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MEASUREMENT REQUIREMENTS FOR THE ADVANCED TOKAMAK OPERATION OF A BURNING PLASMA EXPERIMENT (BPX)

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ABSTRACT

The optimization of the tokamak toward steady state and high performance has been the focus of Advanced Tokamak (AT) research for the past decade. A central theme of AT research line is plasma control: control of the plasma shape; of the profiles of current, pressure, and rotation; of transport; and of MHD stability. To optimize the performance, measurements of crucial parameters such as the current density and the plasma pressure, are required with appropriate spatial coverage and resolution. In addition, measurements of other parameters will be necessary to develop fundamental understanding of the complex nonlinear interactions amongst the current density profile, the pressure profile and transport (e.g., turbulence) in high beta AT plasmas. Present day experiments are providing physics insight into what a Burning Plasma Experiment (BPX) will require as measurements. Recent research has focused on MHD stability aspects such as the neoclassical tearing mode (NTM) and resistive wall mode (RWM) stabilization and control of the current profile. However, in burning plasmas, new factors such as alpha particles, with their heating contribution and their relationship to transport barriers, will be increasingly important. The close relationship between measurements and active control, and the resultant impact on the requirements will be discussed.

I. INTRODUCTION

The progress in the development of long pulse high performance Advanced Tokamak (AT) discharges has been significant in the last decade. With this rapid progress it is increasingly likely that a Burning Plasma Experiment (BPX) will utilize AT discharges in a large fraction of its program for studying the science of burning plasmas and defining an attractive reactor option. Consequently, it is important that we review and identify the measurement requirements for advanced operation scenarios. Previous discussions of measurement requirements covered the needs for specific diagnostics or class of measurements [1] for AT operation, or reviewed the needs in other operation scenarios [2], with the exception of the AT scenario. A review of these AT scenario needs is also becoming imperative, as the design of these experiments is approaching the construction phase. This process may also impact the schedule for installation and commissioning of diagnostics, since profile-measuring diagnostics, for example, will be required for both physics evaluation and control, even in the initial hydrogen phase of the experiment.

One, perhaps unexpected, consequence of the recent success obtained in operating tokamaks is that a very broad family of AT discharges has been obtained, which could ultimately meet the goals of a BPX. These various AT scenarios, or schemes, utilize different feedback tools, actuators as well as sensors. This variety contrasts visibly with the standard ELMy H–mode reference case. One important but often overlooked complication arises from the frequent needs to use reverse toroidal field and/or reversed plasma current direction which has some technical implications for diagnostics. The level of flexibility required for the diagnostic set in an AT scenario is challenging, as can be seen in the measurement requirements. This flexibility has to be achieved within the significant constraints encountered by diagnostics in a BPX environment, with its intense radiation field, difficult access and/or long pulse, just to name a few. In spite of these difficulties, the assembled diagnostic set would be one of the most comprehensive arrays of measurements ever achieved on a tokamak.

For AT operational scenarios, special attention should be given to β limits. The main focus of AT research is to maximize fusion output, in long pulse discharges while minimizing the auxiliary control power, and by relying on high bootstrap fraction. Fusion power and bootstrap fraction both increase the emphasis on optimizing the plasma pressure (β_N), which would be increasingly affected by the presence of alpha particles in a self-organized way. Detection and control of the Neoclassical Tearing and Resistive Wall modes are key in optimizing β and have received much attention in the last few years. The measurement requirements associated with these modes are relatively well understood. However, the scientific investigation and plasma optimization with edge transport barriers (ETBs) and/or internal transport barriers (ITBs) will require additional measurement capability. Scientific investigation of and mitigation of edge localized modes (ELMs) associated with ETB in H–mode plasmas might also lead to specific diagnostic requirements. Furthermore, the results obtained so far, and the goal of steady-state operation indicate a clear need for understanding and controlling momentum and particle control transport, which in turn will create additional measurement requirements.

This paper aims at stimulating discussions in the area of diagnostics and their requirements for a BPX, in the specific area of an Advanced Tokamak scenario of operation. This process remains a work in progress, constantly refining these needs. In addition, the justifications for these requirements are not fully established, and the full details are presently beyond the scope of this contribution to the proceedings. Following a discussion of global issues in measurements and control aspects, we review in section 4 some of the specific quantitative requirements.

II. CRITICAL MEASUREMENTS

While many measurements are normally required to establish, control and understand a given discharge, a few key measurements are required in order to achieve the highest β in an AT plasma. The beta limits are known [3] to depend strongly on plasma shape and position, current, pressure and plasma rotation (primarily for wall stabilization) profiles [5].

