ITER PHYSICS BASIS

CHAPTER 7: MEASUREMENT OF PLASMA PARAMETERS

ITER Physics Expert Group on Diagnostics*
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ABSTRACT. This is Chapter 7 for the ITER Physics Basis. In this chapter, the physics issues of the measurements of the plasma properties necessary to provide both the control and science data for achieving the goals of the ITER device are discussed. The assessment of the requirements for these measurements is first discussed, together with priorities, which relate to the experimental program. Then some of the proposed measurement techniques, the plasma diagnostics, are described with particular emphasis on their implementation on ITER and their capability to meet the requirements. A judgement on the present status of the diagnostic program on ITER is provided with some indication of the Research and Development program necessary to demonstrate viability of techniques or their implementation.

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MEASUREMENTS OF PLASMA PARAMETERS

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7.1. INTRODUCTION

Having reviewed the physics needs and expectations for the plasmas in the ITER device, it is now necessary to consider the requirements for a plasma measurement system to make possible the control of these plasmas, the understanding of the plasma behavior and the optimization of performance. Relative to present experience, there will be many unique features of the plasma with its self-heating by alpha-particles, its large-size, very large power deposition on first-wall surfaces, long plasma time constants and the expected long pulse length. These features set stringent goals for the measurement of plasma parameters. There are also many constraints on the plasma diagnostic capability to fulfill these goals; operation in intense neutron and gamma fluxes with limited viewing access through the necessary shielding and into the divertor region, the capability for components to be able to be maintained remotely, and assured long-term calibration stability and reliability. This section reviews briefly the requirements for the plasma measurements and some of the diagnostic systems which are being designed to meet these requirements. It is clear that there are many challenges to creating a set of diagnostics to meet all the goals, so that a significant research and development (R&D) program is required and has started.

In experiments today, the power of new diagnostics to provide full profile information about many plasma parameters has led to tokamak operation in new regimes and better understanding of the plasma behavior by analyses with theory-based codes [7.1–7.3]. Detailed measurements of plasmas in the divertor are being used to benchmark codes used for optimizing the first-wall hardware design [7.4] and to provide detailed understanding of the critical issues of power and particle handling [7.5–7.7]. These diagnostic techniques are rapidly being improved. Neutron and alpha-particle diagnostics have provided invaluable information about fusion product behavior in the plasma core [7.8–7.10]. The demonstrated capabilities of the instrumentation on operating devices, and the uses made of the data in interpretation and control of their plasmas, provide the basis for setting measurement requirements for ITER [7.11, 7.12]. The specific diagnostic systems now being considered for ITER are proposed mostly because of the demonstrated successes of the techniques. However the ITER parameter range introduces serious difficulties for diagnostics: for example, penetration of a diagnostic neutral beam for spectroscopic measurements is poor and the high T_e greatly affects measurement using electron cyclotron emission (ECE).

The demands of plasma control on the measurement requirements have been considered closely. Real-time control of plasma position and shape, current, density, with some additional permissive controls, is common on operational tokamaks. Control making use of such profile information as q(r) and $n_e(r)$ is now being introduced for improving the performance of advanced tokamak modes as they evolve in time. In addition, for ITER, it will be necessary to maintain stable ignition, and it will be highly desirable to inhibit disruptions. During the basic physics phase, as the plasma performance is improved and understood, the role of diagnostic signals in control will increase. The measurements which are expected to play a role in the feedback control loops have been identified. The diagnostic capability will need to be built up quickly in order to meet the demands of the ITER physics program. A comprehensive measurement capability is needed within a few years to allow the implementation of the full control capability. Diagnostics not needed for the control function will be essential for comparing the plasma behavior with the predictive codes, and therefore for indicating to the operational physicists ways to optimize the performance.

Detailed design of many diagnostics has been started to evaluate the capability for carrying out key measurements and to define R&D programs which might be necessary. Physics issues associated with the measurement techniques have been evaluated. Details in the designs have been mostly limited to the hardware close to the tokamak, where access and radiation effects provide major challenges. Such design is necessary now because of the very complex integration with many major tokamak components whose design must be frozen soon. The need to limit the amount of neutron streaming constrains the penetrations allowable through each port. Thus the sets of diagnostics, with their associated shielding labyrinths, have to be defined on the same time scale. These designs involve ~~novel~~ concepts such as making use of reflecting optical components near to the plasma and each light path goes through a labyrinth with many bends to preserve the shield quality. Care is clearly needed in design of components requiring good electrical isolation or long cables inside the vacuum vessel in a high radiation field. Detailed engineering issues associated with cable bundle sizes and stiffness, connectors, vacuum feedthroughs, etc., are being addressed in the ITER design, but will not be described here.

Most components far from the tokamak will be in a relatively benign environment, similar to that enjoyed by components at present-day tokamaks. Since these components, such as detectors and signal-processing electronics, will be similar to, or better than those presently in use, detailed design is not necessary now.

After discussion of the measurement requirements and the priorities established for them, this section will provide descriptions of some of the key diagnostics. A number of issues remain unresolved: techniques to measure certain important parameters have not been identified; for some of the techniques which have been identified, it is not clear that fully capable implementations can be made in the ITER environment. Future R&D activities are required to address these issues.

7.2. REQUIREMENTS FOR PLASMA MEASUREMENTS

7.2.1 Role of Plasma Measurements

Measurement of the plasma and first-wall parameters will have three main roles on ITER: (a) to provide input for real-time machine protection systems, (b) to provide input for real-time plasma control systems, and (c) to provide data for physics studies and for the evaluation of plasma performance. Some measurements may contribute to all three roles although the required accuracy, resolution, etc. will be different for each role [7.13].

Machine protection will be particularly important in ITER due to the combination of the large stored thermal and magnetic energies (~1 GJ), the large power flow into the scrape-off layer (~200-300 MW) and the long pulse duration which have the potential to cause severe damage to in-vessel components. Even a relatively short (≥1-2 s) contact of the plasma with the first wall could result in localized surface melting. The position of the plasma edge relative to the first wall is therefore a critical parameter for keeping the first wall power load below the allowable limit (0.5 MW/m^2). Similarly, an uncontrolled rise of the fusion power and/or change in the extent of divertor attachment could increase the thermal load onto the divertor plates above the design limit (5 MW/m^2 at steady state). The distance between the separatrix and the first wall, the surface temperatures of the divertor and first wall, and the fusion power are therefore key parameters that have to be measured for machine protection (Table 7-I).

Table 7-I. Parameters To Be Measured For Machine Protection

Parameter	Limit
Separatrix/wall gaps	>5 cm
First wall surface temperature	≤400°C (hot spots)
Divertor plate surface temperature	≤800°C (hot spots)
Fusion power	≤1.8 GW (20% above nominal value)
Locked modes	Avoidance
Runaway electron current	$\leq 0.1 \times I_p$
Type I (Giant) ELMs (at full parameters)	Avoidance
Specific impurity emission	To be defined
Line-averaged density (NBI shine-through)	$\geq 0.3 \times 10^{20}$ m^{-3} for D^0 beams

Plasma disruptions could deposit up to 100 GJ/m^2 of energy on the divertor plates during a thermal quench (1-10 ms) which would cause significant plate damage and erosion. The formation of runaway electrons during the disruption could lead to damage to other plasma facing components. Poloidal 'halo' currents associated with vertical displacement events could cause seriously high mechanical stresses in the blanket/shield structure and divertor cassettes. Disruptions will probably be a major factor determining the life-time of the divertor plates and so the number of disruptions should be kept small and/or their effects mitigated. Experiments show that locked (stationary) modes, especially those induced by the resonant m/n=2/1 error field component, often lead to disruptions. An early and reliable detection of locked modes is a prerequisite of their unlocking (e.g., through a plasma rotation induced by neutral beam injection) or a decision to induce a controlled termination. Measurements of locked modes and runaway electrons are therefore included in the list of machine protection measurements.

The edge localized mode (ELM) can be beneficial for impurity and helium expulsion from the plasma edge and controlled ELMs may be necessary to maintain a steady state H-mode but Type I (giant) ELMs have to be avoided as they could lead to a substantial pulsed heat load (i.e., ~25 MJ for ~0.1-1 ms every 1 s) to the divertor plates. This heat load is close to the ablation threshold and may increase significantly the erosion rate of the divertor plates. Measurement of ELMs is therefore included as a machine protection parameter.

A change in the concentration of a particular impurity, e.g., Be, could be an indication of increased plasma-wall interaction and so specific impurity emission must be measured. The electron density must be maintained above a certain value otherwise shine-through of the NBI may lead to damage to the first wall.

All of these parameters are measured routinely on existing tokamaks: the ITER requirements for machine protection introduce new standards of reliability.

Plasma control is conveniently divided into two areas — the control of the magnetic configuration (equilibrium control) and the control of the other key parameters of the main and divertor plasmas (kinetic control). The former involves the control of the plasma current, position, shape (including the control of specific separatrix-wall distances) and the divertor configuration. The essential requirement is to maintain the magnetic configuration independent of changes in kinetic plasma parameters. Feedback control of the vertical position must be provided in order to avoid unstable vertical displacements, and to keep the plasma-wall clearance gaps within acceptable limits during the 'normal' operation and during expected plasma disturbances, i.e., ELMs, sawteeth and minor disruptions. It leads to requirements for measurements of the plasma current, specific separatrix-to-first wall clearance gaps and the position of divertor strike-points.

Plasma kinetic control addresses mainly the problem of simultaneously controlling the fusion power and maintaining the divertor conditions. The ITER divertor is designed for a steady-state peak power load into the divertor plate $P_{div,ss} \leq 10\ MW/m^2$ with power transients up to $P_{div,tr} \leq 20 MW/m^2$ for $\leq 10\ s$. This means that under the normal conditions only a small fraction (≤12-16%) of the total heating power ($P_{tot} = P_\alpha + P_{aux}$= 300-400 MW where P_α is the alpha particle power and P_{aux} is the auxiliary heating power) can be allowed to reach the divertor plates; the majority, ~ 80%, must be radiated. The intrinsic impurity concentration to meet this radiation goal is not expected to be sufficiently large and it will be necessary to introduce controlled amounts of specific impurities (e.g. Kr, Ne, Ar) into the divertor and scrape-of-layer (SOL) plasma. At the same time it will be necessary to maintain the radiated power loss and plasma dilution in the core at levels acceptable for the plasma burn. Kinetic control must also keep the plasma away from the β-and density-limits and provide a sufficient power flow through the separatrix to ensure operation in the H-mode. These objectives lead to requirements for measurements of a wide range of plasma parameters including fusion power, divertor plate temperature, radiative power loss, plasma density, β, n_D/n_T ratio, locked and rotating MHD modes, plasma rotation rate, ELM characteristics, impurity content and influxes, and a degree of divertor detachment, i.e., 'ionization front' position and/or T_e and n_e at the divertor plate. Specific tailoring of the plasma current profile and control of the pressure profile would be required, in addition to the control of the basic plasma parameters, for more sophisticated confinement scenarios, such as with negative central shear, and will require additional measurements of the profiles of current density as well as the electron and ion densities and temperatures.

There is extensive experience of controlling the magnetic configuration of tokamak plasmas and much of this will be directly applicable to ITER. The principal new aspect of ITER will be the extremely long pulse length which may require the development of measurement methods additional to the traditional inductive magnetic measurements. Similarly, there is a substantial body of experience at controlling the basic plasma parameters involved in kinetic control, and stabilization of neoclassical modes, but for ITER it will be necessary to control considerably more parameters than are controlled on present-day tokamaks. Techniques and methods to achieve the necessary control are being developed in current experimental programs.

Finally, an even wider range of measurements will be required to evaluate the plasma performance and to provide information on key phenomena which may limit the performance. The necessary measurements are discussed in Section 7.2.2 in relation to the topics that are likely to be of principal interest in the experimental program.

7.2.2 Relationship to Experimental Program

ITER will be a substantial extrapolation from present-day tokamaks, especially in plasma size, energy content and pulse duration. The completely new regime associated with an ignited plasma makes it certain that an extensive series of experimental studies will be required in ITER in order to understand the physics of burning plasmas and to optimize their performance. The plasma measurements that will be required for these physics studies will include most of those required for plasma control although in most cases better time and space resolution will be needed. Some of the diagnostic requirements that have been identified for specific areas of physics study are discussed briefly below, but it is important to recognize that these requirements can be expected to evolve both during the ITER construction period, in the light of experience that will be gained in other fusion experiments, and during the life of ITER, in response to ITER results.

Confinement of reactor-scale plasma. To characterize confinement and transport of a reactor plasma, a broad range of plasma measurements is required. The main parameters to be measured are radial profiles of the ion and electron temperatures, electron density, radiated power and heating power density. Also, the particle source strength and the contents of the main impurities, i.e., He, Be and C, must be measured. Ten experimental points along the minor radius are considered to be the minimum needed in order to define the profile shape in the plasma core. In the edge, sharp gradients of some profiles, notably temperature and density, are expected and therefore a spatial resolution of < 1 cm is required. Time resolution of ~100 ms is considered adequate for most studies but faster time resolution (~1 ms) will be required to resolve parameter changes during fast transients (e.g., sawteeth and ELMs).

Operational limits. The principal parameters required for studies of the operational space [7.14] include profiles of β-value, plasma pressure, and electron density as well as various characteristics of the edge and scrape-off layer plasmas. Measurements of MHD activity, edge plasma turbulence, q(r), and poloidal and toroidal plasma rotation velocities will also be necessary.

