatial and temporal dependences of a large number of plasma parameters. The electron density and temperature, the ion temperature, the line and continuum radiation emitted by the plasma, impurity concentrations, and magnetic field strengths are some of the key parameters to be measured for the core plasmas. (More extensive lists are given in the Tables I through VI.) Rapid temporal variations in these parameters indicate the presence of instability of many possible kinds. Identifying these instabilities and relating their presence to the plasma transport across the magnetic fields has been one of the challenges in plasma physics for very many years. The transport of the different particle species can now be modeled extensively with respect to different hypotheses; comparison of profiles (radial dependences) of modeled with actual physical parameters is now commonplace. The interactions of the high-temperature plasmas with cold walls lead to edge regions for which a similar set of measurements is necessary. When plasmas, self-heated by alpha-particles from fusion reactions of the fuel ions, become available, it will be necessary to follow the behavior of these alpha-particles.

The diagnostics that have been developed to make these measurements cover a very wide range of physics. Electromagnetic emissions from the plasma are used over the whole spectral range extending from the radio region for magnetic fluctuations, through the microwave region for electron temperature measurement, the visible and ultra-violet for spectral identification and ion temperature, the x-ray region for ion temperature and high-Z impurity identification and into the γ-ray region for fast electron bremsstrahlung and confined α-particle studies. Reflected and scattered microwaves are used for electron density, fluctuation and confined alpha-particle studies. Transmitted infrared radiation and microwaves are used in interferometry and polarimetry to determine the electron density and magnetic field. Scattering of visible or near-infrared laser light by the plasma electrons is used for determining the electron temperature and density.

Particle measurements also play a significant role. Escaping neutralized fuel-ion particles can be used for determining the ion temperature. Emitted neutrons yield information about the fusion source region and the temperature of the ions. Neutralized alpha particles yield information about those that are confined while charged alpha particles on loss orbits can be measured at the walls. Often the spectroscopic and particle signal strengths are enhanced by an injected source of neutral particles, neutral beams or solid pellets, to increase the neutral particle density in the plasma core, thereby enhancing the probability of charge-exchange reactions. If neutral beams are injected, following the beam particles themselves can yield data about the local magnetic field through their Doppler-shifted polarized emission and about the density fluctuations through variations in their emitted spectral intensity.

The different physics processes involved in the measurement techniques explain the clear need for complementary diagnostic systems. Hence a very large array of instruments is currently assembled on the typical fusion device to carry out the measurements. There can be subtle differences between the results obtained for the same nominal plasma parameter measured by different techniques. For instance, there was a consistently growing departure between the magnitudes of the electron temperature obtained by Thomson scattering and by electron cyclotron emission (ECE) as the temperature increased in the large tokamaks [1]. Ultimately this departure could be assigned to the effects of non-Maxwellian electron distribution functions, caused by strong auxiliary heating [2]. The measurement of temperature using Thomson scattering depends on the Doppler-broadened scattering of an incoming light wave by the electrons of the plasma. The measurement by ECE relies on emission by the electrons. (For cold plasmas, Langmuir probes can also be used.) If density is considered, the intensity of Thomson scattering is dependent on the electron density, while interferometry relies on the density dependence of the refractive index of the plasma and reflectometry depends on the cut-off in microwave transmission due to the density. (Probes depend on the behavior of the current drawn to the probe as the bias voltage is varied.)

Complementary diagnostics are also provided because of their differing operational capabilities. Some diagnostics provide localized information, e.g. Thomson scattering, ECE, reflectometry and probes, while others, e.g. interferometry, integrate across the plasma, so that for spatial resolution, a number of sight-lines are needed. Some methods are continuous in time, e.g. interferometry; some are pulsed, e.g. Thomson scattering (because of the high-power laser required) and some are swept in time, e.g. ECE, reflectometry and probes. Also the ability to operate many diagnostics is constrained by the plasma conditions. Probes can only survive in the extreme outer edges of tokamak plasmas; ECE is not possible at low toroidal magnetic field strengths and can be handicapped by harmonic overlap at low aspect ratio in higher field devices. Reflectometry in the X-mode is limited in high magnetic fields by the availability of sources. Available spectral lines are strongly density and temperature dependent. It will be particularly important to make use of many diagnostic techniques in the next-step devices where key parameters are predicted to depart significantly from the accustomed ranges and where the invested value per shot is so large.

Progress in magnetic fusion research toward steady-state devices with significant plasma power handling and to long-pulse burning plasmas places further demands on the diagnostic measurement capability. Diagnostic signals are necessary for the machine safety issues generated from the high power density, through the role the measurements play in controlling the plasma. The diagnostics will provide the basic information about the plasma status, which must then be transferred, after appropriate manipulation, to responding auxiliary systems. This information may be simple, e.g. the plasma is moving vertically, or complex, involving many diagnostics, e.g. the radial profile of β, a measure of the plasma pressure, compared to the theoretical limit in β derived from other parameters. This profile information for a burning plasma may be critical in protecting the first-wall of the device by preventing the rapid loss of energy in the rapid termination of a tokamak plasma pulse known as a disruption.

Thus, every modern magnetic fusion device must be very well armed with diagnostics, such as the array on DIII-D [3]. This array is shown schematically in fig. 1. The mission of this particular device, to provide a strong physics basis for the next-step burning plasma devices, demands such an array of measurement capability. Most other devices have similar comprehensive armaments of diagnostics, exemplified in this series of Special Issues of *FS&T*. Papers on the overall attributes of the diagnostic complements have already appeared for DIII-D [3], FTU [4], TEXTOR [5, 6], JT-60 [7], JFT-2M [8] and Alcator C-Mod [9]. They provide a valuable compendium of the application of instruments and of observed data for real situations.

The primary mission of this chapter is to introduce the requirements now considered important for plasma measurements in all magnetic devices, but particularly for the next-step tokamak, ITER. The chapter does not attempt to address the physics issues of individual diagnostics that are well covered in the literature (e.g. refs 10 - 12) and in the subsequent chapters in this special issue. Because of the difficult integration of the diagnostic hardware with blankets and shielding in ITER from the outset, it was necessary to define the quality of information about the plasma parameters necessary for understanding, and ultimately controlling, the plasma. Hence, those requirements are detailed in Section II. Also shown are the diagnostics predominantly used today to make these measurements, with pointers to the relevant chapters in this issue where their physics attributes can be found. Note that other diagnostic techniques can provide valuable supporting information. Section 3 examines a different aspect of these measurements and identify those that are expected to feed into the control systems of future devices. Since the stored plasma energy in these devices is so high and could do extensive damage if dumped too fast onto the first wall in a disruption, real-time control of the plasma properties will have become essential. Section IV provides a brief introduction to the challenges of ensuring that the data are of the highest quality through ensuring self-calibration, reliability and robustness of the measurement equipment. Section V is a brief practical discussion of the novel engineering issues created by the hostile environment of a burning-plasma device. (Chapter 12, "Generic Diagnostic Issues for a Burning Plasma Experiment" by G. Vayakis et al., in this special issue of *FS&T*, will give a much more detailed assessment of the challenges.) Section VI concludes with some general comments.

II. The physics need and the measurement techniques:

This section addresses the measurement requirements for developing understanding in specific aspects of the physics of plasma confinement devices. Many of the most-often applied diagnostic techniques will be identified, though other techniques may also be applied. The following areas will be approached separately (accepting considerable interaction between them and that some diagnostic techniques will appear more than once): confinement and transport, instability and turbulence (including MHD), edge-physics (including divertor physics), and energetic particle and burning plasma physics. In addition operational systems, that is those necessary for operation of the device including principal magnetic diagnostics, will be outlined. The first four areas of interest follow fairly closely the agendas of the international groups who have been developing projections for the performance of future devices under the aegis of ITER-EDA and, more recently, of the International Tokamak Physics Activity. Additionaly, section II.F addresses the need for systems, such as diagnostic neutral beams, to enhance the range of capability of many diagnostics for spectroscopic and particle measurements. Each section will be used to point to the other chapters of this special issue in which the physics basis and practical details of the measurement techniques will be presented.

A table is shown in each section to bring out the main points of the measurements. The requirements presented have mostly been derived from the most recently defined set prepared for ITER [13]. The resolutions for measurements are considered to be those necessary for the physics requirements and therefore set clear goals for the instrument fabricators. It is already clear that achieving these goals will be a serious challenge for many diagnostics in ITER, with its high fluxes and fluences of neutrons and the necessary thick shielding. Most of the requirements are not exceptional for current tokamaks and extrapolating from them should be possible for any other device.

II.A. Confinement and Transport in the Plasma Core

The quality of the plasma confinement observed in magnetic fusion devices has been given a variety of names that are a useful descriptive shorthand for comparison of behavior between devices and for projecting the performance of future devices. Such names as L-mode, supershot, H-mode, ELMy H-mode, RI-Mode and reversed-shear advanced tokamak (AT) plasmas bring instant recognition for cognoscenti. They are all definitions based on the observed plasma parameters which reveal the confinement performance.

Categorizing the plasma behavior by name does not necessarily indicate full understanding of the performance. For that, extensive theoretical work and modeling based on different theoretical inputs are being developed. These detailed analyses require extremely accurate experimental input data for many of the plasma parameters. There are many challenges in the defining of very steep spatial parameter changes, requiring ever better spatial resolution. Also the global confinement, observed within each of the named categories, to be used in extrapolating to future performance, requires accurate integrated values of the parameters. An example of such data is shown in fig. 2 where both axes are dependent on integrated products of observed data.

However, the experimental situation is much more complex than can be simply summarized through this or any of the other data-bases currently being developed. Considerable effort is required to obtain the kind of good plasma performance entered into these data-bases. For each plasma pulse, some preparatory wall conditioning and various kinds of heating power are applied, probably carefully programmed in time, to bring up the performance. Fueling must be controlled, impurities must be limited, and plasma current and rotation may be applied. Hence very many parameters must be measured to inform the physicists as they aim for best performance. The necessary measurements characterizing confinement behavior are elaborated upon in Table I.