Although the β-limit is not a directly measured quantity, it will be very important that we know it rather accurately in order to optimize a given discharge. No simple algebraic expression is presently available to express the beta limit for a variety of conditions, so we must presently rely on empirical results or possibly lengthy computer calculations. A control system will need to establish that limit on a real-time basis. In this case, the measurements required to evaluate the limit (such as shape, current, rotation and pressure profiles) must be known accurately as well. The uncertainty in this quantity will represent the minimum "safe distance" below which the plasma beta must be kept. This restriction in the ranges of beta allowed may or may not be acceptable. A possibly more direct approach would be to use a proxy, a measurement that would be a good representation of how close the discharge is from a known limit. For example, we could consider using a series of internal coils for MHD spectroscopy, as found on JET [4] and DIII-D [5] for example, in which a probing perturbation can indicate proximity to a growing instability. Such approach would be best used when dealing with NTMs, RWMs, TAEs (Toroidal Alfvén Eigenmode) and ELMs. However, in these cases the probing coils or antennas need to be installed within the vacuum vessel, as close to the plasma as possible, in order to possess the time response and sensitivity required for the mode to be studied and excited.

Another class of measurements that will be especially important relates to our capability to resolve the various gradients at both edge and core transport barriers. Clearly, it will be necessary to measure temperature and density gradients sufficiently well for their characterization, whereas a measurement of E_r and possibly impurity or radiation profiles would be also required for full control. One presently used empirical technique to reduce impurity accumulation is to make use of ECH, ICRF or LH waves within the barrier[6], although the "cleaning" mechanism is not fully understood. A different approach may be to control the gradients at the edge, and its effective impurity pinch mechanism [7]. The measurement of these gradients will also be needed in order to avoid additional MHD instability such as the resistive interchange mode (for ITBs) and ELMs (for ETBs). The gradients and gradient lengths for an ITB in a BPX are not yet sufficiently well predicted by present models. CORSICA [8] simulations were performed to study an ITB regime in ITER, and consider the requirements in spatial resolution for both ion and electron temperature profiles. The electron and ion temperature profiles are evolved using a model for thermal diffusivities, $\chi_{e,i}$, formed from a combination of the GLF23, neoclassical and Coppi-Tang L–mode (CT). Near the magnetic axis, GLF23 produces thermal diffusivities that can drop below neoclassical values and a minimum thermal diffusivity is imposed. In the edge region, the current GLF23 model does not apply and the CT-model is used. For these internal transport barrier (ITB) simulations, the full CT-model scaling is used at the edge (outside $\rho = 0.85$; ρ=square root of the toroidal flux) to give an L–mode-like edge. An internal transport barrier is formed at 125 s, with a weak negative central shear with $I_p = 12$ MA and 6 MW of auxiliary heating. In these simulations, the required accuracy of 10% and a spatial resolution of $a/30$ were included. As shown in Fig. 1, we can see that we can reasonably resolve the internal barrier region. These results do not verify the proposed resolution requirements as the simulations are not fully self-consistent, yet, and have not been tested against a series of models for transport barriers. Nevertheless, they appear to be consistent with present-day device's results as reported in reference [7], and consequently appear to be a reasonable target with the present understanding.

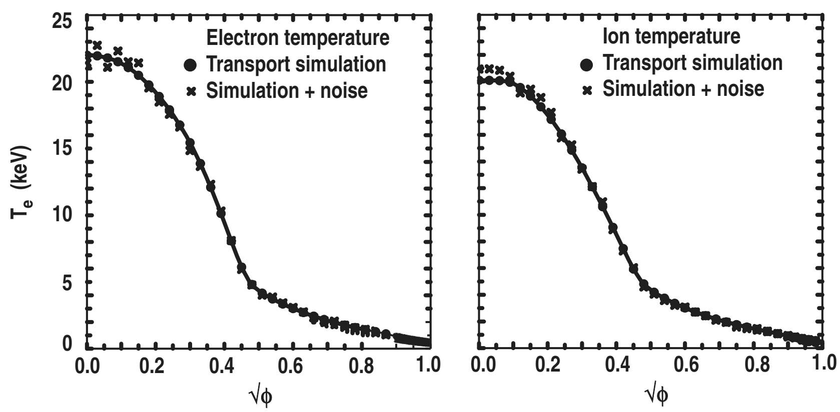

Fig. 1. CORSICA simulations of ion and electron temperature profiles for ITER. X points correspond to simulations including uncertainty in measurement. $\sqrt{\phi}$ is the radial coordinate.