Disruption phenomena. Fast (~ 100 μs) time- and space-resolved measurements of radial profiles of temperature, density, safety factor etc. in the plasma core and boundary will be essential to document the physics phenomenology of disruptions in ITER. Supplemental fast magnetic equilibrium reconstruction data and time- and space-resolved measurements of energy deposition on the first wall, divertor entrance baffle and divertor targets are also essential for documenting plasma configuration and energy partitioning in disruptions. Magnetic measurements of both plasma current and configuration evolution and in-vessel component halo current flow will be required to document VDE and halo current characteristics. These measurements and the associated plasma equilibrium reconstruction methods must be capable of accurately resolving configuration and in-vessel halo current evolution on a 10-ms or faster time scale. Measurement of the toroidal distribution of the in-vessel currents is required. The magnetic measurements and reconstruction methods must be compatible with providing accurate plasma cross-section reconstruction despite the presence of significant (~ 10% I_p) induced toroidal currents in the vacuum vessel and/or blanket-support backplate. The macroscopic effects of appreciable runaway conversion at high plasma current will be readily apparent in the planned array of ITER plasma current and magnetic configuration diagnostics. More detailed measurements of the onset of runaway conversion at lower plasma current and documentation of the physics basis of runaway conversion will require specialized diagnostics to measure the synchrotron radiation of in-plasma runaway and the X-ray emission of runaways lost to in-vessel component surfaces.

Physics of high-power-flux radiative divertor. The following measurements are highly desirable for understanding the operation of the divertor over the full range of its operation from fully attached to fully radiative [7.15]: (i) parallel and perpendicular particle and energy fluxes in the two divertor legs; (ii) gradients in the plasma parameters perpendicular to the flux surfaces with scale lengths of mm's; (iii) 2-D profiles of electron and ion temperatures and densities in the divertor and edge region; (iv) concentrations and fluxes of the impurities; (v) partial pressures of the constituents of the gas in the divertor chamber; (vi) edge plasma profiles during transient phenomena (e.g., MARFE's and ELMs). In practice it presently does not appear possible to make all of these measurements but many relevant measurements will be made (Section 7.3).

Data on the particle and energy transport processes in the divertor are essential for developing a knowledge of the physics of divertor operation that can be extrapolated to the next machine beyond ITER, or to later stages of ITER operation. Spatial resolution on the scale of the density and temperature scale lengths is required to calibrate models of divertor operation. Information on the impurity behavior is essential for developing an understanding of impurity transport and how to use impurity radiation for exhausting most of the fusion power. Transient phenomena (such as ELMs) will lead to severe heat and particle loads on the plasma facing components. An understanding of these effects is essential for developing the ability to predict their impact on further stages of ITER operation and future tokamaks.

Alpha-particle effects. Measurements of escaping alpha-particles, as well as profiles of the density and energy spectra of confined alpha-particles will be necessary for studying alpha-particle physics. While these measurements are expected to be very difficult in ITER they will be essential for validating the behavior of the alpha-particles in ignited plasmas. Fast MHD measurements with good spatial localization will provide information on fast-particle-driven TAE modes.

Studies of ignition domain. Determination of the ignition boundaries and the area of a high-Q driven burn will be an important part of experimental research on ITER. These tasks will require detailed studies of plasma core and divertor characteristics and understanding of their coupling. This leads to requirements for a wide range of plasma measurements, including fusion power; radiated power in the plasma core, plasma edge and divertor; power load on the divertor plate during a quiet phase and at ELMs; and electron densities and impurity contents in the plasma core, edge and divertor.

Steady-state burn studies. The principal plasma parameters that must be measured for optimizing the steady state burn are radial profiles of the safety factor and plasma pressure, profiles of poloidal and toroidal plasma rotation and radial electric field, plasma shape, beta, escaping alpha-particles, and key divertor parameters.

7.2.3 Selected Measurements and Priorities

The above requirements imply a need for a large number of measurements. However, resources such as manpower, budget, port space etc. are limited and so it is necessary to set priorities. Naturally the highest priority must be given to measurements for machine protection and basic plasma control. This leads to a convenient classification of the measurements:

- those that are required for machine protection and basic plasma control (group 1a);
- those that could potentially provide measurements for advanced plasma control (group 1b); and
- those that provide measurements for evaluation and physics studies (group 2)

The separation of control measurements between *basic* and *advanced* is somewhat arbitrary but it recognizes the fact that some parameters will require real-time control for every ITER pulse, (examples include plasma shape and position, plasma current and line-averaged density), while others will be controlled for specific programs. Examples of the latter are the plasma rotation and the q profile which may have to be under real-time control for specific high confinement modes of operation but are not necessarily controlled on every ITER pulse.

The setting of priorities according to the control requirements clearly has merits but it also has some limitations. Measurements of some parameters are essential to evaluate and optimize the plasma performance but are presently not envisaged as control parameters. For example, measurements of confined and escaping alpha particles will be critical for evaluating the fusion performance but will probably not be used as control parameters. It would be inappropriate to limit the measurements to only those necessary for machine protection and plasma control.

Taking into account these considerations, a list of required measurement parameters has been drawn up. This is arranged according to the priorities for control and presented in Table 7-II. The principal diagnostic for each parameter is also shown.

7.2.4 Detailed Measurement Requirements

The choice and design of the diagnostic systems start from the determination of the detailed measurement requirements. These are determined by the role the measurement will play in the experimental and operational program and by the anticipated variations (in space and time) of the parameter concerned. For example, the profile of the electron density will be used in studies of key phenomena such as particle confinement and may be used for advanced plasma control. Accurate measurements (~ 5%) are therefore required. The profile is expected to change on a scale of the order of tens of centimeters in the plasma core but on a scale of tens of millimeters in the plasma edge. MHD activity could change the profile on a time-scale of tens of milliseconds. This leads to the specific requirement for the measurement of the electron density profile:

	Density range	Spatial res.	Time res.	Accuracy
Core	$(0.3–3)10^{20}$ m^{-3}	10 cm	10 ms	5%
Edge	$(0.05–3)10^{20}$ m^{-3}	0.5 cm	10 ms	5%

Table 7-II. Priorities for Control Measurements

GROUP 1a Mach. Protect. & Basic Control		GROUP 1b Advanced Control		GROUP 2 Evaluation & Physics	
Parameter	**Diagnostic**	**Parameter**	**Diagnostic**	**Parameter**	**Diagnostic**
Shape/Position Locked Modes q(a), q(95%) Plasma Current Beta m=2 Mode 'Halo' Currents Loop Voltage	Magnetics	MHD Activity	Magnetics ECE Reflectometry	Fishbones, TAE Modes,	Magnetics, Reflectometry, ECE
Impurity Influx (main plasma & divertor)	Impurity Monitors	Shape/Position (very long pulse)	Reflectometry (plasma posit.)	Confined α-Particles	Collect. Scatt., Knock-on Tail Neutr. Spectr., NPA
Runaway Electrons	Hard X-Rays, Synchrotron Radiation	Neutron Profile, α-Source Profile	Rad. Neut. Cam., Vert. Neut. Cam.	$n_T/n_D/n_H$ (edge)	NPA, H_α Spec., Laser Induced Fluoresce. (LIF)
Line-averaged Density	Interf./Polarim.	n_{He} Profile	CXRS	$n_T/n_D/n_H$ (div)	H_α Spectrosc.
J_{sat} (divertor)	Tile Shunts	Plasma Rotat., T_i Profile, Impurity Profile	CXRS, X-Ray Crystal Spectroscopy, VUV Spectrosc.	T_e Profile (edge)	Thomson Scatt. (edge)
Surf. Temp. (divertor plates & first wall)	IR Cameras	T_e Prof. (core), n_e Prof.(core)	LIDAR (main), ECE	n_e, T_e Profiles (X-point)	Thomson Scatt. (X-point)
Rad. Power from Core, X-point and Divertor	Bolom. Array (main pl. & div.)	T_i Profile (core)	Radial Neutron Spectrometer	n_e, T_e (plate)	Langmuir Probes
Neutron Flux	Neutron Flux Monitors	n_e Profile (edge)	Reflectom. (main)	T_i in Divertor	Imp. Monitor. (div)
n_T/n_D in Plasma Core	NPA, Fast Wave Reflectometry	q Profile	MSE, Polarim. System	Plasma Flow (div)	Imp. Monitor. (div)
Z_{eff} Line-Aver.	Vis. Continuum (single channel)	P_{rad} Profile	Bolom. Arrays (main pl.& div.)	Pellet Ablation	H_α Spectrosc.
H/L Mode Indicator	H_α Spectrosc. (typ. channel)	Z_{eff} Profile	Visible Cont. Array	T_e Fluctuations	ECE, Soft X-Ray Array
ELMs (type)	ECE, Refl.(main) H_α Spectroscopy	n_{He} (divertor)	RGA, Imp. Monitor,	n_e Fluctuations	Reflectometry, Microw. Scatt.
Gas Pressure (div. & duct)	Pressure Gauges	Heat Deposition Profile in Div.	IR Camera	Radial E Field and E Fluctuat.	CXRS (plasma rot.)
Gas Composit. (div. & duct)	RGAs	Div. Ionization Front Position	Visib. Spectrom., Bolometry	Edge Turbulen.	Reflectometry
Toroidal Magnetic Field	Current Shunts	Neutral Density (near wall), Particle Source	H_α Spectroscopy (many chann.), Pressure Gauges	MHD Activity in Plasma Core	ECE, Soft X-Ray Array
		n_e, T_e (divertor)	Reflectom. (div) ECA (div.) Thoms. Sc. (div)		
		Impurity & D,T Influxes in Divertor with Spatial Resol.	Imp. Monitors, H_α Spectroscopy		
		Alpha Loss	Alpha-Loss Det.		
		Neutr. Fluence	Neutr. Act. Syst.		
		ELMs	ECE, Refl.(main), Magnetics		

		Sawteeth	ECE, Soft X-Ray Array		
		Erosion (plate)	Imp. Monitors, Reflectometry		

Similarly, for the total neutron flux and emission profile the target requirements are:

Parameter	Parameter range	Spatial res.	Time res.	Accuracy
Total neutron flux	$10^{14} – 10^{21}$ n s^{-1}	integral	1 ms	10%
Neutron/a source	$10^{14} – 4 \times 10^{18}$ $s^{-1}m^{-3}$	30 cm	1 ms	10%
Fusion power	≤ 2 GW	integral	1 ms	10%
Fusion power density	≤ 10 MW/m^3	30 cm	1 ms	10%

The detailed measurement requirements for all the selected measurement parameters have been determined [7.16]. They act as the starting point for the design of the diagnostic systems.

7.3. BASIS FOR PLASMA MEASUREMENTS

Plasma parameters on existing tokamaks are measured by many different techniques which in turn employ an extensive array of instrumentation. Measurements are made of the electromagnetic radiation emitted by the plasma throughout a wide range of the em spectrum, typically X-ray to microwave region. Measurements are made of particles emitted by the plasma including subatomic particles, such as neutrons and alpha-particles, and atomic particles such as neutral atoms. The magnetic activity of the plasma is measured from DC to many hundreds of kHz. In some cases, active measurements are employed where a beam of electromagnetic radiation (or particles) is launched at the plasma and an interaction between the beam and the plasma particles is employed. Every method has to have a sound physical basis and methods of interpretation which are well founded.

The subject is highly developed and a vast body of experience and literature exists: some general reviews are in references [7.17, 7.18]. Implementation of diagnostic systems on ITER mainly involves finding acceptable solutions to the major practical difficulties which arise from the harsh environment and the restricted access: for example, materials that can withstand an environment of high levels of radiation and temperatures are clearly required for the in-vessel diagnostic components. The requirements for vacuum integrity, tritium containment, maintainability of the equipment inside the tokamak and within its biological shield with remote handling equipment, are all important practical considerations. For most diagnostic techniques the physics basis is already adequate although for a few diagnostics the new parameter range of ITER has required a re-evaluation and extension of the physics. For some plasma parameters, there are serious difficulties in implementing the existing techniques on ITER and new techniques are required and their feasibility must be demonstrated. These needs and the related work in progress are described in Section 7.4.

In this section we summarize the basis of the diagnostic techniques that will be used to measure the main plasma parameters in ITER. We briefly highlight the main problems which arise in their application and we outline the solutions that are being adopted. A few diagnostics are described in some depth to demonstrate the extent of the development which has so far been achieved. Additional information on the diagnostics planned for ITER is given in Refs. [7.11, 7.19].

7.3.1 Magnetic Measurements

Several parameters fundamental to the operation and understanding of tokamak plasmas can be measured by magnetic diagnostics; in particular the plasma current, loop voltage, the plasma position and shape, and the plasma stored energy. Information can also be obtained on instabilities. The measurements are made with simple loops and coils installed at appropriate positions on the tokamak and processed with remote electronic equipment.

There is an extensive body of experience on which to base the design of magnetic diagnostics for ITER [7.20, 7.21]. The physics basis of the technique requires no new development for ITER conditions. However, the implementation requires the solution of several key technical problems including the installation of the coils and loops inside the vacuum vessel and particularly the requirement to be able to maintain the in-vessel sensors even when the structure of the tokamak has become highly radioactive; the potential change in the electrical and mechanical properties of the wires induced by the high levels of nuclear and gamma radiation; and the need to cool or protect the in-vessel sensors from the nuclear heating. Magnetic measurements are conventionally based on inductive methods so that the feasibility of very long pulses on ITER must be addressed; a supplementary technique based on microwave reflectometry is also proposed (Section 7.3.3.2).

The magnetic diagnostic system for ITER consists of several individual sub-systems: pick-up coils and voltage loops mounted on the inner wall of the vacuum vessel; coils, saddle loops and voltage loops mounted on the plasma side of the back plate of the blanket structure; coils mounted in the divertor diagnostic cassettes; four continuous poloidal (Rogowski) loops mounted on the outside of the vacuum vessel; four poloidal diamagnetic loops; and small Rogowski loops on the earth straps between the back plate and the blanket/shield modules, for the measurement of 'halo' currents. The magnetic system on the back plate and in the divertor will provide the measurements necessary for determining the plasma shape and position, plasma current and loop voltage and will permit the study of MHD modes and fluctuations. The system on the vessel wall will provide the vessel voltage, some equilibrium information near the divertor and the stored plasma energy. An overall view of the poloidal locations of the coils and flux loops is shown in Figure 7-1.