The quality of the plasma in its core must be determined through knowledge of the magnitudes and profiles of the electron density and temperature and the ion temperature. (For the purposes of this discussion, the core is assumed to be inside the radius, $r < 0.8a$, where $r = a$ indicates the plasma boundary and a plasma with a circular cross-section is assumed.) Tight spatial resolutions are necessary for accurately defining the transport barriers and the mode in which the plasma is operating. Much is to be learned from measurements of the current density. The internal magnetic configuration set by the current density distribution, or the safety factor q, plays a significant role in the behavior of AT plasmas. The contribution to the plasma current made by the pressure gradient through the bootstrap effect requires accurate profile measurements. Since it is predicted theoretically that shear in the rotation can have a strong stabilizing effect on instability modes, the spatial dependence of rotation of the plasma is another key measurement. The radial electric field is often inferred from the measurement of these kinetic parameters, but it would be preferable to measure it directly.

The performance can be drastically affected by impurities so that they must be measured and controlled. Usually line radiation and the overall radiated power levels are measured. On the other hand, small quantities of impurities can benefit the measurement capability for spectroscopic diagnostics where a charge-exchange reaction results in emission of a visible spectral line; its line width or Doppler-shift can be measured, e.g. in charge exchange recombination spectroscopy (CXRS) for ion temperature or plasma rotation respectively. An assessment of the effective charge of the plasma, Z_{eff}, must be made because of its impact on the plasma conductivity. Many studies of the transport of impurities in plasmas have been made, particularly of the transport of helium in hydrogenic plasmas because of its importance as a fusion-reaction product. This measurement will be more critical for burning plasmas because of the helium "ash" from cooled alpha particles being a potential damper of fusion reactions.

Many of the more recent improvements in confinement, particularly AT-modes, have resulted from carefully controlled applications of the heating systems with very close monitoring and feedback by diagnostic measurement [15]. This type of linking of transport study and control will become more highly developed as the magnetic fusion program moves to burning plasmas at the limits of the hardware capabilities.

II.B. Instability and Turbulence:

Departures from the predicted cross-field neo-classical transport are known to be due to the presence of plasma instabilities and there has been an extensive effort over most of the fusion program to validate this conclusion. Hence, measurements of fluctuating parameters in the plasmas have been a source of constant interest to try to identify the observed behavior with a relevant theoretically predicted mode, or, in a few cases, to try to relate the observed confinement to the transport caused by the observed mode (e.g. for particle confinement [16]). The observations range from relatively large-scale, low-frequency magnetic field fluctuations to low amplitude, very high-frequency turbulence. Very high-k turbulence (e.g. due to electron-temperature-gradient (ETG) modes) can theoretically contribute to energy transport (with $k_{perp}\rho_e \sim 1$, where k_{perp} is the perpendicular wave-number of the turbulence and ρ_e is the electron gyro-radius). A simplified listing of the measurements necessary for defining these instability modes is given in Table II. It also includes identification of diagnostics presently used in following the particular fluctuating behavior. Some estimates of the actual scale of the fluctuating signals to be measured have also been included. These small amplitudes make very serious demands to minimize the background noise levels inherent in the measuring equipment. Equally the observation of turbulence to ever higher frequencies stretches the capability of the equipment.

Unfortunately, measurements of electric-field fluctuations, necessary in quadrature with density fluctuations to cause cross-field particle transport, are very difficult in high-temperature plasmas, so that much of the interpretation of the role of the fluctuations in enhancing transport is tied to theoretical expectation. As an example, complex modeling might project that an applied plasma rotation should stabilize some modes and the confinement is improved and some diminution in the fluctuations is observed. Some macroscopic transport deteriorations have been observed due to the presence of large-scale fluctuations and stabilization of these modes has shown improvement in the confinement. In particular, RWM (resistive wall modes) and NTM (neoclassical tearing modes) have been greatly reduced by stabilizing external coils and localized electron cyclotron heating respectively [17, 18]. Also, while sawteeth magneto-hydrodynamic (MHD) modes have been studied for many years, the diagnostic capability for imaging the thermal transport locally has only recently become available [19]. Other large-scale MHD modes, possibly associated with minute defects in the externally-provided magnetic geometry, may result in slowing or stationary perturbations causing rapid plasma loss. Thus the measurement of these modes is important in the maintenance of the plasma quality and feeds into the control system. MHD fluctuations can grow and lead into disruptive termination of the discharge, so continuously following the behavior of these modes is essential. The energy available in the disruption of a burning plasma device could potentially cause major damage of first-wall components, especially the divertor, and so identifying warning precursors is an urgent need. Whether it will be possible to avoid disruptions by using measurements of MHD activity, or whether the avoidance of such events will only be possible from measurement of the plasma beta is still an open question.

There have been direct measurements of the transport of fast ions due to instabilities in the plasma, notably trapped alpha particles by sawteeth in TFTR [20]. Other MHD activity and kinetic ballooning modes (KBMs) at much higher frequency have led to direct loss of high-energy particles and even ion cyclotron resonance frequency (ICRF) heating waves can lead to loss of alpha particles [21]. Fast-ion-driven ion cyclotron emission (ICE) seems only to be an indication of the fast ions or alpha particles passing through the low-density edge plasma on the low field side [22].

In developing an understanding of the impact of the observed fluctuations on the plasma behavior by comparison with theoretical modeling, many of the measurements described in section II.A. are obviously required. Even now, there are new observations and theoretical projections of new instability modes, mostly in the high frequency Alfvén eigenmode (AE) range, which may require additional observational techniques. Turbulence theory and measurement capability are moving forward rapidly. Scattering techniques are now being applied to turbulence with very small spatial scales, concurrent with theoretical predictions of instabilities and their impact on transport. The frequency range of the measurements is also increasing with expectation of fluctuations beyond 1 MHz. Several of these modes spaced across the plasma radius could lead to significant energy transport. Studies of 2-D turbulence structure are now possible in the outer regions of the plasma using beam emission spectroscopy (BES).

Edge localized modes (ELMs) of different frequencies and magnitudes are observed, frequently spectroscopically, in the outer regions of the plasma. They are particularly apparent in H-mode plasmas with rather flat density in the core region. During each cycle, a relatively large amount of energy can be deposited at the separatrix strike-points on the divertor surfaces, leading to concern about material ablation. Careful controlling of these modes will be essential to limiting the heat loads.

II.C. Edge Physics: Issues for the plasmas in the edge and the divertor:

In devices such as tokamaks and stellarators, plasma diffuses out of the core until it crosses the last closed magnetic flux surface (the separatrix) beyond which it moves primarily along field lines to the wall. This edge plasma is frequently diverted by shaped magnetic fields to terminate on wall surfaces a little distance from the main plasma. This limits the return to the main plasma of neutralized plasma or of impurities newly generated from the wall. The plasmas in the edge region (sometimes called the scrape-off layer or SOL) and in the divertor are known to play a critical role in the performance of the high-temperature plasma core. The condition of the wall surface can modify the core plasma behavior significantly; it is important to learn how this happens through the edge plasma across the separatrix. Most of the fueling and applied auxiliary heating of the core plasma interacts strongly with this edge. Modeling of the edge plasma is very complex because of the geometry and of the additional atomic processes occurring in these regions relative to the core. The wall is strongly affected by the edge plasma through deposition and desorption of neutral gas, sputtering and radiation. Measurements of the plasma in the divertor region tend to be difficult because of their very wide ranges in density and magnetic fields and the relative narrowness of the plasma around the separatrix. Access to this plasma for measurement is usually poor. The concept of detaching the plasma from the divertor surface in tokamaks, possibly by injecting a noble gas, in order to reduce the power flow to the divertor, adds further geometric complexity. Also, the reality of experimental studies may cause the edge to move large distances, compared to its width, as the plasma beta, elongation or divertor configuration might be changed.

Table III gathers together a list of plasma measurements for the edge and divertor regions. Many of the parameters are the same as shown for the core in Table I, but usually the spatial specification is tighter because of the steeper gradients expected in the edge. For ITER and FIRE [23] a geometric edge region has been loosely defined as $r/a \geq 0.8$ for diagnostic designers; for elongated plasmas the width can extend beyond $r/a = 1$. Modelers of the behavior of the plasma refer to the pedestal region rather than using the geometric dimensions. The specifications shown are those developed for the proposed burning plasma device, FIRE.

The edge region always has a very complex geometry that leads to a need to make measurements of the same plasma parameter in many locations. Pedestals in the edge electron density and temperature can be measured at the device midplane but profiles are also desirable at the X-point and across the separatrix of the two diverted legs in the diverted plasma. Such information is needed for modeling the plasma. Line radiation of the fuel atoms and ions and impurities gives information on the densities of these particles; sometimes the measurements are assisted by laser-induced fluorescence (LIF) but it is difficult to apply at many locations (chapter 5). Probes, just protruding from the wall surface, can be used to determine the plasma density, temperature and flow very close to the strike point on the divertor. In cases where heat loads to plasma-facing components such as the divertor strike points might become excessive and are to be limited by local injection of an impurity gas, the control depends on spectroscopic measurement. Edge turbulence (see section II.B) can be made visible by injection of puffs of gas into the edge and detected by a fast camera, viewing along the edge field lines.

Because of the strong interaction of the plasma with the walls of the containment chamber, it has become customary to consider the monitoring of the behavior of the wall as part of the edge diagnostic set. Temperature of the surface, either determined slowly by thermocouple or more rapidly by infrared camera, can provide a warning of excessive heat loads. Even visible-light cameras can show undesirable interaction of the plasma with the wall. It is evidently easy to cause erosion (and redeposition) of surfaces, particularly where the preferred graphite materials have been used for the plasma-facing components of the wall and this process creates dust. The dust normally has little importance for plasma operation.

For burning plasma devices, the ability of the edge and divertor to process the tritium and the cold helium residue from the fusion reactions must also be understood. Tritium retention in wall materials (much higher for graphite than for high-Z metal walls) can drastically limit operation while helium must be transported away from the plasma core to maintain the burn and its production. The dust in these devices gains in importance: the relatively high levels of erosion can lead to large quantities of dust in the vacuum vessel and this could be a radiological hazard in the event of an accident. Because of the high surface area of the accumulated dust, tritium attachment may even become an issue for the tritium inventory in the device. A measurement of its rate of production is therefore needed and limits may be set on how much can be permitted in the vacuum vessel (chapter 12).