One of the most exciting and important physics questions that would be addressed in a BPX regards the role of alpha particles. Compared to previous experiments which involved DT operation, it is expected that the alpha particles in a BPX will modify the plasma dynamics. Such effects in an AT discharge will be important, and a proper diagnosis of the confined alpha population (energy, time and spatially resolved) is required. The study of their effects on the stability of the plasma and transport will be crucial to the experiment. By the same token, the presence of 1 MeV deuterons (from the heating beam, in case of ITER) will also affect the dynamics and transport of the discharge and require also diagnosis, preferably dissociated from the measurements done on the alphas. As shown in Table 1, the time resolution should be comparable to the slowing down time, which would also be smaller than the particle confinement time. However, the requirements in energy resolution are not clear, and a detailed study of requirements versus stability analysis (such as TAEs) is needed. Clearly, in the presence of important feedback circuits (heating, momentum and particle control), one should be able to properly identify the driving terms and consequently use the proper actuator (i.e. fusion versus neutral beam sources).

III. CONTROL ISSUES

Even in existing devices, an increasing number of measurement signals are used in the plasma control system, and feedback loops. Accurate and reliable measurements for plasma control are especially important for the optimization of an AT discharge. One subtle but crucial aspect of the control system requires that sensors (diagnostics) remain largely independent of the actuators [e.g., electron cyclotron heating (ECH) or neutral beams]. It is clearly increasingly difficult and complex to modify the plasma behavior when the measurements are no longer independent of the actuators. Consequently, independent sources for the diagnostics, e.g., diagnostic neutral beams or microwave power, will be essential.

The present understanding of the physics and the extrapolation to a burning plasma experiment significantly impacts the anticipated measurement requirements. One example includes the mitigation of NTMs. We must be able to detect them early enough to limit their effects on the plasma and thus minimize the power requirements from electron cyclotron current drive (ECCD), for example. A clear analysis is needed to close the "loop": the amount of power available will dictate the width and amplitude that can be mitigated, which in turn dictates the sensitivity of the diagnostics required for their detection. It is presently projected that the threshold width (NTM) could be as small as $0.03\,a$ (a being the minor radius) [10]. Electron density and temperature measurements would marginally meet this requirement (resolution of $a/30$). However, it is not presently known if this seed island could be detected by its magnetic signature alone. The present design of the magnetics diagnostic, its expected signal to noise ratio, which for example, is expected to be affected by the radiation environment[11], are not sufficiently known to guarantee that a measurement can be done at that level. Similarly, it is expected that the detectable amplitude of RWMs will be quite low, and can be dependent on the stabilization mechanism, being rotation or direct stabilization by coils mounted in close proximity to the plasma [5,11]. The response time of the control system (e.g., internal and external coils) will dictate the time response of the measurements and/or its sensitivity, therefore requiring a complete analysis of the feedback circuit, including the sensors.

IV. SPECIFIC REQUIREMENTS

The specific requirements that are directly related to an AT scenario are shown in Table 1. This list is based on FIRE characteristics, although only minor differences would be found for ITER. For clarity, this table focuses on very specific needs for AT operation scenario, while obviously many other measurements are needed for operation of the tokamak but not necessarily for an AT. The complete list of measurement requirements can be found in Ref. [14]. Shown in bold face are the requirements for which the AT scenario necessitated a revision. Overall the need to control the shape and position of the plasma will require a precision of 2mm for the equilibrium reconstruction of the separatrix around the discharge. This, in turn, requires that high resolution diagnostic systems, such as the Thomson scattering and the Charge-Exchange Recombination Spectroscopy system planned for the pedestal region, have sufficient spatial coverage to include the steep (e.g., gradient) regions. In order to support the study and control of AT discharges, which include a variety of shapes (elongation and triangularity), and retain the capability of the diagnostic, it is important to extend the range in minor radius to $0.8 < r/a < 1.1$. (Note that in Table 1, a is defined as the radius of the first wall at the midplane, an engineering definition applicable to locating sight lines, etc. For each diagnostic, expert evaluation for establishing these sight lines will be necessary, depending on its location on the tokamak, during its final design and implementation phases.) In addition, this precision in flux mapping (magnetic structure) will ensure that measurements done at various poloidal locations can be compared on the same basis.

Table I. FIRE Measurement Specification (Fourth Revision 9/4/03). Shown in bold face are the entries that necessitated a revision for the AT operation scenario.