The equipment mounted inside the vacuum vessel has to fulfill requirements on vacuum integrity, radiation tolerance, temperature excursions and extreme reliability. This is solved by the choice of the construction and the materials and by using naturally shielded locations. In addition wiring, connectors and vacuum feedthroughs having similar requirements will use similar design solutions. The wires used for coils and loops are of mineral insulated (MI) cables with alumina insulation. Fixed connections between wires are welded; connectors use spring loaded contacts in alumina insulators with guiding rods to assist the remote handling.

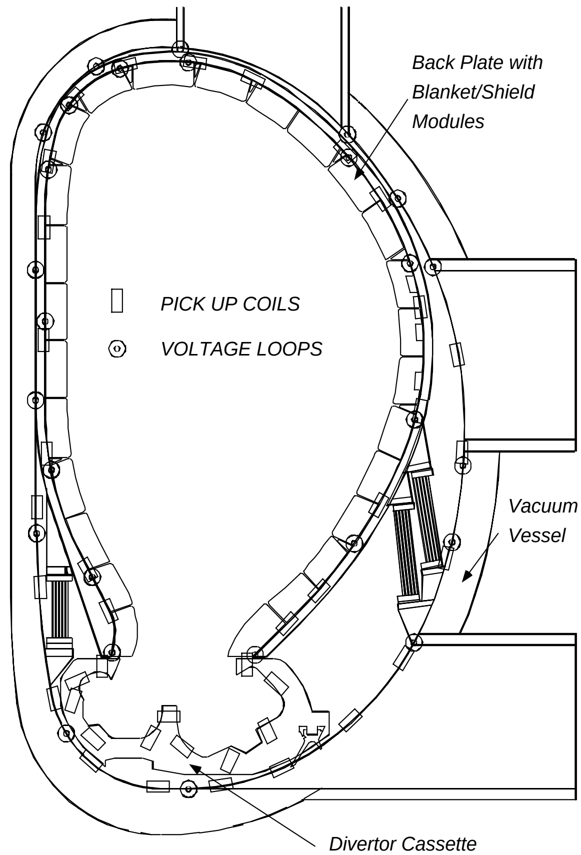

FIG. 7-1. Cross section of ITER showing the pick up coils and voltage loops on the inner wall of the vessel, on the plasma side of the back plate and embedded in the divertor cassette (saddle loops not shown). The set on the back plate (for flux and tangential field measurements), and in the divertor (for field measurements), are used for equilibrium reconstruction.

The pick-up coils mounted on the inner wall of the vacuum vessel are coils of MI cable wound on a stainless steel form with a protective cover. They are cooled by conduction. The voltage loops are also MI cables with a bridge at every sector joint which can be repaired if it is necessary to replace a sector. The coils and saddle loops mounted on the back plate and in the divertor cassettes will provide the plasma shape and position on a fast time-scale (from 10 ms and longer) and are not affected by axi-symmetric eddy currents. Special coils with high frequency response mounted near the poloidal gap between shield modules will give information on MHD modes up to 200 kHz. These coils are made from ceramic insulated wire wound on a ceramic former. The coils, the MI cable and other ceramics are at least moderately shielded by placing them in the shadow of the shield modules so that their life time is comparable or longer than the life time of ITER. The Radiation Induced Conductivity (RIC) remains below 10^{-6} S/m which is sufficiently good insulation. Necessary parts of this subsystem can be replaced when the blanket modules and/or divertor cassettes are replaced.

The flux and field values are obtained from the output of the loops and coils by electronic integration at the input circuit. For ITER with its projected pulse duration in excess of 1000s with a possible extension to steady state, classical integrators show a too large drift. Long term drift compensation schemes have recently been developed based on the intermittent observation of a dummy channel with the same integrator [7.22, 7.23]. These schemes have the potential to reduce the drift to < 1 cm for a 1000s pulse.

The system of coils and loops provides also the means to measure the MHD mode activity from locked modes to TAE-modes up to 200 kHz. Locked modes are measured by poloidally extended saddle loops with a fixed spatial Fourier analysis with m=1,2,3; n=1, low frequency oscillations by the equilibrium coils (<10 kHz); and the highest frequencies (>10 kHz) by special high frequency coils. The poloidal and toroidal extent of the coils is such that sufficient mode analysis capability ($m \leq 9$ and $n \leq 5$) is available. It is the intention to use the fluctuation measurement system as the monitor for dangerous MHD activity.

Examination of the factors which determine the lowest detectable signal (effective area of the coils and loops, gain of the front end integrator or amplifier, expected noise and uncompensated drift (off-set) of the input circuit, interference) suggest that the plasma current can be measured at 1% accuracy from 1 MA and used to control currents greater than 0.1 MA. The performance of the proposed set of magnetic sensors has been tested in a simulation of the measurement using several different reconstruction methods [7.24]. For reference plasmas, the shape and position of the plasma boundary has been determined. The equilibrium reconstruction reaches its full potential at 2.5 MA with its accuracy impaired by the small size of the plasma (large extrapolation). Numerical simulations, including statistical noise, have shown that the position of the plasma boundary should be determined to within 2 cm, which is close to the target requirement

(1 cm) for this parameter and believed to be adequate for control purposes for the full discharge duration.

The response of the diamagnetic loops will be slowed down by the presence of the back plate, which is electrically connected to the divertor cassette forming a poloidal shell. The reference toroidal flux will be provided by either a large pick up coil near the outer midplane or a loop covering the space between vessel wall and back plate. The compensation of erroneous pick up will be prepared for in the initial operation stages by measuring the mutual inductances between the diamagnetic loop and the various coils and the plasma current loop.

For very long ITER pulses (> a few thousand seconds), alternative approaches are required because of the possibility of unacceptable errors arising from drifts in the inductive methods. Direct measurement of the magnetic field using non-inductive methods are being considered. In addition, a completely independent method of determining the position of the plasma boundary using microwave reflectometry is being prepared (Section 7.3.3).

Many operational issues remain to demonstrate the full feasibility of the magnetic diagnostics. The techniques to be used for calibration of the system have yet to be addressed. Another example is a case study to determine ITER operational capability if some number of coils involved in control are damaged.

7.3.2 Fusion Product Measurements

The products of the fusion reactions - energetic neutrons, gamma rays, and alpha particles, are potentially a rich source of information on the plasma and it is expected that measurements of them will play a prominent role in the control and evaluation of the thermonuclear plasmas in ITER. As in present D-T experiments, measurements of neutron yield and of fusion power and power density are essential. The spectral width of the 14.1 MeV $t(d,n)\alpha$ neutron emission should be a reliable indicator of ion temperature in an ignited plasma. New techniques may allow measurement of steady state confined alpha particle populations. Diagnostics must also be provided which enable avoidance of first wall damage and impurity influx arising from localized losses of alpha particles.

In order to make these measurements, ITER will have a dedicated set of fusion product diagnostic systems. The proposed set includes radial and vertical neutron cameras, neutron and gamma-ray spectrometers (possibly), internal and external fission chambers, a neutron activation system, and diagnostics for confined and escaping alpha particles [7.25].

Most of the fusion product diagnostic systems planned for ITER are based upon methods commonly used in contemporary large tokamaks. However, the systems must be tailored to a much more severe nuclear environment than that encountered in present-day experiments. Although central fusion power densities in ITER will be comparable to those observed in high power D-T operation on JET [7.26] and TFTR [7.8], the neutron flux on the first wall will be ten times higher, fusion power will be 100 times higher, and the neutron yield per pulse will be more than 10^5 times as large. Design of the fusion product diagnostic systems must not only account for the higher fluxes and fluences but also adjust to constraints imposed by the attendant massive radiation shielding.

The Radial Neutron Camera [7.27, 7.28] measures line-integral neutron emissivity along collimated chords of a fan-shaped array, viewing inwardly through a midplane port and spanning as much as possible of the plasma cross section. Summation of the chordal signals, together with knowledge of the plasma major radius, gives the global neutron source strength, hence the total fusion power. Combined with data from the Vertical Neutron Camera, these measurements allow tomographic reconstruction of the spatial distribution of neutron emissivity, $n_D n_T \langle\sigma v\rangle$, which determines the alpha particle source profile and fusion power density and constrains inferred values of fuel concentrations and effective ion temperature.

The neutron cameras can be patterned after systems used on JET [7.26] and TFTR [7.8]. The principal design constraints for the cameras on ITER are (a) the resolution of interfaces with the blanket, backplate, vacuum vessel, cryostat, and bioshield; (b) the selection of detector/collimator combinations which enable coverage of a very wide range of neutron fluxes with adequate accuracy and energy resolution; and (c) the development of suitable calibration methods.

Interfaces between the neutron cameras and major tokamak components must provide unencumbered diagnostic access to the neutron emitting region of the plasma while preserving the heat load handling capability of the first wall, the radiation shielding properties of the blanket, backplate, vacuum vessel, and bioshield, and the tritium containment characteristics of the vacuum vessel and cryostat. Figure 7-2 shows the present configuration of the Radial Neutron Camera and illustrates the design approaches. The arrangement provides three separate collimated flight tubes and detector housings for each poloidal angle, thus offering a variety of choices of collimator/detector combinations.

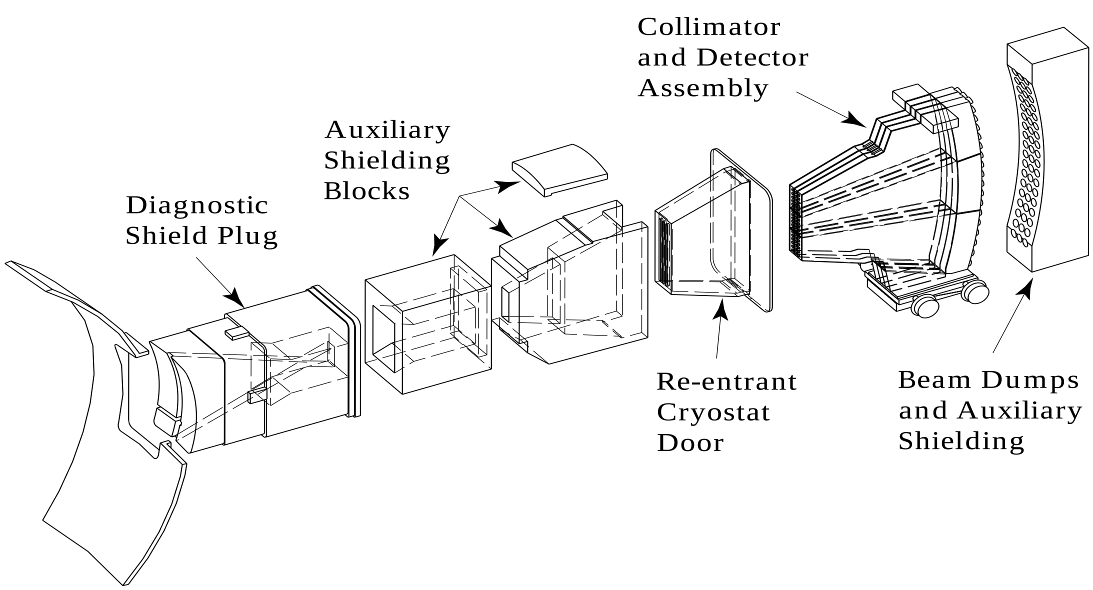

FIG. 7-2. Exploded isometric view of the proposed Radial Neutron Camera.

The Radial Neutron Spectrometer is an adjunct to the Radial Neutron Camera. Some of the flight tubes will be fitted with high resolution neutron spectrometers. In order to measure ion temperatures from the Doppler width of the 14.1 MeV emission, an energy resolution of about 3% is needed. Among systems which may have sufficient resolution are magnetic proton recoil (MPR) spectrometers [7.29], natural diamond detectors [7.30], and various time-of-flight spectrometers [7.31]. Some of the spectrometers are quite bulky and not suitable for deployment around ITER except in small numbers. The natural diamond detectors are very compact and may serve as general purpose spectrometers in conjunction with flux detectors in the neutron cameras.

Gamma-ray spectrometers [7.32] may be installed on some of the flight tubes of the Radial Neutron Camera. The measurement capability of such spectrometers in the ITER environment is under study.

The Neutron Flux Monitor System [7.33] will provide time-resolved measurements of neutron source strength. The proposed system will consist of a set of conventional U^{235} fission chambers. Each counter will be housed inside a moderator, to give flat energy response, and will be shielded against gamma-rays. Detectors containing different amounts of fissionable material, and therefore having different sensitivities, will be chosen in order to span the necessary range in count rate. The sensitivity may also be varied by locating some detectors near the plasma, *e.g.*, in the pre-shield of the Radial Neutron Camera, while other detectors will be situated in shielded locations outside the vacuum vessel.

A set of Microfission Chambers [7.34] is proposed to augment the Neutron Flux Monitor System. These are pencil-size fission chambers, similar to those used for flux measurements inside fission power reactors, and will be installed in several locations inside the ITER vacuum vessel. Two-dimensional Monte Carlo calculations suggest that detectors containing U^{238} may be placed on the front side of the back plate in the gap between blanket modules, while the area behind blanket modules is a suitable location for U^{235} microfission chambers.

The proposed Neutron Activation System [7.35] for ITER is similar to those used successfully on JET and TFTR. Encapsulated foils are transferred pneumatically to an array of irradiation stations, deployed poloidally at two toroidal locations. After irradiation for about 100 sec, the samples are retrieved to remote counting stations, where gamma-rays from the induced radioactivity will be measured. Because of the high neutron and heat fluxes on the first wall and the long pulse lengths expected in ITER, special care will be required in the design of irradiation stations and re-entrant transfer tubes and in the handling of the extremely radioactive exposed samples.