II.D. Energetic Particle and Burning Plasma Diagnostics:

Fast particle studies achieved full importance with the production of significant numbers of fusion product particles [20]. This section will concentrate on the measurements necessary for the study of burning plasmas [24]. Measurements of fast ions generated by auxiliary heating systems, such as neutral beams or radio frequency resonance heating, are essential for understanding the heating and transport processes of these particles. For burning plasma devices, these measurements provide validation of the performance of individual alpha particle diagnostics prior to fueling with D-T.

Diagnostics for burning plasmas must provide the quality of measurement expected for current devices for those measurements discussed in the three previous sections. They must also provide information about the production of alpha particles and their behavior in setting up sustained burning fusion plasmas. Since these very fast particles may well drive high frequency instabilities, potentially damaging to the confinement of themselves and the plasma, a very good understanding of the instabilities is needed. This section will describe the challenges for making the measurements for the burn phase, but will leave the discussion of the technological challenge of operating in the necessarily high radiation environment to section IV. Understandably there has been little investment in diagnostics for this operational regime, but it has now become an urgent necessity.

Some alpha particle diagnostics were conceived for the D-T campaigns in TFTR and JET and provided a remarkable amount of information about the alpha particle behavior and its relation to instabilities [25]. These diagnostics are named in table IV as part of the potentially available diagnostics. The array of neutron diagnostics performed well with the change in flux of two orders of magnitude in going from D-discharges to D-T discharges easily accommodated. The newly developed alpa particle diagnostics were very successful but had clear limitations, and the programs did not last long enough for remedies to be developed.

Consider first the measurements of confined alpha particles. Microwave collective scattering was not effective in either TFTR or JET and it is only recently that it has been demonstrated as a robust diagnostic on TEXTOR [26], where the primary study has been of the fast-ion particle redistribution during MHD activity. The shorter wavelength CO_2 scattering has also been developed and is awaiting test on JT-60U [27], but its need for very small-angle scattering makes its installation problematic, and the achievable spatial resolution very poor, though the hardware components are relatively easily available. The spectroscopic technique, alpha-CHERS, makes use of a very precise measurement of the tail of the 568.6 nm helium spectral line seen in the presence of a neutral beam. The measurement entails a very precise comparison with the visible-bremsstrahlung background, but yields information on the α-particle distribution function only up to about 1/4 of the beam energy [28]. Unfortunately in TFTR the fiberoptics taking the image from the tokamak to the instrumentation was not sufficiently shielded so that a neutron-induced fluorescence was superimposed on the bremsstrahlung and could not be removed with sufficient accuracy. Hence most data was taken immediately after the neutral beams were turned off (the alpha particle slowing down time was very long compared to the beam particle slowing) and the neutron source greatly reduced.

Another technique, called pellet charge exchange, makes use of lithium pellets fired radially across the plasma. The alpha particles collide with the low-Z atoms in its very dense ablation cloud to become neutralized and escape to a neutral particle analyzer, which viewed very nearly along the pellet flight path in the plasma. This system provided good energy distributions and spatial distribution (making use of the time-of-flight of the pellet) [29]. This system again only functioned well in a plasma pulse after the beams were stopped. In this case the reason was the failure of the pellet to penetrate deep into the plasma while the beams were on.

The high-energy alpha particles collide with the fuel ions accelerating them and producing a small high-energy tail of the fuel ions. These ions contribute to a very small (only $\sim 10^{-3}$ of the total DT neutrons have energy above 15.5 MeV under ignition conditions) high-energy (so-called "knock-on") tail in the neutron energy spectrum. Källne et al. successfully did archaeology at JET on the neutron spectra at very high energy and found sufficient counts integrating over many full discharges to get a single energy distribution of the alphas [30]. Meanwhile Fisher had tried to get a single-shot measurement of this high-energy part of the spectrum using bubble chambers with very steep low-energy cut-offs but failed because of issues with the detectors [31].

Measurements of the confined alpha particles by gamma spectroscopy have proven very effective in JET under circumstances where the neutron background (and, therefore, scattered gamma background) is not large. Such conditions exist where ICRF is the dominant heating technique and there are impurities with high (α,γ) cross-sections in the plasma, e.g. Be^9 [32]. This technique was of particular value in preparatory studies using trace quantities of tritium.

A rapid prompt loss of alpha particles can be expected in devices with low magnetic field strength. Hence the studies in TFTR of anomalous losses and their association with ripple diffusion and instabilities were of particular value. Scintillator probes, with scintillators mounted behind defining access slits under thick graphite mushroom-shaped tile protection from plasma-induced heat loads, were mounted in the lower outer quadrant of the first wall. These scintillators were imaged onto high-purity quartz optical fibers which took the emitted light to distant cameras or photomultipliers. Figure 3 shows a schematic of the detector (without its cover and protection) and an experimental image seen by a camera. The spatial arrangement of the light can be interpreted in terms of energy and pitch-angle of the incoming particles [33]. Another technique makes use of a Faraday cup concept with thin foils behind an entrance aperture. An early attempt on JET with this system, which can identify the energy, but not the pitch-angle, of individual particles, failed because of its being mounted too far back relative to the first wall [34] and a recently installed system is now operating [35]. Fast ion losses, He-ions accelerated by ICRF impinging on the wall, have been measured by IR cameras on JT-60U [36], but here the measurement is necessarily quite slow and integrates over the total particle energy lost. Recently a technique using activation foils has been tried on JET [37].

A serious concern is the impact that alpha particles have on the plasma stability, and conversely how the instabilities affect the fast-ion confinement. Low frequency MHD modes, such as fishbones and sawteeth, were found to affect the confinement of the fast alpha particles in TFTR [21], and potentially greater losses could be expected with the alphas driving them. Because of their potentially key role in the confinement of the plasma and the alpha particles, the high-n kinetic ballooning modes and Alfvén-type instabilities [20] mentioned in section II.B have also been included in table IV. An interesting extension of the study of these modes has been conducted on JET, where the stable modes can be driven unstable by an external coil and their growth and damping mechanisms can be studied [38].

II.E. Operational Systems:

There are several essential measurements which must be made in order for the magnetic fusion device to operate at all, or to provide the operators with a simple sense of the performance. These are summarized in table V. The quality of the vacuum in the various sections of the vacuum vessel, the divertor and the pumping ducts during pumpdown, discharge cleaning and conditioning and operation must be monitored continuously. Imaging in visible-light of the plasma and first-wall gives operators some confidence in the operation.

A fundamental aspect of establishing plasmas in any magnetic fusion device with satisfactory magnetic configurations must be ensuring good magnetic diagnostics (chapter 2). These diagnostics provide the essential data on the magnetic field strengths, the plasma shape and position, the loop voltage provided by the driven current, and the integrated plasma pressure. They also provide fundamental input to the analysis and modeling codes used in understanding the physics. Often the small loops used for defining the plasma equilibrium can be applied to instability studies, particularly of the low-frequency MHD modes.

The presence of runaway electrons has played some role in the start-up of the plasma and the relatively low-energy electrons escaping to hit a local wall have been measured by X-ray systems outside the vacuum vessel. A non-Maxwellian tail can be discerned by ECE systems. Much higher energy, very damaging runaway electrons might be produced in the thermal quench stage of disruptions and their measurement is much more challenging. Their bremsstrahlung emission will be tightly focused in the forward direction and discriminating their emission from the scattered gammas present in a burning plasma device may be difficult. Techniques for suppressing these electrons, e.g. by injection of high-density gas, have been envisaged, but the measurement used to trigger the suppressor will be of some other parameter.

The last system shown in table V, vacuum vessel illumination, is only of value between plasma operations when the capability of the first-wall imaging system can be exploited for inspection of the wall for defects. While clearly not a plasma diagnostic enhancement, it is associated with the diagnostic set because of its relationship to the visible cameras. They provide the possibility of a visual inspection of a large fraction of the first wall without venting the vessel to atmosphere. The illumination has to be very bright, relatively uniform in coverage of all the first-wall surfaces and fully vacuum compatible. Moreover the lamps must be covered and protected behind the first wall during the plasma operation. These systems first became essential with the large–sized tokamaks whose components became activated even with deuterium discharges and human access was no longer immediately possible [39].

II.F. Systems in Support of Active Diagnostics:

Over time, many diagnostic techniques have evolved such that their best operation is dependent on externally provided signal enhancers. The requirements and the purposes of these support systems - diagnostic neutral beams (and heating beams which have often been used), lithium beams, impurity-pellet injectors and gas-puffs are briefly described here. They are also highlighted in table VI. Heavy-ion beams and launched waves, such as those used in Thomson scattering, interferometry and reflectometry, are not described as they are considered an intrinsic part of the measuring equipment.

Many of the sophisticated and essential spectroscopic diagnostics depend on injected neutral particles to provide sufficient signal strength at the core of the large high-temperature plasmas. The natural fuel species and low-Z impurities are fully ionized for most of the plasma volume. There are high background signals. The electrons emit a broad visible bremsstrahlung radiation from this region and any spectrometer must look through the plasma boundary region with its radiating neutral and low-ionized-state particles. Dedicated diagnostic beams have not been very successful at enhancing the neutral population at the plasma core, principally because of inadequate intensity (signal strength) or energy (penetration), usually as a result of limited budgets. The most successful diagnostic exploitations have used the high-density hydrogenic heating beams oriented tangential to the plasma axis. In general, the positive-ion-based neutral beams provided for heating the plasma work well for plasma diagnostics in present-day devices. The beams penetrate well to the plasma center and provide good signal strengths. Since they are targeted at heating the plasma and not at diagnostic applications, they tend to be wide spatially, limiting fine-scale spatial resolution for an observing system, and cannot be modulated at a high repetition rate, helpful for enhancing the signal-to-noise ratio in a spectrometer. Beams oriented radially to the plasma penetrate more easily to the plasma axis, but evidently cannot be used for motional Stark effect (MSE) measurement.