MEASUREMENT	PARAMETER	CONDITION	RANGE or COVERAGE	T or F	X or k	ACCURACY
1. Plasma Position and Shape	**Reconstruction Accuracy, Δ_{sep}**	$I_p >$ 0.2 MA, full bore	-	10 ms	-	**2 mm**
		I_p Quench	-	1 ms	-	5 mm
	Divertor channel location (r dir.)	Default	-	10 ms	-	2 mm
		I_p Quench	-	1 ms	-	5 mm
	dZ/dt of current centroid	Default	0 – 100 m s^{-1}	0.1 ms	-	3 %
2. Error Field, Locked Mode and RWM Identification	$\Delta B_p/B_p$, $\Delta B_r/B_T$		10^{-5} – 10^{-2}	100 ms	TBD	30 %
	$\Delta B_r/B_p$		10^{-4} – 10^{-2}	1 ms	(m,n) = (2,1)	30 %
	$\Delta B_p/B_p$		10^{-4} – 10^{-2}	1 ms	TBD	30 %
3. Low (m,n) MHD Modes, Sawteeth, Disruption Precursors	Mode complex amplitude at wall		TBD	DC – 10 kHz	(0,0) < (m,n) < (10,2)	10 %
	Mode - induced temperature fluctuation		TBD	DC – 10 kHz	(0,0) < (m,n) < (10,2) Δr = a /30	10 %
	Other mode parameters		TBD	DC – 30 kHz	Integral	10 %
4. Neutron Flux and Emissivity	Neutron / α source		$1 \cdot 10^{14} – 5 \cdot 10^{18} nm^{-3}\ s^{-1}$	1 ms	**a/30@ center, wider @ edge**	10 %
	Fusion power density		0.1 - 20 MW m^{-3}	1 ms	**a/30@ center, wider @ edge**	10 %
5. Plasma Energy	βp; compare to β - limits	Default	.01 – 1	1 ms	Integral	5 % @ β_p =1
		Thermal Quench	.01 – 1	0.1 ms	Integral	~ 30 %
6. Electron Temperature Profile	Core T_e	**r/a < 0.8**	0.5 – 15 keV	10 ms	a/30	10 %
	Edge T_e	**r/a > 0.8**	0.05 – 5 keV	.01 ms	5 mm	10 %
7. Electron Density Profile	Core N_e	**r/a < 0.8**	$3 \cdot 10^{19} – 1 \cdot 10^{21} m^{-3}$	10 ms	a/30	5 %
	Edge N_e	**r/a > 0.8**	$5 \cdot 10^{18} – 2 \cdot 10^{20} m^{-3}$	.01 ms	5 mm	5 %
8. Ion Temperature Profile	Core T_i	**r/a < 0.8**	0.5 – 15 keV	10 ms	a/30	10 %
	Edge T_i	**r/a > 0.8**	0.05 – 5 keV	10 ms	5 mm	10 %
9. Current Density Profile	q(r)	**r/a < 0.8**	**0.5 - 5**	10 ms	a/30	10 %
			5 – TBD	10 ms	a/30	0.5
		r/a > 0.8	**2 - 5**	10 ms	5 mm	10 %
			5 – TBD	10 ms	5 mm	0.5
	r(q=1.5,2)/a	NTM feedback	0.3 – 0.7	10 ms		**20 mm**
	r(q_{min})/a	Reverse shear control	0.3 – 0.7	10 ms		**20 mm**
10. Plasma Rotation Profile	V_{TOR}		1 – 100 km s^{-1}	10 ms	**a/30**	**10 %**
	V_{POL}		1 – 50 km s^{-1}	10 ms	**a/30**	**10 %**
11. Radial Electric Field Profile	E_r (r ,t)	**r/a < 0.8**	**5 - 100 kV m^{-1}**	10 ms	**a/30**	TBD
		r/a > 0.8	**5 - 100 kV m^{-1}**	10 ms	**5 mm**	TBD
12. High frequency instabilities (MHD NTMs, AEs, turbulence)	MHD, NTMs			10 –100 kHz	10 mm	-
	AE Mode – induced perturbations in B,T,n		n = 10 - 50	10 –300 kHz	10 mm	-
	High frequency turbulence	Correlation	-	10 –300 kHz	10 mm	-
13. Edge Turbulence	TBD	r/a > 0.8	TBD	< 200 kHz	5 mm	
14. Radiation Profile	Main Plasma P_{RAD}		0.01 – 1 MW m^{-3}	10 ms	a/15	20 %
15. Confined Fast Ions	**Energy Spectrum**	Energy resolution TBD	0.1 – 3 MeV	100 ms	a/10	20 %
	Density Profile		$(0.1 – 4) \cdot 10^{18}\ m^{-3}$	100 ms	a/10	20 %
16. Confined Alphas	Energy Spectrum	Energy resolution TBD	0.1 – 3.5 MeV	100 ms	a/10	20 %
	Density Profile		$(0.1 – 4) \cdot 10^{18}\ m^{-3}$	100 ms	a/10	20 %

V. SUMMARY

The measurement requirements for a BPX are focused on a few important parameters, related mostly to shape, stability and profile information. While many of these requirements were already sufficiently well defined in respect to other operating scenarios, we identified a few important differences. Flexibility in measurement coverage and resolution, especially at the edge, is noteworthy. In addition, increased needs in rotation accuracy, q profile measurements and the need in diagnosing fast ions (such as beam ions) are required for stability control of the discharge. However, these additional requirements are nonetheless relatively few, and increase confidence that such scenario could be studied and used on a BPX.

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