Calibration methods for neutron systems on ITER will be similar to those used on JET [7.36] and TFTR [7.37]. A variety of techniques will be utilized, including *in situ* calibration with a neutron generator, activation measurements coupled with neutron transport calculations, and laboratory calibration of individual detectors. However, the thick blanket and vacuum vessel will make some of the traditional techniques less useful than on contemporary tokamaks. A comprehensive calibration program is under development [7.38].

A consideration of alpha particle physics issues in ITER indicates the need for measurements of alpha instabilities, confined alphas, and lost alphas and suggests measurement requirements and possible techniques [7.39]. Alpha heat losses and instabilities manifest themselves indirectly by giving rise to local hot spots on the first wall, observable with infrared cameras and thermocouples, and to plasma fluctuations, observable by magnetic loops, reflectometry, etc. The direct measurement of confined alphas would be of great interest, but the present status of confined alpha diagnostics only allows a discussion of possible candidates and requires further R&D. Among the candidate techniques for monitoring the steady state alpha population are collective scattering in either the microwave range [7.40] or in the infrared [7.41] and measurements of neutron knock-on tails [7.42]. Measurements of the redistribution of alphas due to MHD events might be performed in the range r/a>0.4 - 0.5 through the use of neutral particle analyzers (NPA) with a helium diagnostic beam (200-400 keV, ~3 MW) [7.43] or by means of Pellet Charge Exchange (PCX) [7.44], but such systems are not presently planned for ITER.

7.3.3 Measurement of Electron Density and Electron Temperature

An accurate knowledge of the electron density and electron temperature is required for plasma control and for evaluating the plasma performance. On present-day tokamaks, the electron density is usually measured by interferometry, although recently reflectometry is receiving widespread application and has some advantages. The measurement of the electron temperature is usually made by the scattering of high power laser light from the electrons (Thomson scattering), and by measuring the emission due to the gyration motion of the electrons around the lines of the magnetic field (electron cyclotron emission (ECE)). Thomson scattering can also provide measurements of the electron density providing an absolute calibration of the measurement system can be achieved. Application to ITER appears feasible for all three techniques although some key technical problems have to be solved. Measurements are required in three different regions of the plasma - core, edge and divertor - and the practical difficulties are different for each region. For interferometry, reflectometry and Thomson scattering, the established physics bases are adequate for ITER conditions. For ECE the higher temperatures have required a re-evaluation of the physical basis of the method, and this has largely been carried out.

7.3.3.1 Interferometry

In interferometry a beam of coherent radiation at infrared or submillimeter wavelengths is passed through the plasma and the change of phase relative to a reference beam is measured. From the basic expression of the refractive index of a plasma, the change of phase is simply related to the line integral of the electron density: $\Delta\phi \propto \lambda \int n_e \, dl$. Measurements along several lines of sight provide sufficient information for the profile of the electron density to be unfolded by an inversion procedure. The technique is used extensively on present day devices. The technology of lasers, detectors and transmission lines are all well developed.

On ITER access constraints dictate that the plasma should be probed in the equatorial plane. A vibration-compensated interferometer is proposed [7.45]. By probing with two beams at different wavelengths, 10.6 μm and 3.39 μm, it will be possible to compensate for vibrations of the mechanical structure. Also for the proposed geometry, the deflection of a 10.6 μm beam at the retroreflector, due to refraction, is only about 0.5 mm. Further, the tangential geometry gives the possibility of making both an interferometer measurement and a measurement of the Faraday rotation using the same optics and indeed the same laser beam. With toroidal lines of sight for which the magnetic field intensity is almost constant, the Faraday rotation ($\Phi \propto \lambda^2 \int n_e \boldsymbol{B} \bullet \mathbf{d}\boldsymbol{l}$) provides a measurement of density. The measurements are complementary in that the interferometer, for a given wavelength, has better density resolution, while the Faraday rotation does not necessarily have the problem of keeping track of fringe shifts caused by rapid changes in density which is a common problem in interferometry. The radiation will be transmitted to and from the plasma through shielding labyrinths in an equatorial port and by small (~100 mm diameter) retroreflectors mounted in other equatorial ports to return the radiation. The lasers and detectors will be sited remotely.

The key practical problem concerns the plasma facing launch and collection mirror and the installation of the retroreflectors shown in Fig. 7-3. The mirrors will be metallic, actively cooled and embedded in shield modules. Preliminary thermal analysis shows that it will be possible to maintain the required optical quality of these components [7.45]. The retroreflectors will be mounted in diagnostic port plugs and will also be actively cooled. The interferometer/ polarimeter is expected to be both accurate and sensitive: the line integral of the density along each line of sight is expected to be determined to an accuracy of a few per cent while density changes of less than 1% should be measurable. To guarantee the measurement accuracy in polarimetry, a recent study [7.46] has shown the importance of compensating for the unwanted Faraday rotation, expected for a beam going through the vacuum window material in a significant magnetic field. The measurement will be adequate for the basic control of the plasma density but will not provide the spatial resolution required for some possible advanced control scenarios, and for the detailed evaluation of the plasma performance. Measurements with good spatial resolution will be made by other techniques.

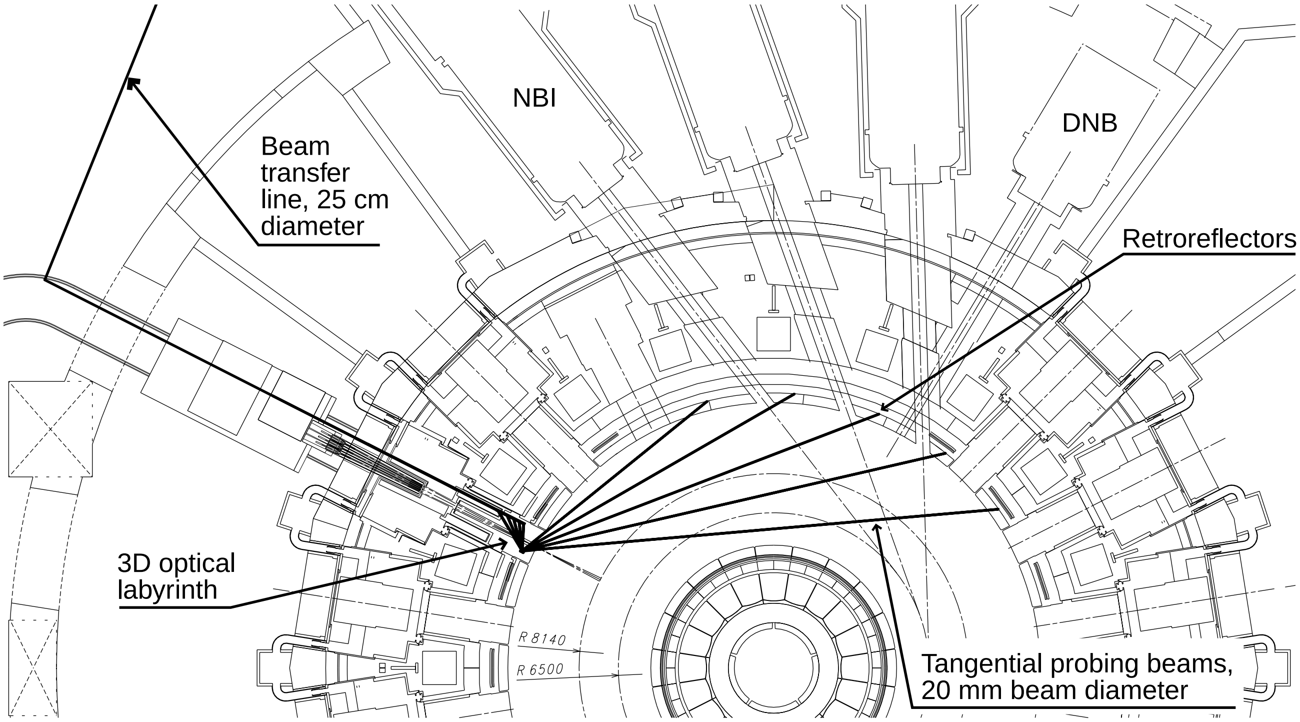

FIG. 7-3. Proposed layout of the interferometer/polarimeter in the equatorial plane

7.3.3.2 Reflectometry

Reflectometry relies on the fact that, as an electromagnetic wave propagates through the plasma, its phase is shifted due to the departure of the local refractive index from the vacuum value. At a certain critical density, this refractive index goes to zero. This density is the density of the cutoff layer. Reflectometry profile measurements are based on measuring the group delay to this critical layer as a function of frequency. In a simple 1-D picture, the density profile is then recovered by an inversion from the (group delay, frequency) space to the (cutoff frequency, radius) space. Measurements of density fluctuations can be made by holding the frequency constant, and observing the time evolution of the phase or group delay. The inversion to obtain the density fluctuation level is not straightforward except for long wavelength, low amplitude modes for which simple 1-D theory can be adequate.

The basic physics of the technique is well established and has been confirmed by experiments [7.47]. A wide range of appropriate hardware exists commercially and measurement systems can be readily designed and built. As a result, reflectometry is developing into the instrument of choice for the measurement of edge profiles with high resolution. However the diagnostic is only now becoming routinely used. This is largely due to the recent development and application of strategies to avoid or minimize the effect of density fluctuations on the measurement. Continuing operation of the measurement through ELMs has been demonstrated on DIII-D [7.48], but further experimental demonstration is required.

For ITER, the main reflectometer system will be used for the systematic measurement of profiles, MHD activity and transient events near the plasma centerline. In addition a new application of reflectometry is planned to supplement the magnetics for position control, by measuring a portion of the density profile near the separatrix to deduce the plasma-wall distance at selected points around the periphery of the main chamber, and a reflectometer system to measure profiles in the divertor legs [7.49]. The details of the reflectometers to be used in each region depend on the key factors of accessibility to the cutoff, constraints on the hardware and present operational experience. The degree of extrapolation from present knowledge to the state required for ITER measurements varies by system and will be discussed below.

To measure the full density profile in the core plasma for typical ITER plasma parameters (see Fig. 7-4), it is proposed to use both O and upper X-mode cutoff from the low field side, and the X-mode lower cutoff from the high field side (see, e.g. [7.50, 7.51]). Present operational experience of low field side measurements has been used to arrive at the corresponding ITER design with some confidence. Multiple sightlines will accommodate vertical plasma movements. When paired, the same sightlines can obtain the propagation characteristics of low (m, n) modes. For example, both the O-mode (e.g. [7.52]) and the upper X-mode cut-off [7.53] have been used for measurement of high-frequency TAE modes. To reduce the likelihood of unwanted reflections, and have the side benefit of allowing local measurements of higher (m, n) mode wavenumbers, separate transmit and receive antennas will be used on each sightline. On the high field side, there is limited but good experience in O-mode for the measurement of profiles [7.54], and fluctuations [7.55], and in X-mode (lower cutoff) for the measurement of core fluctuations [7.53]. The present ITER design is integrated with that of the reflectometer for plasma position (see below). It is unlikely that transmission lines optimized for this measurement can be installed, so that to proceed with any confidence will require the testing of a prototype on a plasma.

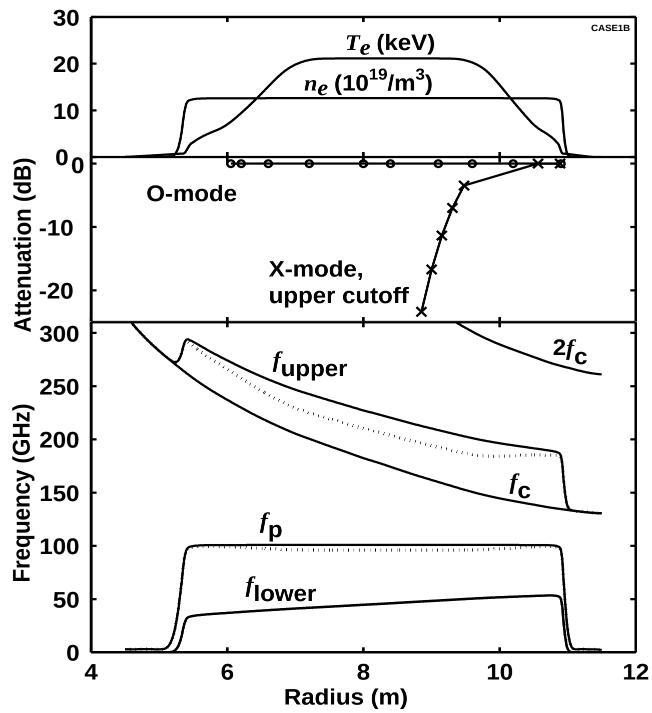

FIG. 7-4. Accessibility diagram for the standard ITER plasma near the midplane [7.50]. The high T_e in the confinement zone, in addition to altering the cutoffs, results in prohibitive absorption inside r/a~0.6 for X-mode launch aimed at the upper cutoff. O-mode is capable of accessing the core, provided that the peak density remains below ~2×10^{20} m^{-3}. To measure near the core given a relatively flat density profile requires the use of X-mode launch from the high field side, aimed at the lower cutoff layer.

In the proposed application for measurement of plasma position and shape [7.49, 7.56], an O-mode system has been selected, because of the important benefits of ruggedness and independence from the local magnetic field. The antenna and transmission line are heavily constrained by the first wall and backplate construction. The high field side waveguides are routed in a set of small conduits between the blanket and backplate structure, and they have to view the plasma through the toroidal gaps between blanket modules (20 mm). The limited conduit space is shared with the high field side components of the main reflectometer [7.57]. A thorough prototyping of this technique is required since issues such as whether the edge density may not be a flux function can be resolved.