The workhorse charge-exchange spectroscopy (CXRS) for the ion temperature, impurity density, plasma rotation, radial electric field (indirectly) and slowing-down alpha particle distribution function has worked well using heating beams. The use of the neutral particle analysis (NPA) technique was revitalized by the enhancement arising from neutral particles provided by the beam in the core. The measurement is of the plasma particles in both cases. The MSE measurement of the internal magnetic field structure makes use of the polarization due to the Stark-effect on the emission by the neutral particles in the beam itself for its signal source. Perturbations of the neutral light intensity emitted by the beam atoms caused by the fluctuations in the main plasma electron density, particularly in the outer regions of the plasma are the basis of the BES technique for evaluating edge fluctuations

The optimum energy of the beam atoms for the atomic physics processes in the plasma in ITER is calculated to be ~100 keV/amu, which is low for good penetration of the plasma and high for optimizing the neutral flux emerging from the beam [40]. Since this is a much lower energy than that of the heating beams, a dedicated beam has been proposed for the diagnostics. A very high source current density has been specified for this diagnostic beam, at least one order of magnitude higher than has been obtained to date in any source making use of negative-ions. A beam using a positive-ion-based source might be considered, but it also requires considerable extrapolation in neutral current density from that presently available. A novel suggestion of using a high-current, repetitive short-pulse-length beam has been proposed and some development has started [41].

Higher energy beams, such as the tangential heating beams proposed for ITER at 0.5 – 1 MeV, should work well for MSE and BES because of the large effective electric field and good beam penetration to the plasma center. There may be an issue if, as anticipated, the current-density profile information provided by MSE is to be fed into the control system where the heating beam is one of the plasma-control actuators.

Another kind of beam has been developed for diagnostics of the plasma edge and in the divertor region. The driver here is the need to improve the spatial resolution relative to that possible for the diagnostics used in the plasma core. Lithium-beams have been used to allow measurement of the electron density in the edge from the intensity of the line-radiation as the lithium penetrates through the plasma (JET) and also, with higher neutral-current density for observation of the Zeeman components of a spectral line. This latter measurement provides a much better spatial resolution of the plasma current-density in the outer regions of the plasma [42] than is possible using MSE.

Lithium injection was also developed on Alcator C-Mod, but in pellet form for a different purpose [43]. While the benefits of lithium pellets in conditioning the wall of TFTR became widely appreciated, the injector was installed to allow the measurement of the poloidal magnetic field using the polarization of one of the Li^+ spectral lines [44]. The spatial variation was obtained by following the position of the pellet. Injected lithium, boron or carbon pellets proved invaluable in the first measurement of the spatial profile and energy distribution of the fusion-generated alpha particles in TFTR [29,45].

One advantage of the large size of the TFTR plasmas is the ability to get quite fine relative spatial localization from the pellet motion, as is also true for studies of the fuelling pellets using spectroscopy in JET and other devices.

A very simple signal enhancement is provided by gas injection of fuelling or light impurity gases. The technique of gas puffing is derived from the type of gas injector installed on all devices. It is used especially for imaging of density turbulence in the plasma edge [46]. Often many small gas injectors are used to increase the coverage of fluctuations seen in the visible light of the neutral gas. High-Z impurity gases have to be injected to make core x-ray spectroscopy feasible. Injection of metals (e.g. lithium, germanium and titanium) for wall conditioning, impurity transport studies or to support x-ray spectroscopic studies in clean devices has made use of laser ablation [47].

III. Measurements for Plasma Control in Operating and Burning Plasma Devices

Plasma measurements have been used in feedback to control plasmas for a long time. Signals from the magnetic diagnostics have been used to control the current, position and shape of the plasma in tokamaks. Other diagnostics have provided permissive signals for auxiliary heating systems. More recently, the progress to operation with high plasma energies and the use of advanced operating modes has led to major advances in diagnostic use. These advances have enabled the extension of the plasma pulse length and improved plasma performance by constraining the approach to stability limits, by observation of instability modes and by following changes in the plasma profiles meticulously. Better performances, for example higher-β, can be achieved by operating in the AT regimes where the profiles of the plasma are manipulated in a controlled fashion by an external actuator such as electron cyclotron heating or oscillating magnetic fields. Thorough examinations of the need for control in ITER are available [48, 49].

It should be noted that some of the complex feedback scenarios for tokamak plasmas may not be necessary in devices such as stellarators with their different vacuum magnetic field configurations. However it is true that the need for plasma control has become more apparent with improvements in the plasma performance seen on the most advanced tokamaks [15]. It can be expected that the entry into plasma regimes where the heating is dominated by the alpha particles in a burning plasma device will provide new control challenges. An example is maintaining the location of the plasma during the very rapid rise in plasma-β occurring when the burn takes over.

Table VII summarizes the present expectation of diagnostic involvement in control of burning plasmas in a tokamak. It does not address the relatively few responding actuators such as auxiliary heating systems, active feedback external coils or fuel injection. For the goal of optimizing the plasma performance, including extended operation of full-power D-T operation, control of a wide range of parameters is required. The table separates the control requirement into four areas.

Establishing the plasma scenario already makes use of feedback in real time.

The maintenance of the operational scenario for long times is an active area of development. The plasma equilibrium, current profile, steep density and temperature profiles (transport barriers), impurity accumulation and fueling for long pulses must all be monitored. Rapid data-interpretive codes and neural network analyses are being applied now to make this control feasible. There will be new aspects for control in D-T plasmas where the fueling ratio must be maintained and where the build-up of helium as an ash must be limited, aside from the issues arising from the very rapid rise of β as the burn takes off.

The emergence of instabilities can drastically affect the confinement so must be constrained. Extensive studies of controlling lower frequency modes like NTMs [17], and RWMs [18] are in progress but it is conceivable that the higher frequency AE modes could be significant for burning plasmas with their fast-ion driving component.

The energy deposited by ELMs could be very damaging to the strike points of the divertor for the high-energy-content burning plasmas and hence their type and magnitude must be controlled. Disruptions at full energy must be prevented, though the control input is likely to be from a β measurement rather than from some precursor measurement of fluctuations. Also the divertor surfaces themselves must be protected from excessive heat loads so the plasma interaction with them is controlled. Hence detachment of the plasma from the strike point, by, for example, injecting a noble gas in the divertor, is being actively studied.

Access to the plasma of a final reactor configuration will be much more limited than for ITER. Hence it is to be hoped that the number of detailed plasma measurements involved in feedback will be much less for a fusion reactor. Some of the research in ITER must be devoted to this simplification.

IV. Calibration, Precision and Robustness of Diagnostic Systems:

The quality and reliability of the plasma diagnostics on operating devices is already extraordinary. But maintaining this quality is extremely labor intensive and has often only been achieved after an extensive period of adjustment and trouble-shooting. Now that the data is becoming more critical in plasma control, it is clear that the accuracy of the information, and its validity over long periods of time, and many plasma pulses, must be assured. Also, with so much data being fed directly into analysis codes, whose published output seldom shows any form of error discussion, it is important that the data has a high precision. It is interesting to note that the reliability of diagnostics on TFTR increased significantly when access to the tokamak became restricted for DT operation.

Calibration of many diagnostics has always provided interesting challenges. These challenges will become even more severe for ITER where its very long pulses will probably provide coatings of plasma-facing mirrors. The evolving high temperatures of the vacuum vessel and in-vessel components will change alignments. High radiation levels will dictate special choices of components to be mounted permanently close to the plasma-facing walls to enable through-the-system calibration. The type of before-run and after-run calibrations now used, often involving significant structures inserted into the vacuum vessel will have to be rethought in view of the very long "use-time" experienced by each diagnostic. In-situ, real-time calibration concepts must be considered. These concepts should include methods making use of specific reference plasma conditions and movements of the plasma for cross calibration.

The reliability of a diagnostic can normally be separated into three spatial segments of the diagnostic; close to the fusion device, the instrumentation and operations area, and the intermediate connection. Mechanical reliability usually dominates at the device end. The components must be designed to operate for very many cycles, preferably tested in a prototype, with all aspects such as magnetic field forces (including those caused by disruptions), temperature excursions and radiation (including remote handling) taken into account. These components are almost certainly one of a kind and must be engineered for ITER as though they were being sent to operate in outer space. Equipment in the instrumentation area consists mostly of commercial components and therefore tends toward reliable operation both of the hardware and operating software. But, for many-channel systems with many component duplications, one must still be sure to select high-quality parts. There are some cases, e.g. high repetition-rate multi-pulse laser systems, where determining and meeting the design specifications is critical. A planned maintenance and replacement program may be essential. For the intermediate regions, where most of the components will be passive, everything must be mounted securely to ensure correct alignment. Flexibility must be allowed where necessary. Everything possible should be done to protect the transmission, e.g. enclosing optical paths in boxes. Finally extreme care must be taken to ensure a good single-point electrical grounding arrangement for all the diagnostics such that there exist no huge ground loops which can introduce extraneous noise onto the instrument signals. If the grounding is not perfect, installation of any new equipment elsewhere on the device can introduce new noise into a diagnostic.

Robustness is a major contributor to reliability particularly for the components close to the device and intermediate region. The ability to maintain alignment through many magnetic-field, temperature and radiation cycles, and particularly the shock of disruptions, is critical to the diagnostic operation and must be designed into each diagnostic. Similarly wear and tear on components must also be considered, e.g. a mirror's response to millions of high-power laser pulses, in the design, prototyping and testing programs.

Diagnostic maintenance on TFTR was carefully planned and integrated with maintenance schedules of other systems, such as the tokamak itself, computer systems, the air conditioning and fire-protection. Because of the aggressive operational schedule routine maintenance could not be performed as readily or as often as on earlier smaller devices. Once D-T operation started and the tokamak became activated and less approachable, maintenance of diagnostic components close to the tokamak became much less frequent. For ITER the in-vessel components will be largely unmaintainable over extended periods of time, but a careful maintenance schedule for the other parts of the systems will have to be developed taking account of the maintenance requirements of the facility.

V. Engineering challenges for diagnostics for burning plasma devices:

The combined physics and technology mission of ITER leads to conflicting requirements for diagnostics. On the one hand, the diagnostic instrumentation is complex and requires good access to the plasma. On the other hand it has to be highly reliable and must not generate significant neutron leakage. The components in the vacuum vessel must be capable of being manipulated by remote-handling equipment [50]. The first attempts to meet these challenges were made on TFTR and JET with their D-T programs [51,52]. In those devices, the main physics understanding was developed in operation in deuterium only, and could then move forward to comparable D-T plasmas, in the absence of a few radiation-sensitive diagnostics. For example on TFTR, the diodes in the x-ray imaging system, initially extremely important in the study of MHD, saturated in the neutron and gamma fluxes from D-T plasmas without additional shielding. The situation should be better for a fusion reactor in that one expects that a simpler, and more robust, set of diagnostics will be necessary for the control implementation.