For the divertor plasmas, it is proposed to use both polarizations to reconstruct the density profile accurately, along multiple sightlines to give low (~ 0.2 m) resolution along the leg [7.57, 7.58]. The divertor will pose the hardest hardware challenge, both near the front end and in the instrumentation. It combines the difficulty of obtaining sight-lines through the surfaces with the highest heat loads on ITER, which restrict views to poloidal slits (<20 mm), with very wide (20 GHz-1 THz) frequency range. Additionally there is a limitation on the total number of transmission lines that can be fitted to the two available ports. With the exception of the sub-mm domain and the transmission inside the cassette, the hardware is a small extrapolation of present systems. In the sub-mm domain, simplified measurements of the first few moments of the density profile across the leg are likely to be used instead [7.57, 7.58]. Operational experience of profile reconstructions is limited, but shows that measurements are possible with accuracy comparable to reflectometers in the main plasma region [7.59]. A simple "comb" system to measure the peak density has been used with some success on JET [7.60].

Of the ITER requirements regarding accuracy and resolution [7.16], most have been demonstrated. In particular even the fastest time resolution requirement of 1 ms (for the divertor) has been exceeded for profile measurements by a factor of >~50 on many devices (see e.g. [7.61]). Real-time calibration will be essential for the long ITER pulse. In general, the biggest uncertainty in predicting ITER performance is the effect of density fluctuations, which link the time resolution to the reconstruction errors. Experience to date suggests that, by using appropriate averaging techniques, the profile errors can be reduced to below the ITER requirements within the required time resolution. Modeling efforts in this area are beginning to produce results [7.62, 7.63] and will be useful in extrapolating to ITER conditions. The first results of 2D full wave studies on the effect of the violation of the WKB approximation at low frequencies and steep profiles suggest that it is not a concern [7.64]. The remaining uncertainty is in the semi-systematic errors due to profile initialization and the optimization of the antenna distribution, which are particularly important for the plasma position system, although its localization requirements are relatively weak (10–20 mm). Both are the subjects of ongoing study [7.48, 7.56, 7.65].

7.3.3.3 Thomson scattering

In Thomson scattering radiation from a high power laser is injected into the plasma and the radiation scattered incoherently by the electrons is measured. The electron temperature is obtained by measuring the spectral width, and the electron density is obtained by measuring the intensity, of the scattered radiation [7.66, 7.67]. Two different approaches have been developed for the implementation. In one case, the input beams and collection beams follow different paths (frequently orthogonal) and the measurement is made at their intersection. The spatial resolution is provided by the image of the intersections. This is the 'conventional' arrangement and is the one that has been most widely applied. In the second case, the input beam and the collection beam follow the same path, the backscattered radiation is measured and spatial resolution is obtained by the time-of-flight principle. This is known as LIght Detection And Ranging, or LIDAR, technique [7.68]. Each technique has its own merits and which is best for any particular application depends on the measurement requirements and the practical circumstances. The conventional arrangement gives better spatial resolution relative to LIDAR which is limited mainly by the time response of detector. The LIDAR technique has the advantage that only one optical beam is required so that it is not dependent on the demands of image focusing, which will be affected by any movement of

ITER structure, and makes it possible to reduce the alignment requirements significantly. In general, if the LIDAR technique can provide the required spatial resolution then this is the preferred technique.

Common to all Thomson scattering set-ups is the need for calibration on a regular basis. This calibration is two-fold: since the electron temperature is deduced from the shape of the scattered spectrum, a calibration of the sensitivity of the spectral channels relative to each other is required. More challenging is the calibration of the absolute sensitivity of at least one spectral channel which is necessary for the density measurements. In existing tokamaks this is usually carried out by filling the vacuum vessel with a neutral gas of known pressure and performing Rayleigh or Raman scattering. Both calibrations have to include all the optics, including the inaccessible front-end optics.

For ITER, Thomson scattering can potentially meet the requirements for measurements in the plasma core, edge and the divertor regions. The measurement requirements and practical circumstances are very different in each region and so different systems are envisaged employing their respective advantages.

In the core the requirement is to measure the electron temperature and density over the entire plasma cross section with a spatial resolution of 10 cm. A LIDAR system is well suited for this measurement and so this is the approach that is being pursued [7.69]. The system under design consists of the laser and detection sub-systems located outside the biological shield, the optical transmission lines for the laser and the collected light, a common window for input and output at the flange of the standard equatorial port, and collection optics located within the port.

Figure 8-5 shows an overall view of the front part of the system. Both the laser beam and the collected light pass the same optical components, so a movement of any of these components does not lead to misalignment between them, provided the laser beam and the collected light are aligned outside the biological shield. The optical system points radially inward towards the tokamak with the chord passing the plasma axis. To reduce the escaping neutron flux through the blanket penetration a 3-fold labyrinth with minimized cross section is used.

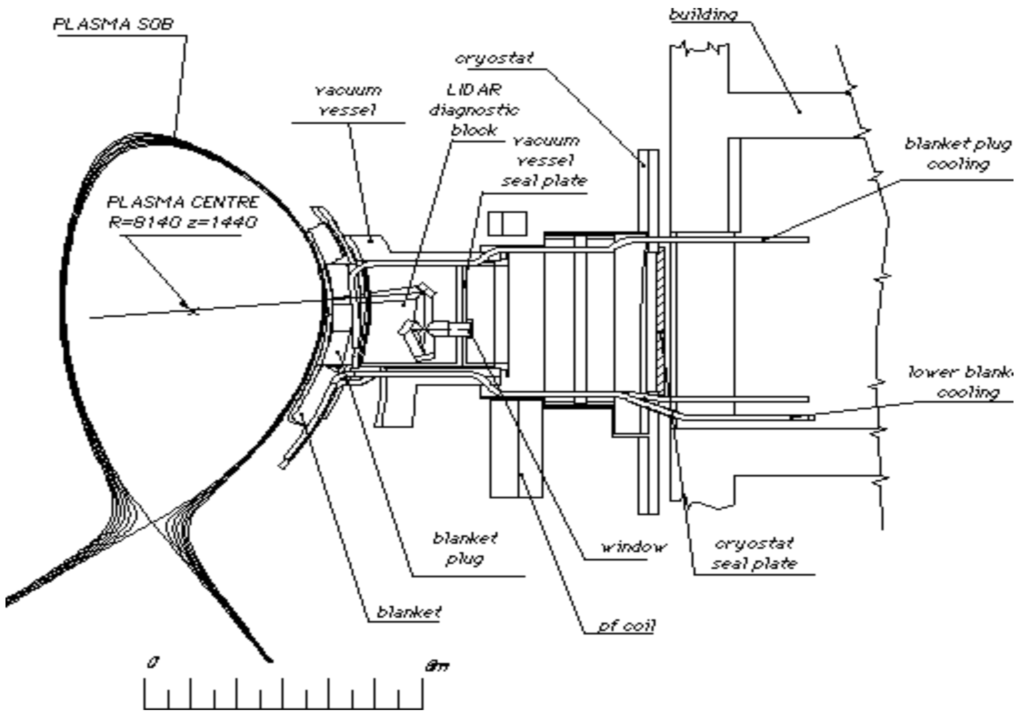

FIG. 7-5. LIDAR diagnostic, front end.

The space inside the cryostat is bridged by two mirrors which image the primary vacuum window onto the cryostat window. The alignment of these mirrors is actively controlled to cope with the movement of the vessel with respect to the cryostat (and building). Outside the cryostat both the incoming laser beam and the outgoing collected light pass the biological shield via a second labyrinth. From thereon two separate optical transmission lines extend to locations in the building where the laser and the detection equipment can be operated hands-on.

The most critical element in the system is the front end mirror. This has to face the plasma and to transport the high power laser beam and scattered radiation and must maintain a high optical quality throughout the visible region. The mirror is subject to charge exchange neutral erosion and to surface heating from the high power laser, both of which can lead to damage. It is also subject to bulk heating from the nuclear radiation which can lead to distortion. In the design, the laser beam is expanded as much as possible to reduce the surface heating and the front surface is coated with a thin layer of rhodium, a material which has a high reflectivity over a broad spectral range and a low sputtering coefficient [7.70, 7.71, 7.72]. The mirror is actively cooled to cope with the bulk heating. The investigations have shown that the sputtering and nuclear heating problems have been solved but more information is needed on the laser damage threshold; this is a subject of on-going R&D.

The relative spectral sensitivity of the detection system will be achieved by illuminating the system with a calibrated, tunable, monochromatic light source. This light source will be placed outside the vessel and thus the transmission of the window and of optical in-vessel components cannot be re-measured after the startup of the plasma experiments. However, the changes in the transmission of inaccessible optical components whose spectral calibration may change over time (e.g. vacuum windows, mirrors inside the vacuum system), can be determined by making the scattering measurements with two lasers at different wavelengths [7.73]. Consideration is being given to using this method on ITER.

The absolute calibration can also be performed by comparison with other reliable electron density measurements. Since with the LIDAR scheme the whole electron density profile is measured with only one detection system, and since the variation of the solid angle of collection along the line of sight is well known, the calibration can be performed either against the density measured at a point in space with another diagnostic (e.g. reflectometry), or against the line integral of the density as measured by interferometry/polarimetry. This method is being used on JET [7.68]. However, stand-alone calibration of the system should also be possible by performing pure rotational Anti-Stokes Raman scattering from a low pressure deuterium fill of the vessel [7.74, 7.75].

Some elements of the design remain to be completed, especially the engineering details of many of the components, and some further R&D is required, but it is believed that the system will be able to meet the specified measurement requirements.

In the edge region it is required to measure the electron temperature and density over a relatively small spatial range (typically 0.3m) but to a high spatial resolution (5 mm at the midplane). A conventional arrangement is best suited to this measurement. Access through a vertical port allows the measurement to be made at the top of the plasma where the physical size of the edge region is expanded by about a factor of three due to the flux expansion. This is the geometry being developed for ITER [7.76]. The laser beam is introduced through one labyrinth and the scattered radiation is collected and transmitted with a second. After passing the vacuum window a transition to optical fiber is made, working in the range 800-1200 nm where luminescence and darkening due to nuclear radiation are known to be low. The study has shown that the measurement requirements can be met in principle but the details of the implementation have yet to be developed.

For the divertor region two systems are being investigated: one probes the plasma near the X-point while the other probes the plasma in the divertor legs. In the X-point region the flux surfaces are expanded by about a factor of 10 and are therefore accessible by LIDAR [7.77]. The access for this system is provided by the 2 cm wide, 40 cm high slot between a target plate of a diagnostic cassette and its neighbor. The long narrow slot leads to a light collecting mirror of 10×45 cm^2, which has a curvature in the horizontal direction of 5.3 m. A 3 J laser output is required to achieve a sufficient SNR. At the low density of 10^{19} m^{-3}, this system is at the limit of the SNR so extremely short pulse lengths are required, and a new laser and the use of a streak tube are being investigated.

Along the divertor legs, a good spatial resolution in the measurement of the temperature and density is required for the evaluation of the divertor performance and in particular for the detailed evaluation of the spectroscopic measurements. The specified resolution is 10 cm along and 3 mm perpendicular to the flux surfaces. The only possibility for making the measurements is a conventional Thomson scattering system used in combination with a reflectometry system in selected locations. A feasibility study of such a system is currently in progress.

7.3.3.4 Electron cyclotron emission

The frequency of electron cyclotron emission (ECE) is directly related to the local field with $\omega = neB/m_e$, where n is the harmonic number. In large tokamaks, the plasma is optically thick to the ECE radiation. As a result, the emission region behaves as a black body, and the intensity of the radiation is related directly to the local electron temperature by the standard black-body emission formulae. The technique consists of measuring the intensity of the emission as a function of the radiation frequency, and carrying out a frequency-to-space transformation to obtain $T_e(R)$. The spatial dependence of the magnetic field is used in the transformation and this is obtained from the magnetic diagnostics and equilibrium codes.

Measurements of ECE have been made on all major tokamaks for the last 20 years. A large body of experience has been acquired [7.78]. The microwave transmission system, measurement instrumentation, calibration methods and interpretation techniques are all highly developed. More recently, ECE has been used successfully on several machines for the measurement of small amplitude temperature fluctuations, using correlation techniques [7.79]. Finally, the theory of ECE in thermal plasmas is well developed [7.80]. Thus the design is based on reliable predictions of the emission and a long history of experimental success.

ECE measurements on ITER will occupy a similar role to that which they have on existing tokamaks: they will provide detailed information about the spatial profile of the electron temperature and its temporal evolution. The measurements will be made in both the core and edge plasma, and will encompass not only the high performance phase of the discharges, but also the startup, ohmic and termination phases. The entire ECE spectrum will be measured, and this will allow estimates of ECE power loss to be made.

The parameters of the ITER plasma, in particular its large size, small aspect ratio and high temperature, have a significant impact on the key issues of accessibility and resolution for T_e measurements. Detailed modeling has shown that relativistic effects become extremely important for accessibility and spectra of the ECE [7.81]. For the second harmonic extraordinary mode, harmonic overlap sets a limit on access to the central region of the plasma which becomes increasingly severe at high T_e. The spatial resolution of the first harmonic ordinary mode is somewhat poorer, but most of the plasma cross-section is accessible. The steepness of the gradients of T_e and n_e in the plasma edge determines the resolution limits and the lowest T_e for which well localized measurements are possible [7.81, 7.82].

The instrumentation requirements for ECE can be met by limited extrapolation from existing diagnostics [7.82]. However, major challenges in this system arise from the distance between the plasma and the measuring instruments (long transmission waveguides) and the access restrictions on the machine. The preliminary design study indicates that it will be possible to install antennas, mirrors and windows which meet the ITER vacuum and radiological constraints, and which will have the required narrow antenna pattern. Absolute spectral calibration of the system is essential for independent T_e measurements. The design includes a dedicated calibration source built into the optics in the port plug to ensure that the accuracy of the ECE measurements can be maintained over the life of the machine [7.83].

The spatial resolution target (~0.1 m) for T_e measurements [7.16] should be met for the core plasma, but in the edge the resolution is predicted to be ~0.05 m, significantly short of the target value. The calculated effective minor radius resolution for the core and edge regions is shown in Fig. 7-6 [7.80]. Depending on plasma profiles, it may not be possible to measure T_e values lower than 1-2 keV in the plasma edge. The accuracy of the T_e values from the diagnostic will be largely determined by the residual systematic errors in the calibration system. Experience on present machines suggests that ±10%, and possibly better, can be achieved. The temporal resolution of the measurements will determined by the instrumentation. It should not be difficult to achieve the target value [7.16] of 10 ms. However the use of ECE for the measurement of small scale temperature fluctuations (micro turbulence) will be limited by the spatial resolution.

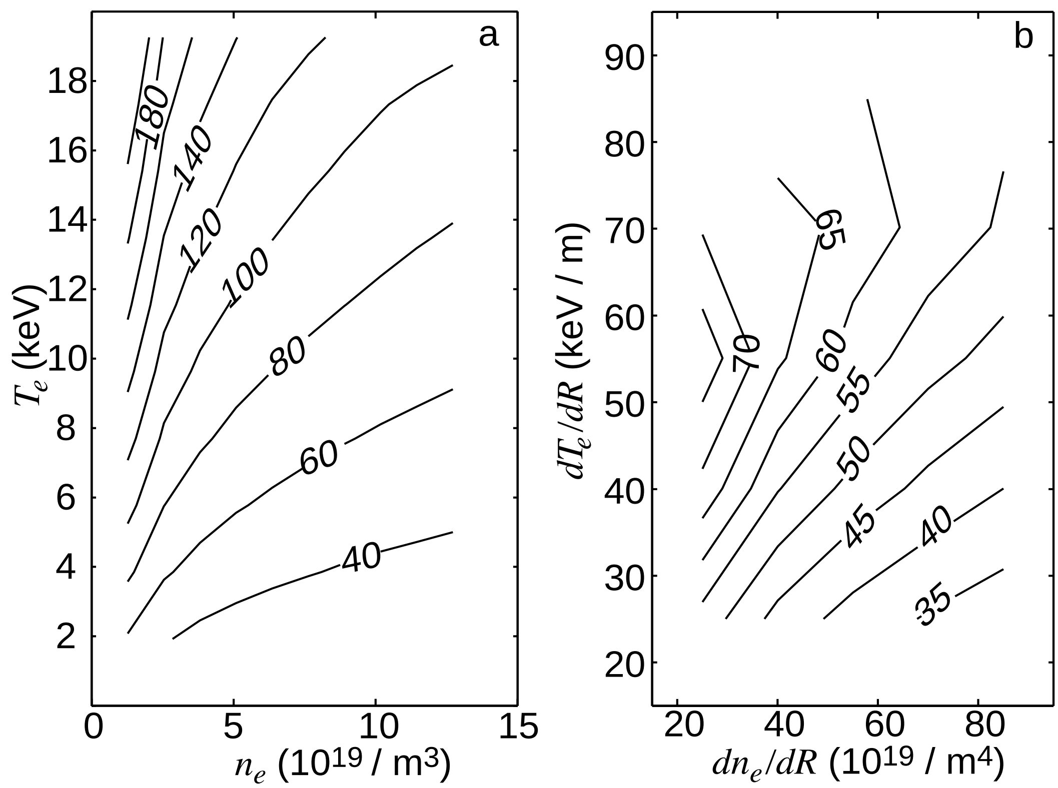

FIG. 7-6. Contours of effective resolution in minor radius (in mm) in X-mode for (a) the core region (ψ = 0.25), as a function of peak temperature and density, with the profiles scaled from an ITER reference case with a peaked temperature profile (CASE 2B, T_o =28 keV), and (b) the edge region, as a function of temperature and density gradients at a fixed emission temperature of 2 keV. The resolution in O-mode is somewhat poorer, but accessibility extends to ψ =0 (accessibility in X-mode is limited by harmonic overlap).

7.3.4 Measurement of Radiated Power

In ITER a major fraction of the power from α-particle heating must be radiated to obtain sustainable divertor operation. This will be achieved by impurity radiation from the edges of the main plasma, the region near the X-point and in the divertor legs. In the preferred operating point of semi-attached plasmas in both divertor legs the main contribution to the radiated power comes from the divertor. However, for slightly higher densities the power is radiated from the X-point. For the control of the operating point the amount and the balance of the power radiated in each of these areas is of critical importance. In addition to this control, measurements of the radiated power are required for evaluation purposes, in particular spatially resolved measurements are required in studies of the power balance.

The measurements will be made with an extensive bolometric system. The specification of the system is to provide the radiated power with a spatial resolution of 20 cm in the main plasma and 5 cm in the divertor and X-point region. The proposed method, as used on many contemporary tokamaks, is sparse data tomography. With this method use is made of the magnetic configuration [7.84] to reduce the large number of channels which would otherwise be necessary to a manageable quantity. The total number of lines of sight (LOS) proposed for ITER is approximately 300.

The bolometer arrays will be installed in the equatorial and vertical ports, and in the specially instrumented divertor diagnostic cassettes. Design options for installing them at selected locations on the back-plate viewing through the gaps between adjacent blanket/shield modules are also being explored. From each of these locations several arrays of LOS observe the plasma. From the equatorial port, the inner divertor leg (high resolution) and the main plasma are viewed; from the vertical port, the main plasma, the area of the X-point and the largest part of the divertor legs (high resolution) can be seen. This last view can integrate the total radiated power. The bolometers mounted on the back-plate view the plasma through the poloidal gaps between adjacent blanket/shield modules. This provides some shielding from the nuclear radiation.

The reference detector design is based on the bolometer used on JET [7.85]. It has a resistance bridge on a mica substrate. The radiated power ($\lambda \leq 200$ nm) is absorbed by a gold film and the temperature rise of the thin substrate is measured by a classic resistance bridge. The edges of the foil are cooled to obtain the desired frequency response. One branch of the bridge is shielded from the optical radiation and used to automatically compensate for variations in temperature caused by such sources as nuclear heating. In the divertor the bolometer measurement will be affected by the high neutral pressure from two effects: the CX neutrals will deposit their energy on the foil and the high local cold gas pressure causes extra heat removal by conduction. Both effects can cause substantial errors. By installing a number of channels looking into the divertor along typical affected chords from different sites (ports and back plate) it is possible to successfully correct for these effects.

The resistance bolometer offers also the possibility of in-situ calibration and a further feedback loop to stabilize the operating point. The JET bolometer has a proven performance in an ITER-like situation but the degree to which it is radiation hard is not known and this is being studied in on-going R&D. The mica substrate will probably have to be changed to ensure survival in the ITER nuclear environment. A significant design issue for the large number of proposed bolometer arrays is the integration of the detectors with mineral-insulated cabling, connections capable of being made-up remotely and vacuum feedthroughs.

A new concept is under consideration: in this arrangement the temperature rise of a foil is measured by a remote IR detector [7.86]. This has the advantage of a detection without interference but the sensitivity is lower and the signal transmission suffers from restricted access. This concept can be made into a 2D matrix and is therefore suitable as a survey instrument.

7.3.5 Measurement of Plasma Impurities

Although ITER will be fueled with a mix of pure D and T, other elements will enter the plasma due to the radiation and particle bombardment of the surfaces of the in-vessel systems. From experience on existing machines, it is anticipated that the principal intrinsic impurities will include Be, B, C, N, O, Fe, Cr, Ni, Mo, W and He-ash. In addition some impurities will be purposely introduced for control of radiation and detachment (N, Ne, Ar, Kr). The identification and control of these impurities will be important for the control and optimization of ITER plasmas.

The evaluation and understanding of the plasma performance requires a knowledge of more detailed information such as the influx and concentration of the different impurities.

The impurities can be diagnosed by making measurements of the line radiation. At the temperatures expected in ITER, most of the line radiation occurs in the UV and visible regions of the spectrum but useful information can be obtained by making measurements throughout the x-ray to infrared regions. Edge and divertor influxes can be monitored with visible, near UV, vacuum UV and X-ray spectroscopy. Core levels of higher Z impurities, which are not completely stripped at $T_e = 20$ keV, can be monitored with X-ray crystal instruments [7.87]. Such measurements are made routinely on existing tokamaks and there is a large body of experience on which to build the designs of systems for ITER. The majority of outstanding technical problems relate to interfacing of plasma facing components, primarily mirrors for the IR/visible/near UV, and direct vacuum interfacing for the vacuum ultraviolet and soft x-ray spectral ranges. A review of the relevant atomic processes and many of the diagnostic applications for tokamak plasmas can be found in [7.88] and references therein.

Because of the need for two dimensional array measurements in the divertor, it is likely that most of the spectroscopic measurements in this region of the plasma will rely on visible and near IR instruments ($\lambda > 200$ nm), as summarized in [7.89]. Implementation of VUV measurements for this region is very difficult and it may not be possible to realize an acceptable implementation although an attempt to design a system is being made. The lack of VUV measurements complicates the interpretations of impurity density measurements, especially in the hotter (T_e>10 eV) regions of the divertor plasma [7.90, 7.91], and the inclusion of vacuum interfaces, either with direct views or more probably using multilayer-mirror-based reflective optics [7.92], could significantly improve this situation. In all cases, mirror robustness, particularly in the face of major disruptions, is a key technical issue which is now being addressed (Section 7.4.3).

A useful parameter for characterizing the impurity content of the plasma is the effective ionic charge $Z_{eff} = \Sigma n_z Z^2 / \Sigma n_z Z$ where the summation is carried out over all charge states of all ion species. For a pure plasma $Z_{eff} = 1$. A typical operating value is Z_{eff} between 1 and 2. Profiles of

Z_{eff} can be measured using the visible continuum intensity, which is dominated by bremsstrahlung emission over most of the plasma profile [7.93]. The present design [7.94] assumes a tangentially viewing array operating around λ = 523 nm, which is found to be free of significant line emission. The near-IR has also been used [7.95, 7.96], and could also be considered for ITER. The design of the plasma-facing mirror is an outstanding issue, as is measuring and maintaining the absolute calibration of such an optical component. The latter has been an long-standing problem for operating devices since venting of a well-conditioned tokamak is only undertaken very reluctantly.

In addition to impurity identification and concentration measurements passive spectroscopy can provide ion temperatures and flows, and the electron temperature and density. The ion temperature is determined from measurements of the spectral width of the impurity lines broadened due to the doppler effect while the flows are determined from the line shifts. The electron density and temperature can be obtained by fitting model spectra to measured spectra in conditions where Stark broadening is significant, but this is generally in a limited parameter range. If available, this information is used to supplement that obtained by other techniques.

Fully stripped low Z elements do not radiate and so cannot be diagnosed using passive techniques. A complementary technique of active spectroscopy has been developed for the diagnosis of this plasma component. In this technique a beam of neutral atoms is injected into the plasma and the light emitted by the impurity ions following charge exchange reactions is measured. The line of sight of the viewing optics is arranged to be orthogonal to the beam so that spatially resolved information is obtained. Local plasma parameters which can be measured with these techniques include low-Z impurity densities (e.g., [7.97–7.99]) and impurity transport coefficients (e.g., [7.100]), ion temperature and rotation (e.g., [7.101]), internal magnetic field, (and thus current density and safety factor [7.102–7.105], electron density fluctuations (e.g., [7.106]), and the energy distribution of slowing-down D-T fusion alpha-particles [7.107]. Helium ash accumulation is one of the key physics issues to be tested in long pulse (>1000 second) ITER operation. A diagnostic neutral beam (DNB), combined with charge exchange recombination spectroscopy (CXRS) will be able to provide this measurement. According to a recent assessment the optimum energy for the beam is about 100 keV [7.108]. Since this is significantly less than the energy of the ITER heating beams (1 MeV), a neutral beam dedicated to these measurements is required. (Note that the higher energy beam is much better for the MSE measurement of q-profile.) Such a beam, employing a conventional 5 MW positive ion source neutral beam, with 50% modulation, should provide the helium ash density profile with signal to noise ratio greater than 10 for r/a >0.3 and time resolution of ~0.1 ms [7.109]. The application of an ultra-high power pulsed neutral beam (10^6 A/cm^2 equivalent neutral current, as proposed in ref. [7.110] and [7.111]), would increase the S/N ratio, with a gain of more than a factor of 10 over much of the profile for a 300 pulse/second beam. It has also been pointed out in reference [7.109], that interference from C and Be lines may interfere with the He measurements if the n=4 to n=3 transition at 468.6 nm is used, and consideration should be given to using the 5 to 4 transition at 1013.6 nm.

Local measurements of other light impurity ion densities (Be, C, O, Ne), along with impurity ion temperature and rotation profiles will also be possible utilizing CXRS. Because of the need for spectral resolution to measure the doppler widths and shifts, the temperature and rotation measurements using the conventional beam will have good signal to noise only outside of about r/a = 0.5 for 0.1 s time resolution, at least for the highest density plasmas considered ($n_e = 1.4 \times 10^{20}$ m^{-3}).

Active spectroscopic measurements may also be possible using laser induced fluorescence (LIF). This technique, currently undergoing development, might be applied to measurement of atomic hydrogen densities in the divertor and edge region of the confined plasma [7.112 - 7.115].

7.3.6 Measurement of Ion Temperature

Frequently the ion temperature $T_i \neq T_e$ and so an independent measurement of T_i is required. For ITER no one measurement technique can meet all of the measurement requirements. This is also usually the case on present day tokamaks. On these tokamaks information on T_i is obtained by combining measurements made by several different methods and this will be the approach adopted on ITER.