Most diagnostic components will be protected from the plasma by shielding. Some, however, will have to live in a very hostile environment of neutron and gamma radiation, relatively high temperature, high vacuum, high magnetic fields and mechanical shock from disruptions. The realization of new challenges led to a significant R&D program for ITER [53]. At the highest radiation levels, experienced by the magnetic diagnostics, bolometers, probes and some other components, the most serious impacts are on the prompt electrical behavior of insulating materials, not in long-term electrical degradation or mechanical failure. For ITER, and even more seriously for FIRE where the magnetic diagnostics are in higher neutron fluxes, the radiation-induced conductivity (RIC) and vacuum compatibility force selection of a very narrow range of high quality alumina insulators (and very innovative component design). Even for these insulators, the conductivity of the alumina is predicted to increase by about five orders of magnitude during the neutron production [54]. As described in chapter 12, other degradations in insulation effectiveness manifest themselves in components at high temperature and high voltage and in mineral-insulated cables essential for signal transmission. Active electronic and electrical components, such as photo-diodes for position verification or electrical motors to drive shutters, are very susceptible to radiation-damage and must be located behind thick shielding. Mechanical encoders (with radiation-hard wiring) and air-driven motors will have to be used where close-in application is required.

The technical progress in manufacture of fiberoptics has nurtured advances in imaging optical diagnostics. Unfortunately, even the best fused-silica fibers are very susceptible to neutron (and gamma) effects. During the pulse there is prompt fluorescence and increased absorption and there is a long-time absorption build-up [55]. An example of this fluorescence, at relatively low dose-rates in partially-shielded fibers on TFTR can be seen in fig. 3b. Under the peak in the light caused by the alpha particles, fluorescence causes the clear rectangular pedestal. To try to reduce the transmission loss in fibers, metal-clad heated fibers were tried on JET [56, 57]. More recently, the ITER partners have carried out extensive developments of special fibers [50]. Nevertheless, their front-ends cannot be close to the plasma. Because darkening and the same optical deficiencies as seen in fibers occur in transmission optical components close to the plasma, reflective optics have to be used near the plasma. These mirrors allow the use of labyrinths to preserve the shielding capability even though the penetration size is larger than would have been necessary for a fiber-optic based configuration. However there are serious questions of maintaining reflectivity and polarization information through the mirror train, especially as the front-end mirror may see coating or erosion caused by the nearby plasma [58]. This last issue is one of major concern to the designers of ITER diagnostics who have encouraged many test-programs on operating devices (chapter 12). It is clearly an issue for the component closest to the plasma of any optical diagnostic on a long-pulse device.

The environment leads to many other considerations for diagnostic implementation. The diagnostic must be designed for integration into very deep shielding, to reduce radiation on the coils and in the vacuum seal area, and so become essentially a part of the initial device vacuum vessel configuration. Clearly reliability must be built into, and demonstrated for, the design. All components involved in any devices for intentional motion for real-time alignment, shuttering or calibration must be designed for the local environment and a long life. The component must survive and operate under the most extreme situation expected. Since all the diagnostic parts and the surrounding materials close to the plasma will become highly activated, remote-handling must be built into the design.

The need to integrate front-end diagnostic components into labyrinths in shielding has lead to the concept of a large port plug, with incorporated diagnostic parts, to block the neutron streaming through the port and to seal the vacuum boundary for ITER. This ~ 50 tonne (for the equatorial ITER ports) precision component would be manipulated fully by remote handling. Remote handling was not used outside the vacuum vessel, where most diagnostic components were placed, for TFTR or JET. A few recent structural changes for diagnostics have been made inside the JET vacuum vessel by remote handling, but a very significant change in approach to the application of remote handling to diagnostics for ITER is necessary.

Beyond the fact of the fusion environment, the size and high densities of the plasmas and the magnetic field strengths lead to major extrapolations from present-day diagnostic equipment. Two examples of urgently needed research and development are sources for reflectometry, and a high particle density neutral beam for active spectroscopy. The development required for the latter is so extensive that new ways of determining many plasma parameters without using spectroscopy have been sought for some time without success. As indicated earlier, evolution of a completely new technique to the quality required for modern-day devices is a lengthy process.

VI. Final Comments:

The focus in this chapter has been on the reasons for making the plasma measurements together with providing some idea of the quality of measurement necessary to allow full analyses. Theoretical modeling of the plasma performance has reached the stage that individual measurements can be gathered together and compared to anticipated performance very rapidly. This has been facilitated by the huge advances in computer power, memory and analysis software. Much of the data will have been applied to feedback control of the plasma parameters through manipulation of auxiliary systems in real time. This need for control often sets the requirements for time resolution shown in the tables in sections II. Most of the more established diagnostic techniques have been listed. The physical principles of their operation and their implementation will be detailed in the later chapters of this special issue referenced in the tables. Many of the improvements in implementation that are continually being made will also be shown there.

The construction of ITER should provide a significant boost to the worldwide thinking about, and investment in, plasma diagnostics, similar to the surge caused by the starting up of the three large tokamaks, TFTR, JET and JT-60, in the early 1980s. The diagnostics for these devices were significantly more sophisticated than those in predecessor devices. It became imperative to obtain multi-spatial, multi-time data of many parameters during the same pulse because of the length of pulse and the inefficiency (as well as increased errors) of moving diagnostic sight-lines relative to the plasma between pulses. The shielding necessary for the D-D neutron emission (D-T neutrons in TFTR and JET) constrained ease of access significantly. These devices also provided an advantage relative to smaller devices in that relevant spatial resolutions were easier to obtain because of the plasma size and larger plasma scale-lengths; relevant temporal resolutions were made available by the greater pulse lengths and longer plasma time-scales. And, to some extent, the very large budget outlays required for the tokamaks and for their ancillary systems made it possible to spend money on developments in diagnostics. Additionally major advances being made at that time in digital technology were making the collection and rapid analyses and dissemination of data ever more feasible, a process which continues today. Physics analysis codes could be used between pulses in the control room to give guidance for the subsequent pulse, with more sophisticated transport codes being brought in for overnight and longer-term analyses. During the lifetime of the large tokamaks, feedback using digitized plasma data was used in controlling the plasma performance. In JET, JT-60U and DIII-D more plasma measurements are being continually added into the control systems.

New measurement techniques must be developed to improve the physics studies. Technological advances and the relatively cheap availability of huge amounts of digital memory make two-dimensional imaging of toroidal plasmas, with sufficiently fast time resolution, feasible. X-ray imaging in PBX-M to evaluate the current driven by the lower-hybrid heating system pioneered this field [59]. More recently, faster imaging capability provided information about the stability situation in TEXTOR, by determining the local behavior of the electron temperature with respect to sawteeth [19]. Imaging is obviously desirable for non-axisymmetric devices and is of particular value for stellarators.

For the large, high-density plasmas in ITER where lack of penetration by neutral sources constrains the capability of spectroscopic techniques, alternative ideas for diagnostics should be tried. Sufficient tolerance for failure in developing these ideas, at the worst, and of a long evolutionary period, at the best, must be gained by the relevant funding agencies. However the outstanding measurement performance of recent years makes the risks acceptable.

A word of warning is appropriate here, particularly for those people anticipating instant capability of alpha particle diagnostics. In the last few years less than a handful of genuinely innovative new diagnostic techniques have been installed on tokamaks. Evolution of a technique to the level of measurement capability necessary for use in control and physics modeling activities takes a very long time. Two good spectroscopic examples, charge-exchange spectroscopy (CXRS), for measurement of ion-temperature and plasma rotation, and motional Stark effect (MSE), for measurement of current density and radial-electric field profiles, each took around 10 years to reach maturity. From the first observation of highly excited impurity lines in neutral-beam heated plasmas in 1977 [60] through the first application of CXRS (see chapter 6) for ion temperature measurement in 1983 [61], to initial multichannel measurements in 1986 was nine years. The application for poloidal rotation profiles took longer and interpretations are still strongly dependent on excitation-rate issues. The MSE technique (see also chapter 6) was introduced on PBX-M in 1987 with a single optical system focused on a narrow rotatable (between-shot) diagnostic neutral beam [62]. It matured to a multiposition system with multiple sightlines using a heating beam on TFTR with spatial resolution of 3-5 cm and time resolution of ~50 ms in 1991 [63]. Increasing the number of sightlines allowed definition within a transport barrier and permitted electric field measurement [64, 65]. There are still very many challenges to installing and calibrating a working system on a high-field, high-density device or on a burning plasma device.

One important aspect of burning plasma devices is that the diagnostic equipment has to be fully integrated into the engineering of the device. Diagnostic components are incorporated into the blankets and innermost shielding and essentially become part of the vacuum boundary. The trouble-shooting of the instrument performance, physics experiments in themselves, must be carried out during the earliest hydrogen-phase discharges. The calibration techniques must be built-in and demonstrated before the activation of components makes modification awkward and costly. There will no longer be any financial sense (if there ever was) in delaying development and installation of essential instruments on a device until many years after it becomes operational.

This chapter has concentrated on the diagnostics relevant to the front-line toroidal experimental devices, and especially tokamaks. There are many other devices exploring different possible routes toward a fusion reactor. Many of the same measurement requirements and techniques also apply to these devices. To demonstrate the quality of the plasmas in these devices and to enable that quality to improve through better understanding, serious investment in new and improved instrumentation, and the associated analytic frameworks, must be forthcoming. New diagnostics which can be deployed on these, and on all non-burning plasma devices, will play a major role in developing an understanding of plasma turbulence and its relationship to transport.

Since this paper was first submitted an important review paper on diagnostics for steady state plasmas has been published bu Hartfuss et al. [66].

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List of Figure Captions

Figure 1. Schematic showing the multitude of diagnostics assigned to the ports on DIII-D. The lists in the center (above and below) are of diagnostics behind the center column or behind the camera. The arrangement shown is from 2004. (With thanks to R.L. Boivin.)

Figure 2. A summation drawn from many devices of the thermal energy confinement time to guide the expectation for ITER – data point at upper right (with thanks to B.J. Green for fig 2 from ref 14). The summation is for ELMy H-mode discharges.