Measurements of T_i in the edge region of the plasma can be made by analyzing the energy distribution of charge exchange fast neutrals which escape from the plasma. The measurements are made with a Neutral Particle Analyzer (NPA). Two analyzers are being designed for ITER [7.116]: one is for the energy range 10–200 keV and the other is for the energy range 0.5–4 MeV. The low energy analyzer will provide the measurements of T_i but because the penetration of neutral particles is limited the measurements will only be available in the outer region of the plasma. The analyzer enables the different hydrogenic species to be sampled and so measurements of the fueling ratio n_D/n_T will also be obtained. The high energy analyzer will provide measurements of the confined alpha particles by measuring the energy distribution of alpha particles that become neutralized by charge exchange and then escape from the plasma.

T_i in the plasma core can be determined from measurements of the width of impurity lines as outlined above (Section 7.3.5). The passive spectroscopic measurements provide T_i at the location of the emitting impurity ion. Since there is a temperature gradient, different ions, and different charge states of ions, are located at different radii and so some spatial resolution is possible. Active spectroscopy with the DNB will give the best spatial resolution but the depth of the plasma that can be probed is limited because of the attenuation of the beam. Even with the use of the proposed ultra-high power pulsed beam, the measurements of T_i and rotation will not come close to meeting the specified requirements very close to the axis.

Measurements of the width of the energy spectrum of the neutrons produced in the thermonuclear reactions can give T_i provided that the ion distribution is thermalized. In practice this means that the technique can only be applied to ignited plasmas. Limits in the resolving power of spectrometers will limit the lowest temperature that can be measured to about 3 keV. The measurement along any given line of sight is not localized but by making measurements along many lines of sight the T_i profile can be constructed. Finally, under some conditions, the ion temperature can also be obtained from measurements of the integrated neutron flux.

All of these techniques will be applied on ITER and specific information on T_i will be obtained. However, it will probably not be possible to meet all of the measurement requirements especially the required spatial resolution (30 cm).

7.3.7 Measurements of q Profile

On present day tokamaks, measurements of the q profile are obtained by Multichord Polarimetry at infrared/millimeter wavelengths and by Motional Stark Effect Polarimetry (MSE) with an energetic neutral beam. Both techniques are in principle applicable to ITER but there are significant practical difficulties in each case.

The Multichord Polarimetry technique would be best carried out using laser radiation with a wavelength in the range 10 - 100 μm [7.117]. The plasma would have to be probed along several different chords in the poloidal cross-section. In this radial plane, the rotation of the polarization is dependent on the strongly varying poloidal magnetic field, unlike the measurement described in section 7.3.3.1 where polarization is dependent on the toroidal magnetic field which is almost constant. At these wavelengths it should be possible to transport, launch and receive the radiation without significant difficulty and many of the solutions developed for the density interferometer (7.3.3.1) can be applied. However, it will be necessary to reflect the radiation from the high field side of the tokamak and this will require retroflectors installed in the vessel. The installation and alignment onto these retroreflectors are difficult tasks and are being pursued in present design work. The capability for viewing a sufficient fraction of the cross-section of the plasma to make this technique fully viable has not yet been shown.

In MSE a beam energy in excess of ~500 keV is highly desirable and so use of the negative ion source heating beam will be required [7.118]. Preliminary estimates for such a system indicate that in principle 10 cm spatial resolution and 10 ms time resolution are achievable, and q(0) on axis could be measurable with an uncertainty of 10 % or less. Direct measurement of radial electric field may also be possible using MSE; however, in this case, either beams of different energy must be available, or separate co- and counter-injection is required. In some cases, with transport barriers or negative central shear in the plasma, the effects of radial electric fields may have to be taken into account, if they become significant compared to the Lorentz field seen by the beam particles, in the interpretation of the MSE data [7.119]. In order to make the measurement it is necessary to install a photoelastic modulator (PEM) in the optical train in the diagnostic port and the lifetime of this component may not be sufficient. Other issues concern the access for sufficient sightlines and demonstration that the polarization of the emitted light is preserved through the proposed reflective optics system. A disadvantage of the MSE technique is that it relies on the heating beam and this may be required for other control tasks, for example for inducing plasma rotation, and so the use of MSE as a control diagnostic could be limited.

7.3.8 Operational Parameters

As in present tokamaks it will be necessary to measure a range of parameters to aid the operation of the tokamak and, in some cases, to protect the internal structures. For example, measurements will be required of the temperature and condition of the high heat flux components on the first wall and in the divertor. In some current tokamaks these are observed using visible/IR cameras. Measurements are made during the pulse and recorded on video tape. For many years TFTR has had a system for wide angle viewing for survey work and a telephoto capability for detailed observations [7.120] and a better system with in-vessel optics has recently been developed for JET [7.121]. A system is being designed for ITER based on the experience with the JET system [7.121]. A major difference from the JET system is that the front end optics are entirely reflective. This is to avoid problems which would occur in refractive optics due to radiation enhanced absorption.

Similarly it will be necessary to make measurements of runaway electrons since they can potentially damage the first wall. The established method of monitoring the presence of runaway electrons is to measure the hard X-ray emission but this will be difficult to apply in ITER because of the high gamma background and the very thick wall and in-vessel shielding thickness. Measurements of the synchrotron emission at infrared wavelengths have been successfully made on TEXTOR [7.122] and similar measurements should be possible on ITER.

Additional operational parameters which will have to be measured are the electron density and temperature of the plasma near the divertor target plates, the neutral gas density in the boundary of the plasma and in the divertor, and the composition of the base gas. The plasma near the divertor plates will be probed using specially-developed Langmuir probes, designed to minimize erosion and built into the sides of the divertor cassettes [7.123]. The neutral gas density will be measured using a pressure gauge of the ionization manometer type. This type of gauge is used extensively on present-day tokamaks and, despite some problems with short filament lifetime in present use, should be suitable for ITER with minimal development [7.124]. The composition of the base gas will be analyzed with standard residual gas analyzers with the addition of optical spectroscopy of a Penning discharge, recently demonstrated at JET [7.125].

7.4. R&D NEEDED FOR DIAGNOSTICS

The measurement requirements necessary to carry out the ITER physics mission, given in Section 7.2 set very demanding goals for diagnostics. As design of the individual diagnostics has progressed, the potential quality of the measurements has become clearer. Because of such issues as the restrictions in access or neutral bombardment of the mirrors closest to the plasma, it is possible that some of the measurement requirements will not be met. Some of these problems are well known [7.126], but the design effort is making others clear. The descriptions in Section 7.3 indicate some of the limitations and concerns, some of which may be alleviated by R&D into specific components or by development of new diagnostic techniques. The purpose of this Section is to highlight the areas where the shortfall is likely to be greatest and to outline the work ongoing to address the problems.

7.4.1. Assessment of Measurement Capability

A detailed, but somewhat qualitative, assessment of the status of the measurement capability for each of the main plasma parameters has been carried as a means to identify the most critical areas [7.11]. For each parameter the status of the design of the relevant diagnostic(s) has been assessed and a judgment made as to the probable performance relative to the measurement requirements. The results are combined into a summary table (Table 7-III). The parameters are grouped into three categories: (a) those which appear to be reasonably well diagnosed and for which we expect to meet the target measurement requirements; (b) those which have some difficulties and for which more information is required before it is clear what the final outcome will be; and (c) those for which serious difficulties exist and it is already obvious that it will not be possible to meet the target measurement requirements. Not surprisingly, most parameters are in the middle category at this stage. However, there are some in the first category but there are also some in the third and a few of these are potentially very important. Naturally the current and planned R&D work is concentrating on the areas where the most difficult problems are anticipated. In the table, a number of diagnostic techniques are shown for any one measurement. The assessment of the measurement capability is made on the strength of the best-qualified system. This work is summarized in the following sections.

Table 7-III(a). Parameters which Appear to be Reasonably-Well Diagnosed

Parameter	Purpose	Control Group	Candidate Diagnostics
Plasma current	Current control	1a	Rogowski coils, Magnetic probes
Plasma position and shape	Poloidal field control	1a	Position loops & coils, reflectometry (plasma position)
Loop voltage	l_i, startup	1a	Magnetic flux loops
Line averaged electron density	Plasma operation, disruption avoidance, NBI shine-through	1a	Toroidal interferometer/ polarimeter, LIDAR, visible continuum
m=2 MHD modes	Disruption avoidance	1a	Mirnov loops, ECE, reflectometry
Impurity influx	Impurity control, wall protection	1a	Visible spectrometry, VUV, X-ray spectrometry
ELMs	Optimization control, divertor protection	1a/1b	H_α spectroscopy, ECE, reflectometry, magnetics, IR camera, Langmuir probes
L/H mode indicators	H-mode realization	1a	H_α spectroscopy, reflectometry
Divertor gas press.	Divertor operation	1a	Pressure gauges
Surface wall temperat.	Startup, position & hot spot monitor	1a	Visible & IR TV
Gas pressure (duct)		1a	Pressure gauges
Toroidal magn. field		1a	Current shunts, Rogowski coils
Electron temperature profile	Prof. control with auxiliary heating, confinement studies, plasma energy	1b	ECE, LIDAR
Electron density profile	Transport, fueling optimization, density & beta control	1b	LIDAR, reflectometry, interferometry/polarimetry,
Long term neutron fluence		1b	Activation transfer system
Neutral density (near wall)	L-H transition, particle source	1b	H_α spectroscopy, pressure gauges
Sawteeth	Burn optimization	1b	ECE, neutrons
Edge n_T/n_D, n_H/n_D	Fueling optimization, plasma dilution	2	NPA, H_α spectroscopy

Table 7-III(b). Parameters which Appear to Have Some Difficulties

Parameter	Purpose	Control Group	Candidate Diagnostics
Beta	Disruption avoidance	1a	Diamagnetic loops
Radiated power (core, X-point and divertor)	Disruption avoidance	1a	Bolometer arrays
Total neutron flux	Burn control, fueling control	1a	Neutron flux monitors
Locked modes	Disruption avoidance	1a	Magnetic flux loops, ECE, reflectometry
Z_{eff} (line averaged)	Impurity control, He accumulation	1a	Visible/IR Bremsstrahlung
Runaway electrons	Runaway avoidance	1a	Tang. view X-ray monitor, ECE, synchrotron radiation
Divertor plate temperature	Divertor protection	1a	IR cameras, thermocouples
Divertor radiated power	Divertor operation	1a	Bolometry
In-vessel inspection	View the damage from previous pulse	1a	Viewing camera (with addition of internal illumination source)
'Halo' currents	Monitoring of forces on vessel structure	1a	Magnetic coils on structures
Ion saturation current at divertor plate	Divertor optimization & control	1a	Langmuir probes, tile shunts
Divertor ionization front position	Divertor optimization	1b	Visible spectrometry, bolometry, reflectometry
Neutron emission profile	Burn optimization, fusion power calibration, α-source profile	1b	2D neutron camera
Ion temperature profile	Burn optimization, transport, reaction rate control	1b	X-ray crystal spectr., neutron spectrometer, neutron flux, CXRS
He density in core	Burn optimization & transport, reaction rate control	1b	CXRS with DNB, vis. Bremsstrahlung, fast wave reflectometry
Impurity density profile	Tokamak condition, necessary if high Z; for radiative loss	1b	CXRS, VUV spectrometry, X-ray crystal spectrometry
Zeff profile	Impurity transport	1b	Bremsstrahlung
Impurity & D,T influxes in divertor with spatial resolution	Divertor optimization & control	1b	Impurity monitors (div), H_α spectroscopy (div)
Radiated power profiles in core, edge, divertor	Divertor optimization & control	1b	Bolometer arrays
Heat deposition profile in divertor	Divertor optimization & control	1b	IR camera
Divertor He density	Divertor optimization & control He pumping physics	1b	Impurity monitors, laser induced fluorescence (LIF)
n_e, T_e at divertor plate	Divertor optimization	2	Langmuir probes
Fishbones, TAE modes	Burn optimization, beta limit indication	2	Mirnov loops, reflectometry, ECE
n_T/n_D, n_H/n_D in divertor	Fueling optimization, plasma dilution	2	H_α spectroscopy
n_H/n_D in plasma core	Plasma dilution	2	NPA, fast wave reflectometry

Table 7-III(c). Parameters which Appear to Have Some Serious Difficulties

Parameter	Purpose	Control Group	Candidate Diagnostics
n_T/n_D in core	Fueling control	1a	NPA, fast wave reflectometry
Toroidal & poloidal plasma rotation	Disruption avoidance confinement optimization	1b	CXRS with diagn. NB, X-ray crystal spectrometry
Net divertor plate erosion	Plate erosion	1b	Impurity monitoring, reflectometry
q(r) profile	Performance evaluation, plasma stability, transport, advanced modes	1b	MSE with heating NB, polarimetry
Escaping alphas	Performance evaluation	1b	Lost alpha detectors, IR TV, thermocouple arrays in first wall
n_e, T_e in divertor	Divertor optimization & control	1b	Thomson scattering, Visible/UV spectrosc., reflectometry, ECA,
Confined alphas	Alpha transport & confinement, fusion heating	2	Collective scattering, CXRS/NPA with DNB or pellets, neutrons via knock on tail effect, gamma spectrometer
T_i in divertor	Divertor optimizat. & control	2	Spectroscopy
Plasma flow in divertor	Divertor optimizat. & control	2	Spectroscopy

7.4.2. R&D Program on Materials Inside the Vacuum Vessel

Diagnostic components which are mounted inside tokamak vacuum vessels are conventionally constructed from a wide range of different materials: for example, metals, insulators, and optical materials. In ITER, depending on their exact location, they will be subject to high levels of neutron and gamma radiation and to different thermal environments. Potentially the radiation can change the mechanical, electrical and optical properties and the changes can be either dynamic, that is only during the presence of the radiation, or permanent. The experience on present-day machines is limited because few of them produce high levels of neutron and gamma radiation, although relevant experience has been obtained on TFTR [7.127] and JET. Hence, to provide a data-base on which the design of diagnostic components for ITER can be based, an extensive irradiation effects program has been in place since the beginning of the EDA.

In the program, a wide range of materials have been tested and the effects induced by the neutron and gamma radiation determined. The materials investigated include ceramics and metals which will be used to construct diagnostic components such as magnetic pick-up coils, bolometer heads, cables, optical windows and optical fibers. In ceramics the potentially important effects are radiation induced conductivity (RIC) and radiation induced electrical damage (RIED) [7.128]. RIC is a dynamic effect and RIED is an accumulated effect. The tests have determined the radiation levels at which these effects occur and established guidelines for the use of ceramics on ITER. In summary the results show that it is possible to use ceramics in ITER but they must be selected carefully and their locations, for example behind the blanket shield modules, must be considered in their use. In these locations they should last for the life-time of ITER.

An important component which uses ceramics is insulated cable. The tests have shown that specifically-selected mineral insulated (MI) cable can be used inside ITER in moderately shielded locations and should last for the life-time of the tokamak. The radiation can induce an emf (RIEMF) but in practice differential inputs will be used so this should not cause difficulties.

For optical materials including optical fibers, radiation induced absorption and radioluminescence are important effects [7.127]. The tests have established the levels at which these effects occur and guidelines for the use of optical materials have been determined. In general, optical materials have to be used in well-shielded locations and so it will not be possible to use them close to the plasma. On the other hand, metallic mirrors can survive the radiation close to the plasma but will be subject to nuclear heating and neutral particle bombardment (Section 7.4.3).

Some simplified key results of the tests are summarized in Table 7-IV. In interpreting the results in the Table it should be kept in mind that a first wall load of 1 MW/m^2 is expected in ITER. This corresponds to a total neutron flux of 3×10^{18} $m^{-2}s^{-1}$ (energy >0.1 MeV) and ionization dose rate of 3×10^3 Gy/s. At these levels a total damage of 5 dpa will occur in ceramics and 2-3 dpa in metals assuming a fluence of 0.3 MWa/m^2 at the end of the basic physics phase (BPP). The total ionization dose will be 10^{10} Gy.

The irradiation effects program is expected to continue into the post-EDA phase but to concentrate on the testing of prototype diagnostic components.

Table 7-IV. Results of Irradiation Tests on Candidate Materials for Diagnostic Components

Diagnostic components	**Materials tested**	**Irradiation levels and accumulated effects examined**	**Irradiation levels and dynamic effects examined**
Ceramics (electrical insulators)	Single crystal sapphire and polycrystal alumina (Al_2O_3)	3 dpa in helium atmosphere RIED: no degradation	10^4 Gy/s RIC: $<10^{-6}$ S/m
Wires/cables	MI-cables: SUS, Inconel (sheath)/MgO, Al_2O_3 (insulator)/ Cu, Ni (center lead)	1.8 dpa RIED: no degradation	10^4 Gy/s RIC: $<10^{-6}$ S/m 10^3 Gy/s RIEMF: ≤few volts
Windows	Fused Silica/Quartz	10^{-3} dpa Transmission in range 400-1200 nm: 5% degradation	Radioluminescence: 10^7 photons/Gy.A.sr.cm^3
	Sapphire	0.4 dpa Transmission in range 800-5000 nm: no degradation)	Radioluminescence: 10^{10} photons/Gy.A.sr.cm^3 at λ = 410 nm
Optical fibers	Pure silica (core)/F doped (clad)/Al jacket	Visible region: 10^7 Gy Transmission: 2-2.5 dB/m IR region: 2×10^{-5} dpa Transmission: 10 dB/m	
Mirrors /reflectors	First mirrors: metal (Cu, W, Mo, SS, Al)	40 dpa (Cu) Reflectivity: no degradation	
	Dielectric mirrors: (HfO_2/SiO_2, ZrO_2/SiO_2, TiO_2/SiO_2)	$<10^{-2}$ dpa flaking, blistering of layers	
	Layered synthetic microstructures (LSMs): (Mo/Si, W/B_4C and W/C)	$<10^{-2}$ dpa Shift of the peak reflectivity to shorter wavelength	
	X- ray crystals: Ge, Si, SiO_2, graphite	10^{-2} dpa	

7.4.3 R&D Program on Mirrors

Extensive use of metallic mirrors is foreseen as the plasma viewing device for all optical diagnostics of ITER to ease the problem of radiation damage on optical components. It was first recognized in referencves [7.129, 7.130] that the optical quality of these mirrors could be seriously degraded by the bombardment due to charge exchange neutrals. Mirror surface morphology is expected to change mainly because of sputtering: bombardment by fast neutral erodes predominantly prominent surface structures such as edges, corners or peaks because exposed features receive a larger amount of flux. For polycrystalline materials different erosion rates for different grains are observed, depending on the orientation of the grain to the incident flux. As a result slopes develop at grain boundaries and the surface roughness is increased with a consequent variation of the reflection coefficient which affects the diagnostic calibration. Damage of the mirror could also occur when reflectivity is reduced and the mirror is used to direct high intensity laser beams.

An experimental study of this problem was performed by Bohmeyer et al. [7.131] who simulated the effect of neutrals with the use of the plasma generated by a stationary arc discharge in an axial magnetic field. It was found that a fluence of 1.5×10^{25} m^{-2} of D^+ ions with average energy of 200 eV is sufficient to produce a substantial reduction of reflectivity of standard mirror materials like Al, Cu and Au. The reduction is more severe at short wavelengths and is accompanied by a corresponding increase of diffuse reflectivity. These observations are consistent with increased surface roughness as the underlying mechanism for degradation of optical properties. It was also found that recrystallization of the mirror material with a consequent drastic change of reflectivity can occur when it is not properly cooled and a critical temperature is reached.

When sufficient cooling was provided, a better resistance with respect to surface damaging was found for materials with a low sputtering yield like Rh and Mo. In this case the change in reflectivity was negligible in the visible range up to the fluence of 1.5×10^{25} m^{-2} while in the near UV still a reduction of about a factor 2 was observed.

Projecting these results to ITER requires an estimate of the charge-exchange flux near to the first wall. Since the back flow from the divertor is negligible, it is expected that in ITER, far from the gas injection system, the neutral density in the main chamber will be determined by local recycling. The main uncertainty for the prediction is due to the lack of firm knowledge of SOL parameters and to the influence of plasma-wall distance. A preliminary evaluation leads to a flux of 2×10^{19} m^{-2} s^{-1} which would imply that a Rh or Mo mirror can be used for visible measurements at least for 200 hours of plasma in ITER.

For the divertor region, where the charge-exchange flux is expected to exceed by four orders of magnitude the main chamber value, the mirrors can survive only when the neutral average energy is below the sputtering threshold. This could require the use of shutters or baffles to expose the mirror for only short periods during the discharge [7.132].

7.4.4. Development of Diagnostic Components and New Concept Diagnostic Techniques

As the design of the diagnostic systems become more developed the requirements for specific components become better defined. In many cases these components can be designed and constructed on the basis of experience with similar components on existing large tokamaks. In some cases, however, the ITER requirements are significantly different and new developments are required. The design process has also shown that for some plasma parameters the established techniques cannot be applied to ITER. In these cases new techniques need to be developed. The subject of plasma diagnostics is rapidly developing and fortunately some techniques which potentially can meet some of the ITER requirements are already under development. The development of others needs to be initiated if possible. The R&D program which supports the design process has these two topic areas.

The list of diagnostic components currently under development is varied. Many are developments of “standard” components, necessary to enable operation in the ITER environment. The list includes electrical vacuum feedthroughs, integrators for magnetic diagnostics for very long-time operation, multipin high temperature connectors, radiation-hard bolometers, diagnostic windows and window seals, synthetic and natural diamond detectors for neutron spectrometry, and neutral particle analysis detectors with reduced sensitivity to neutron and gammas. For the post EDA phase, it is planned to continue the development of such components but the construction and testing of key parts of diagnostic systems will also be carried out. For example, a typical first mirror mount with shutter will be built. To make the maximum use possible of the diagnostic ports, several different diagnostics will be installed in each port allocated to diagnostics. The diagnostic components for the different systems will be integrated into a port plug which will also have substantial neutron and gamma shielding. In operation it will be necessary to install the entire port plug with remote handling equipment and to maintain it in the ITER Hot Cell. It is planned to construct a prototype port plug for developing the methods and equipment for maintenance.

A few techniques currently under development can potentially meet the measurement requirements for some of the parameters presently judged as very difficult to measure on ITER. For example, Fast Wave Reflectometry can potentially measure the fueling mix ratio, n_T/n_D, in the plasma core, a parameter which can not be measured with the established technique (NPA) due to insufficient penetration of the neutrals. This technique is currently under development on DIII-D [7.133]. Similarly, Collective Scattering can potentially measure the density, energy distribution and location of the confined alpha particles. The technique is still under development [7.40, 7.41] and it may be able to meet some of the ITER requirements. An improvement in the measurement capability for escaping alpha particles is also required and a device which potentially can be applied to ITER is under development [7.39]. For some parameters, an improvement in the measurement capability is required but no new techniques are presently under development. For example, measurement of the helium density in the plasma core can only be made by CXRS with the DNB. The DNB would be an expensive and large scale facility and, moreover, the measurements can not generally be made in the plasma center because of insufficient penetration of the beam. An alternative method of making this measurement would be beneficial. Similarly, the two established methods for measuring q(r), that is MSE and Multichord Polarimetry, both have difficulty (Section 7.3.7). An alternative method which would be better suited to implementation on ITER would be valuable. Finally, improvements of the measurement capability in the divertor region are required. In particular, improved techniques for measuring the condition of the divertor plates, and for measuring the parameters of the plasma in this region are required. Preferably, the electron density and temperature would be measured in the poloidal cross-section with good spatial resolution. The outstanding measurement requirements and related issues are summarized in Table 7-V.

Table 7-V. Outstanding Measurement Requirements and Related Issues

Requirement	Issue
Development of alternative methods for measuring core n_D/n_T ratio.	Needed for control of fuel mix but existing techniques are not applicable. *Neutron spectrometry:* insufficient s/n due to high background. *NPA:* insufficient penetration
Development of methods for measuring divertor tile erosion in real-time	Needed for protection of divertor tiles.
Determination of life-time of plasma facing mirrors used in optical systems	The first mirror in all optical systems will be subject to the nuclear and gamma radiation, intense VUV, and bombardment by energetic neutrals. The mirrors in the laser systems will in addition have to withstand the high power laser radiation
Determination of life-time of optical elements in the divertor	Divertor tile material redistributed by disruptions could seriously damage optical elements in the divertor.
Development of methods of measuring the energy and density distribution of confined alpha particles	Measurements of confined alpha particles still in infancy. Very limited design base on which to design systems for ITER.
Development of methods of measuring the number and energy of escaping alpha particles.	Existing techniques for measurement of escaping alpha particles do not easily extrapolate to ITER. Very limited design base on which to design systems for ITER.
Devise new concepts for measuring light in-core impurities (e.g. He ash) that do not rely on the diagnostic neutral beam (DNB)	A DNB has high cost and complexity for ITER. There would be considerable benefits if alternative methods for measuring the light impurities could be found.
Devise new concepts for measuring j(r) that can be applied to ITER with sufficient spatial resolution.	Existing techniques either rely on modulating the heating beam (*MSE*) which will be very limited on ITER because of the heavy power requirements and thermal fatigue, or require extensive access (*Polarimetry*)
Determination of impurities in divertor using only visible and uv spectroscopy.	The difficulties of access in the divertor will probably prevent good spatial measurements in the VUV where most power is emitted.
Development of new methods to measure steady state magnetic fields accurately in a nuclear environment.	Several methods exist, e.g. Hall probe and jxB force, but they are not suitable for a nuclear environment or are too elaborate for use in-vessel
Demonstration of direct measurement of local electric field.	E_r is an inferred parameter now. Has high importance in understanding of ITG plasma stability, transport barrier formation.
Investigation of spectroscopic methods for measuring the spatial distributions of n_e and T_e in the divertor.	The modeling requirements for interpreting divertor plasma behavior require accurate spatial information about the n_e and T_e profiles.
Measurement of runaway electrons	Conventional method of measuring ruanaway electrons (*Hard X-ray Emission*) difficult to impelement because of narrow pencil of emission and intense gamma backround.
Measurement of core density profile and MHD activity	Reflectometry using O-mode from the low field side cannot probe the plasma center because of flat profiles and EC absorption.
Measurement of density profile across divertor leg.	Measurement of n_e in divertor required for optimisation of divertor but design base for *Reflectometry* in divertor is very limited.

7.5. SUMMARY

The measurement mission for ITER is very clear but there are some unusual aspects and challenge which have been brought out in this chapter. Diagnostic equipment has been designed with these factors predominantly in mind, because, on the whole, the major system components like detectors, lasers and data-processing electronics will be little different from those presently in use on fusion devices. The environment, and the associated poor access, set some constraints whose effects will only be fully understood once the designs of individual diagnostics are more advanced. The requirements set for the measurement accuracy and resolutions, necessary for the control of the plasmas and for understanding the physics, particularly where steep transport barriers are expected, are very demanding.

Many of the measurements can be made. For some there is some doubt that the proposed diagnostics will be able to meet the requirements. For others, including some key control-related parameters, there is no present technique in which there is full confidence. Hence an extensive R&D program has been put forward and is being supported. The demands for control of the plasmas making use of many different diagnostic signal outputs place harsh demands on the maintenance of calibration and reliability of the diagnostics. These topics are now beginning to be addressed now that the physics issues associated with the diagnostic methods have been understood for ITER.

Any discussion on measurement capability for a device written many years before it starts up will become out-dated as new measurement techniques are developed. New operational regimes and the need to know some different plasma parameter are always being found and the ITER diagnostic set will continuously evolve into its operational life. The groundwork is being laid now, but to meet the physics program goals much more work is required.

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