Figure 3. a) A schematic drawing of the use of a scintillator for measuring escaping fast ions and α-particles on TFTR. Thick thermal protection and light covering must be incorporated. (With thanks to S.J. Zweben and *Nuclear Fusion* for fig. 7 of ref. 21.)

b) The image of the scintillator, showing the background fluorescence generated by neutrons and gammas in the fiberoptic taking the signal to the observation camera (with thanks to D.S. Darrow).

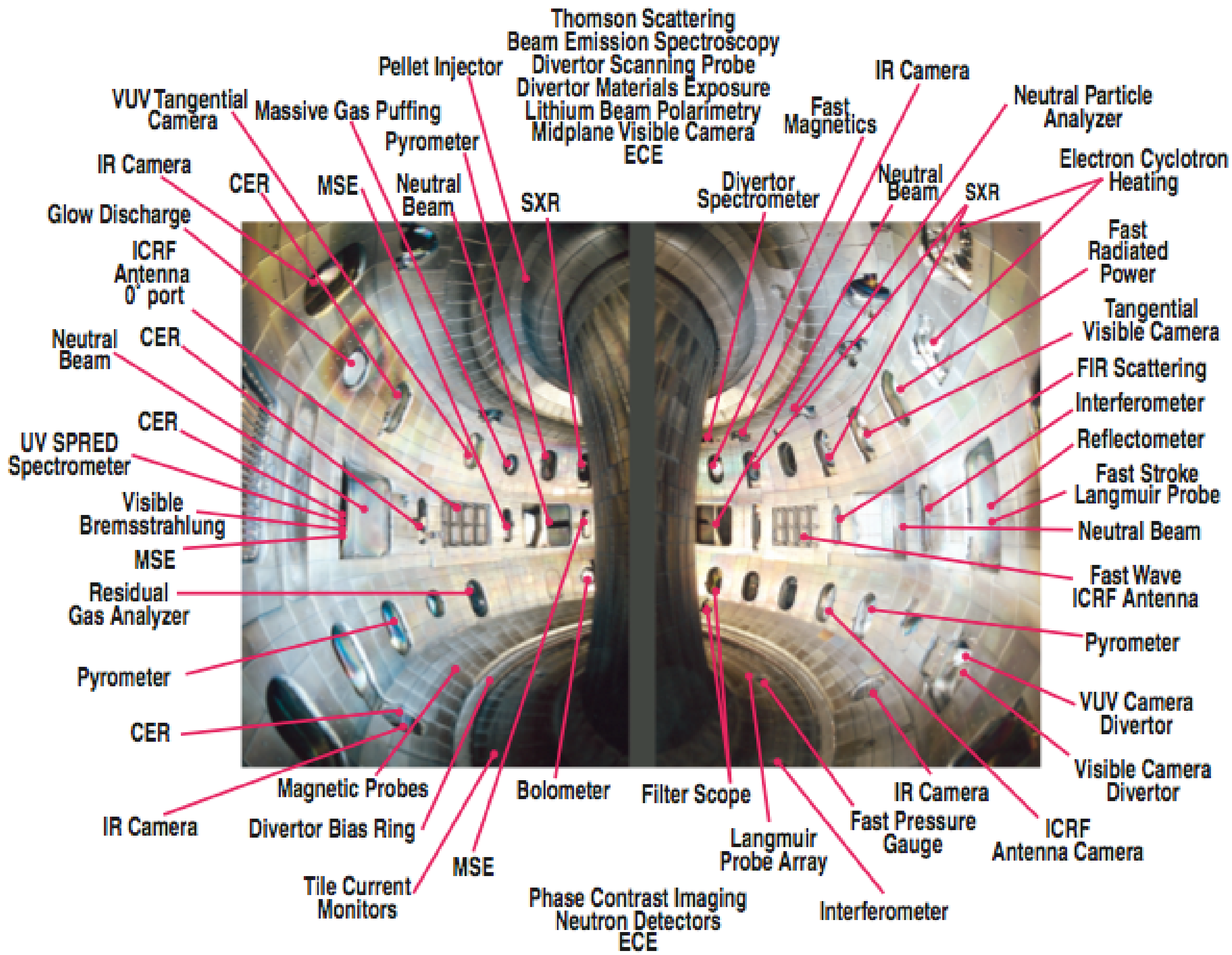
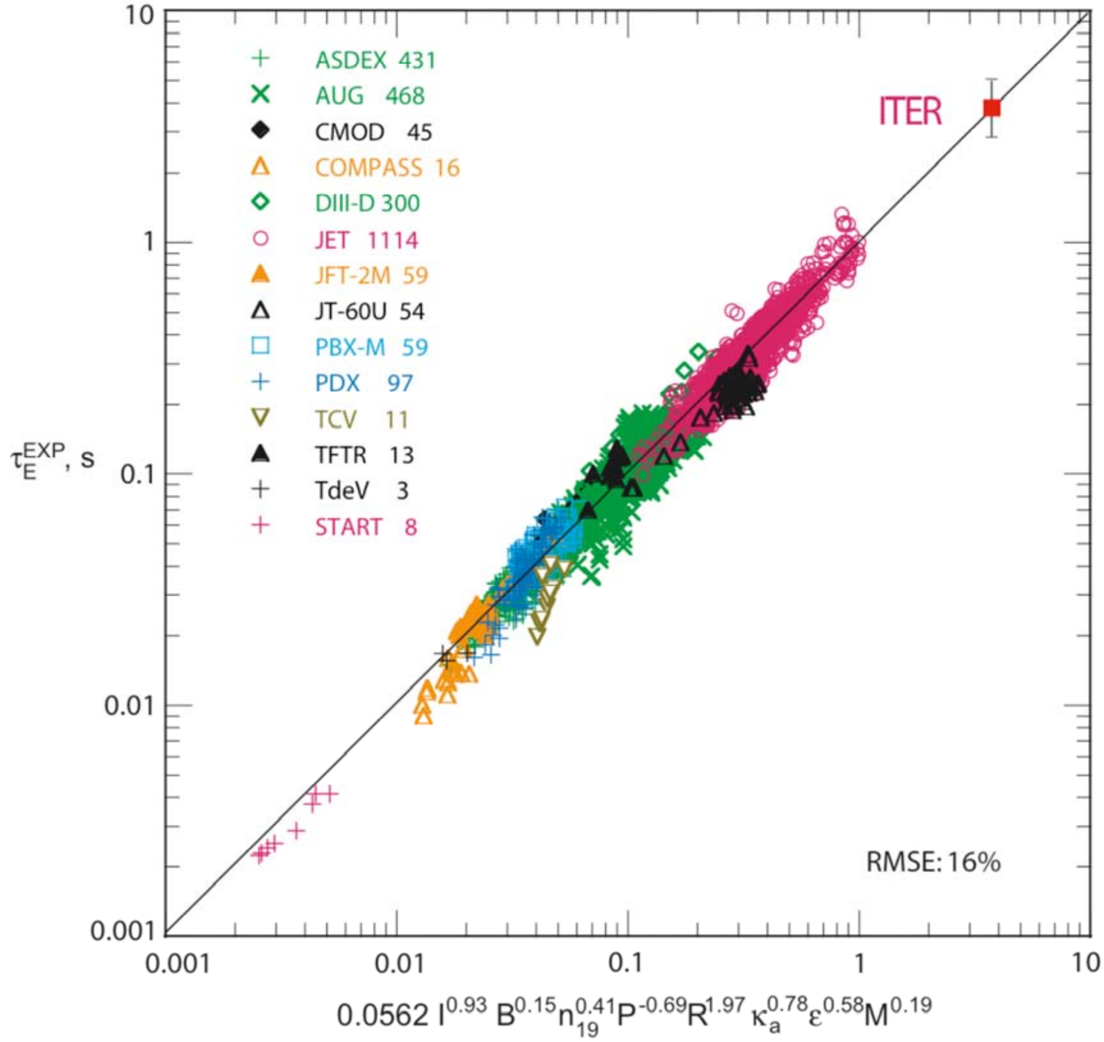
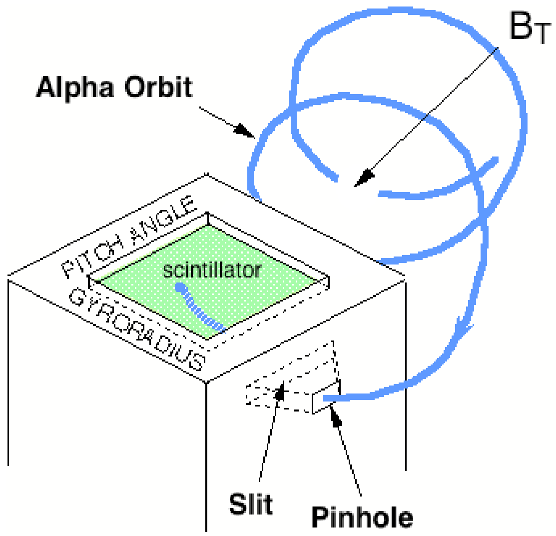

Thomson Scattering
Beam Emission Spectroscopy
Divertor Scanning Probe
Divertor Materials Exposure
Lithium Beam Polarimetry
Midplane Visible Camera
ECE
Pellet Injector
VUV Tangential Camera
Massive Gas Puffing
Pyrometer
IR Camera
CER
MSE
Neutral Beam
SXR
Glow Discharge
ICRF Antenna 0° port
Neutral Beam
CER
CER
UV SPRED Spectrometer
Visible Bremsstrahlung
MSE
Residual Gas Analyzer
Pyrometer
CER
IR Camera
Magnetic Probes
Divertor Bias Ring
Tile Current Monitors
MSE
Bolometer
Filter Scope
Phase Contrast Imaging
Neutron Detectors
ECE
Langmuir Probe Array
Interferometer
Fast Pressure Gauge
IR Camera
ICRF Antenna Camera
Visible Camera Divertor
VUV Camera Divertor
Pyrometer
Fast Wave ICRF Antenna
Neutral Beam
Fast Stroke Langmuir Probe
Reflectometer
Interferometer
FIR Scattering
Tangential Visible Camera
Fast Radiated Power
Electron Cyclotron Heating
Neutral Particle Analyzer
SXR
Neutral Beam
IR Camera
Fast Magnetics
Divertor Spectrometer

Figure 1

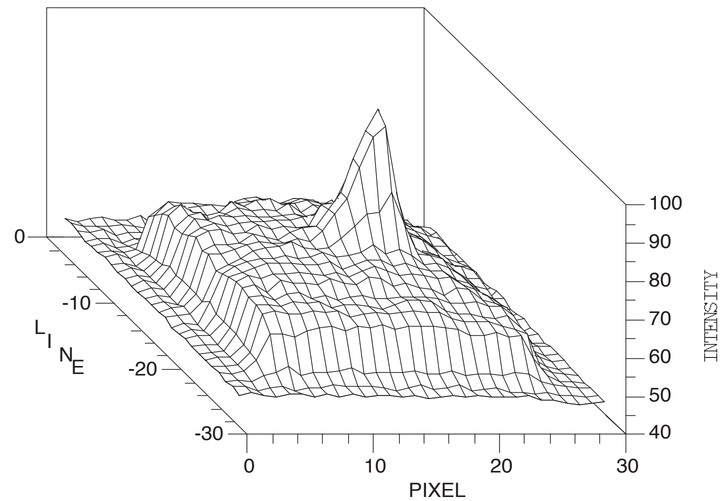


ASDEX 431
AUG 468
CMOD 45
COMPASS 16
DIII-D 300
JET 1114
JFT-2M 59
JT-60U 54
PBX-M 59
PDX 97
TCV 11
TFTR 13
TdeV 3
START 8
ITER
RMSE: 16%
τ_E^{EXP}, s
$0.0562\ I^{0.93}\ B^{0.15} n_{19}^{0.41} P^{-0.69} R^{1.97}\ \kappa_a^{0.78} \varepsilon^{0.58} M^{0.19}$

Figure 2



Alpha Orbit
B_T
PITCH ANGLE
scintillator
GYRORADIUS
Slit
Pinhole



0
-10
-20
-30
LINE
0
10
20
30
PIXEL
100
90
80
70
60
50
40
INTENSITY

a) b)

Figure 3

Table I. Diagnostics for transport studies.

Plasma Parameter	Typical range and coverage	Spatial; Temporal Requirements	Typical precision requirement	Available Diagnostic Techniques	Comment	Chapter
Electron density profile	1×10^{19} - 5×10^{20} m^{-3}	a/30; 10 ms	5%	Interferometry; polarimetry parallel to B-field	Continuous; many wavelengths can be used	3,4
				Reflectometry	Swept, microwave	3
				Thomson Scattering	Imaging or LIDAR for spatial dependence	4
Electron temperature profile	0.5 – 15 keV	a/30; 10 ms	10%	Electron cyclotron emission (ECE)	Swept, microwave	3
				Thomson Scattering	Imaging or LIDAR for spatial dependence	4
				X-ray pulse-height analysis	Counting in time-bursts, multichannel	5
Ion temperature profile	0.5 – 15 keV	a/30; 10 ms	10%	Charge exchange recombination spectroscopy (CXRS)	Continuous during beam pulse, multiple sightlines	6
				X-ray crystal spectroscopy	Counting in time-bursts, multichannel with impurity seeding	5
				Neutral particle analysis	Counting in time-bursts, needs neutral beam, multichannel	8
				Neutron spectroscopy	Counting in time-bursts, multichannel	9
Plasma rotation profile	v_{tor} = 1 – 500 km/s	a/30; 10 ms	10%	X-ray crystal spectroscopy	Counting in time-bursts, multichannel with impurity seeding	5
	v_{pol} = 1 – 100 km/s	a/30; 10 ms	10%	Charge exchange recombination spectroscopy (CXRS)	Continuous during beam pulse, multiple sightlines	6
Plasma pol-oidal beta, β_p	.01 – 5	Integral: 1 ms	5% @ β_p=1	Diamagnetic loop	Closed loop around the plasma	2
Current density profile (or q-profile)	q(r) >0.5	a/30; 10 ms	10% (q<5); 0.5 (q≥5)	Motional Stark effect (MSE)	Continuous during beam pulse, multiple sightlines	6
				Polarimetry	multiple sightlines in radial plane	4
Profile of the radial electric field	5 – 100 keV m^{-1}	a/30; 10 ms	TBD	Motional Stark effect (MSE)	Continuous during beam pulse, multiple sightlines	6
				Heavy ion beam probe	Sweep beam in angle, multi-detector	8
Radiation profile	0.01 – 1 MW m^{-3}	a/30; 10 ms	20%	Bolometry	Many arrays for tomography	7
Z_{eff} profile	1 < Zeff ≤ 5	a/10; 100 ms	10%	Visible continuum array	Continuous, multi-detector	5
Impurity concentrations	Z≤10 ions (e.g. Be, O, N, C)	Integral; 10 ms	10% (rel)	Visible, UV spectroscopy	Continuous, few sightlines to different regions of plasma	5
	Z≥10 ions (e.g. Cu, Ne, Ar, Kr,W)	Integral; 10 ms	10% (rel)	X-ray spectroscopy	Continuous, few sightlines to different regions of plasma	5

Table II. Diagnostics for studies of instability and turbulence

Plasma Parameter	Typical range and coverage	Spatial; Temporal Requirements	Available Diagnostic Techniques	Comment	Chapter
Low (m,n) MHD modes, sawteeth, locked modes, and disruption precursors	$\delta B/B \geq 10^{-3}$, $\delta T/T \geq 10^{-3}$, $\delta n/n \geq 10^{-4}$, $(0,0) < (m,n) < (10,2)$	$\Delta r = a/30$; 0 – 30 kHz	Mirnov coils outside plasma	Coils distributed around large and small circumference to provide (m,n)	2
			X-ray imaging system	Many sightlines provide correlation	7
			Electron cyclotron emission (ECE)	Fixed-frequency channels; ≥2 toroidal locations; imaging	3
			Reflectometry	Many channels for different radii	3
			Collective scattering	Needs high power microwave source, multiple viewing lines	3
			Neutron scintillator array	Integrates along lines-of-sight	9
			Moveable magnetic probes	For low temperature plasmas only	10
			Heavy ion beam probe	Integrated over path	8
High frequency instabilities (MHD, NTMs, KBMs, AEs, turbulence)	$\delta B/B \geq 10^{-5}$, $\delta n/n \geq 10^{-5}$, $n = 10 – 50$	10 mm; 10 – 1000 kHz	Mirnov coils outside plasma	Numerous coils	2
			Reflectometry	Many channels for different radii	3
			Microwave scattering	Variable frequency systems	3
			Beam emission spectroscopy	Many observing sightlines	6
			Phase contrast imaging	Wide beam, multiple detector array	4
			Langmuir probes	Correlation between pins and many locations (mostly edge)	10
ELMs		5 mm for r/a> 0.8	H-alpha spectroscopic arrays	Measurement of bursts of H_α light	5
Edge turbulence	$k\rho_i \sim 0.1$	~ 1 cm for r/a> 0.8, <200 kHz	Langmuir probes	Correlation between pins and many locations	10
			Fast visible imaging camera	Supported by gas puff	10

Table III. Diagnostics for the plasmas in the edge and divertor.

Plasma Parameter	Typical range and coverage	Spatial; Temporal Requirements	Typical precision requirement	Available Diagnostic Techniques	Comment	Chapter
Electron density profile	$5x10^{18}$ - $2x10^{20}$ m^{-3}	5 mm; > 100/s or 10 per s.	5%	Reflectometry	Swept, microwave	3
				Thomson Scattering	Imaging or LIDAR for spatial res.	4
				Langmuir probes	Probe moved radially	10
Electron temperature profile	0.05 – 15 keV	5 mm; > 100/s or 10 per s.	10%	Thomson Scattering	Imaging or LIDAR for spatial res.	4
				Langmuir probes	Probe moved radially	10
Ion temperature profile	0.05 – 15 keV	5 mm; 10 ms	10%	Charge exchange recombin-ation spectroscopy (CXRS)	Continuous during beam pulse, multiple sightlines	6
				Neutral particle analysis	Counting in time-bursts, multichannel	8
				Optical spectroscopy	Doppler broadening	5
Edge current density	q(r) > 2 - TBD	5 mm; 10 ms	10%	Motional Stark effect (MSE)	Continuous during beam pulse, multiple sightlines	6
				Lithium beam polarimetry	Visible spectroscopy while beam is on	6
				Magnetic probes	Probe moved radially	10
X-point, MARFE region radiat-ion profile	TBD - 300 MW m^{-3}	5 mm; 10 ms	20%	Bolometry	Continuous	7
Divertor P_{RAD}	TBD – 100 MW m^{-3}	50 mm; 10 ms	30%	Bolometry	Continuous	7
				UV-spectroscopy	Continuous	5
Divertor surface temp.	200 – 2500°C	3 mm: 20 ms	10%	Infra-red imaging	As wide coverage as possible toroidally	10
Divertor net erosion	0 – 3 mm	10 mm apart; 1 s	0.2 mm	Speckle reflectometry	Continuous	10
Dust monitoring	TBD	Several locations; TBD	TBD	Quartz microbalance, electrostatic monitor	Continuous capacitance measurement; in early stages of development	10
Position of ionization front	0 – 0.3 m	20 mm; 1 ms		Visible spectroscopy	Multiple sightlines along outboard separatrix	5
Divertor density profile	10^{19} – 10^{22} m^{-3}	20 mm along leg, 3 mm across leg; 1 ms	20%	IR interferometry	Many sightlines along separatrices	4
Divertor electron temperature	0.3 – 200 eV	20 mm along leg, 3 mm across leg; 1 ms	20%	Thomson scattering	Imaging system	4
Ion temper-	0.3 – 200 eV	20 mm along leg, 3	20%	UV spectroscopy	Many sightlines along separatices	5

ature in the divertor		mm across leg; 1 ms				
Plasma flow in divertor plasma	TBD – 10^5 ms^{-1}	20 mm along leg, 3 mm across leg; 1 ms	20%	Mach probes	Moveable probe	10
n_e at target	10^{18} – 10^{22} m^{-3}	3 mm; 1 ms	30%	Langmuir probes	Embedded in divertor target	10
T_e at target	1 eV – 1 keV	3 mm; 1 ms	30%	Langmuir probes	Embedded in divertor target	10
Fuel ratio in edge (n_H/n_D, n_T/n_D)	0.01 – 100	Integral; 100 ms	20%	Spectroscopic technique	Many sightlines	5
Fuel ratio in divertor (n_H/n_D, n_T/n_D)	0.01 – 100	Integral; 100 ms	20%	Spectroscopic technique	Many sightlines	5
Divertor helium density	10^{17} – 10^{21} m^{-3}	Integral; 100 ms	20%	Penning gauge	Close behind divertor plate	10
Neutral density between plasma and first wall	10^{18}–10^{20} at $m^{-2}s^{-1}$	Integral at several locations; 100 ms	30%	Pressure gauges?		10

Table IV. Diagnostics for burning plasma studies.

Plasma Parameter	Typical range and coverage	Spatial; Temporal Requirments	Typical precision requirement	Available Diagnostic Techniques	Comment	Chapter
Neutron- and α-source profile	$1x10^{12} – 4x10^{18}$ n m^{-3} s^{-1}, for r/a≤0.75	a/30; 1 ms	10%	Neutron cameras	Many sightlines integrating along line of sight; need careful calibration with flux monitors	9
				Microfission chambers	Many may give spatial profile	9
Total neutron yield	0.1 – 60 MJ m^{-2}	10 s	10%	Neutron flux monitors	Numerous detectors to compensate for material arrangements near detectors; suitable for 2.5 and 14 MeV neutrons	9
				Neutron activation systems	Many foils close to the plasma	9
Core thermal Helium density (ash)	$1 < n_{He}/n_e < 20\%$	a/10; 100 ms	10%	Charge exchange recombination spectroscopy (CXRS)	Continuous during beam pulse, multiple sightlines: very challenging	6
Fuel ratio in the plasma core	$.01 < n_H/n_D < 100$, $.1 < n_T/n_D < 10$	a/10; 100 ms	20%	Fabry-Perot spectroscopy	Many sightlines	6
				Neutron spectroscopy	Mounted in camera	9
Confined fast ion and α-particle energy spectrum & density profile	0.1 – 3.5 MeV, $(0.1 – 2)x10^{18}$ m^{-3}	a/10; 100 ms	20%	Collective Thomson scattering (microwave, FIR, CO_2)	Microwave scattering has best spatial resolution, not yet demonstrated for fast ions	3
				Alpha-CHERS	Uses neutral beam; measures up to ~1/4 of beam energy; beam penetration critical	6
				Pellet charge exchange	Requires impurity pellet penetration; rapid pulsing, location from time of flight	8
				Knock-on tail neutron spectroscopy	Very small tail for time resolution in neutron spectrometers; very sharp energy edge needed for bubble detectors	9
				Gamma-spectroscopy	Best in fast-ion studies with little neutron background	9
				Ion cyclotron harmonics	Ambiguous results on JET and TFTR	Ref. [22]
Escaping fast ions and α-particles (first wall flux)	TBD – 2 MW m^{-2}, TBD – 20 MW m^{-2} in transients	a/10 along poloidal direction	10%	Scintillators, Faraday cups, activation foils, diamond detectors mounted in protected pinhole camera arrangement	Mounted on outer quadrant of 1st wall (depending on preferred B-field direction); activation foils integrate over pulse, Faraday cups measure energy, scintillators also measures source localization	9, Refs. [33 – 37]
				Infra-red camera	Measures total energy deposited by alphas	10
High frequency instabilities (MHD, NTMs, KBMs, AEs, turbulence)	$\Delta B/B \leq 10^{-2}$, $\Delta n/n \leq 10^{-2}$, n = 10 – 50, $\Delta\phi \leq 10^{-2}$	10 mm; 10 – 1000 kHz		Mirnov coils outside plasma	Numerous coils	2
				Reflectometry	Many channels for different radii	3
				Beam emission spectroscopy	Many observing sightlines	6

Table V. Diagnostics to enable plasma operation:

Plasma Parameter	Condition	Typical range and coverage	Spatial; Temporal Requirments	Typical precision requirement	Available Diagnostic Techniques	Comment	Chapter
Gas pressure in main chamber and vacuum ducts	Between pulses	1.10^{-7} - 1.10^{5} Pa	10 s	10%	Ion gauges		10
	During pulses	1.10^{-7} - 20 Pa	100 ms	20%	Ion gauges	Must operate in changing magnetic fields	10
Gas composition in main chamber and ducts	A = 1-100, ΔA=0.5, between pulses	TBD	10 s	10%	Residual gas analysis, Penning gauges	Locations must be determined	10
	A = 1-100, ΔA=0.5, during pulses	TBD	1 s	50%	Residual gas analysis	Locations must be determined; helium content in divertor will be important	10
Magnetic field configuration	Toroidal Field, B_T	0.1 - 10 T	None; follow pulse shape.	0.1%	Rogowski coil on buswork		
	Plasma current, I_p	0.1 - 20 MA	None; 1 ms	10 kA for I_p<1 MA 1% for $I_p \geq 1$ MA	Rogowski coil around plasma	Discrete coils could be summed	2
	Plasma position		10 mm; 10 ms	2 mm	Small solenoids ins-ide vacuum vessel	For reconstruction	2
					Reflectometry using profile	Proposed for long-pulse in ITER	3
	Plasma pol-oidal beta,β_p	.01 – 5	Integral: 1 ms	5% @ β_p=1	Diamagnetic loop	Closed loop around the plasma	2
Voltage around torus	For normal operation and disruptions	0 – 30 V 30 – 500 V	1 ms	5 mV, 10%	Voltage loop	Several toroidal loops	2
Runaway electrons	At start-up and thermal quench	1 – 20 MeV; (0.05 – 0.7).I_p	10 ms	30%	X-rays, ECE, forward luminescence	Most serious at thermal quench	5,3
First wall visible image and wall temperature	First wall images	As much as possible	100 ms; 1 mm		Digital cameras		10
	Wall surface temperature	200 – 1500°C	10 ms; 10 mm	20°C	IR Cameras	Faster times desirable for fast-ion losses	10
Vacuum Vessel Illumination	Provide lighting of the first-wall surfaces between shots				Several sets of 4 ~600W lamps[A]	Inspection of the first-wall surfaces between shots	ref. 39

[A] Approximate operational numbers for system on TFTR.

Table VI. Systems in support of active diagnostics.

System	Purpose of system	Requirements	Used in diagnostic:	Reference
Fuel-particle Neutral Beam	Provide neutral population to enhance signals by charge-exchange processes	~ 100 keV/amu, modulated, ~200mm x 200mm, ~ 50 mA/cm^2.[A]	CXRS, H_α-monitor, He-ash monitor, Alpha-CXRS, charge-exchange analysis	[40]
		~ 100 keV/amu, .05 m^2, 50kA, 1 μs, repetitive	Same	[41]
	Provide fast-moving excited atoms	~ 500 keV/amu, ~ 20 mA/cm^2.[B]	MSE, Beam emission spectroscopy (BES)	
Impurity Neutral Beam	Provide excited neutral atoms for intensity and polarimetry measurement	Li, 30 keV, 10 mA, 1 - 2 cm dia.[C]	Edge density, toroidal current density in edge	[42]
Impurity Pellet Injector	Provide neutral population to enhance signals by charge-exchange processes	Li, B,C; ~2mm dia., ~600 m/s, ≤ 4/pulse[D]	Pellet charge-exchange	[45]
	Provide fast-moving excited ions	Li, ~2mm dia., ~700 m/s, ≤ 4/pulse[D]	q-profile from polarization of Li^+ light	[43 – 44]
Gas Puff	Provides neutral atoms to highlight edge density turbulence or highly-ionized ion states in the core	10^{19} - 10^{20} atoms/s, 3 mm dia nozzle, D, He or Ar.[E]	Edge imaging of fluctuations	[46]
		10^{18} – 10^{20} atoms/s, Ar, Ne, Kr	X-ray diagnostics, especially crystal spectroscopy	
Impurity Laser Ablation	Inject neutral metal atoms for edge spectroscopy	~1J, 30 Hz laser pulse on thin film of impurity	Edge spectroscopy	[47]

[A] This ITER/FIRE specification is for a negative-ion based beam.
[B] This specification is for the ITER heating beam.
[C] Approximate operational numbers for system on DIII-D.
[D] Approximate operational numbers for system on TFTR.
[E] Approximate operational numbers for system on Alcator C-Mod.

Table VII. Measurements required for control of a burning plasma in a tokamak:

	Parameter to be Controlled	Measurement(s)	Control Response Action
Establishing plasma scenario	Plasma current magnitude	Plasma current	OH current drive, NB or RF current drive
	Plasma position and shape	Edge magnetic fields, edge density profiles	Poloidal field currents
	Plasma density	Integrated line density	Gas or pellet fueling
	Electron temperature	Electron temperature profile	RF heating
	Transport barrier formation	Electron density profile, electron temperature profile, ion temperature profile, current density profile	Fuel injection, current, heating systems
Maintaining Scenario	Plasma-β profile	Electron density profile, electron temperature profile, ion temperature profile, current density profile, magnetic field strength	Fuel injection, current, heating systems
	Relative fuel-ion density profile	Spectroscopy of H, D, T in core	Gas and pellet fueling
	Fusion power (burning plasma control)	β -profile (above), neutron flux	Injection of fuel species, heating systems
	Build-up of helium-ash	He-spectroscopy in core, He-density in edge/divertor	To be determined
	Disruption remediation	β-profile (above), fast edge magnetic MHD	Intense impurity gas or pellet injection
Stabilizing gross instabilities	Error fields and locked-modes	Low-frequency MHD, core ion rotation profile	Current/rotation drive
	Low (m.,n) MHD modes and sawteeth	Low frequency MHD	OH current drive, NB or RF current drive
	Resistive wall modes (RWMs)	Edge magnetic MHD	Active stabilizing coils
	Neo-classical tearing modes (NTMs)	Edge magnetic MHD, core density fluctuations, q-profile	Electron cyclotron heating
	Plasma rotation profile	Ion rotation profile	Current drive
Plasma control in the edge and divertor	ELMs and edge turbulence	H-alpha profile	Poloidal field, current drive
	Strike point location	Divertor plate temperature	Poloidal field supplies
	Detachment from divertor plates	Divertor-plate temperature	Impurity injection in divertor

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Information Services
Princeton Plasma Physics Laboratory
P.O. Box 451
Princeton, NJ 08543

Phone: 609-243-2750
Fax: 609-243-2751
e-mail: pppl_info@pppl.gov
Internet Address: http://www.pppl.gov