OPERATION AND CONTROL OF ITER PLASMAS

ITER JOINT CENTRAL TEAM, ITER HOME TEAMS AND ITER DISRUPTION, PLASMA CONTROL AND MHD EXPERT GROUP

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Abstract

Features incorporated in the design of the International Thermonuclear Experimental Reactor (ITER) tokamak and its ancillary and plasma diagnostic systems that will facilitate operation and control of ignited and/or high-Q DT plasmas are presented. Control methods based upon straight-forward extrapolation of techniques employed in the present generation of tokamaks are found to be adequate and effective for DT plasma control with burn durations of ≥1000 s. Examples of simulations of key plasma control functions including magnetic configuration control and fusion burn (power) control are given. The prospects for the creation and control of steady-state plasmas sustained by non-inductive current drive are also discussed.

1. INTRODUCTION

The design of the International Thermonuclear Experimental Reactor (ITER) developed during the course of the Engineering Design Activities (EDA) [1, 2] incorporates all of the provisions needed for the reliable operation and control of ignited and/or high-Q driven-burn DT plasmas with fusion powers in the 1-1.5 GW range and fusion burn durations of ≥1000 s. The nominal size (R = 8.14 m, a = 2.80 m, κ_{95} = 1.6), on-axis toroidal field strength (5.7 T) and nominal plasma current (21 MA) of the ITER tokamak are chosen such that with presently-foreseeable physics basis assumptions about attainable energy confinement, attainable plasma density and projected plasma impurity content (including thermalized alpha particles), sustained D-T burn with power ≥ 1 GW is possible [3]. Auxiliary heating and/or current drive powers of up to 100 MW are provided for the initiation of ignited burn and for the sustainment of high-Q (Q ≥10) driven burn. The in-vessel plasma facing surfaces and nuclear shielding modules are designed with steady-state power handling capabilities such that sustained fusion powers of up to 1.5 GW with 10-s duration transients of up to 1.8 GW can be accommodated. The Poloidal Field (PF) coil system is sized and configured such that static and dynamic plasma equilibrium control at plasma currents of up to 24 MA are possible, and the PF system supplies sufficient inductive current drive to enable nominal 21-MA, 1600-s duration pulses (including a 1000-s burn) to be produced. Shorter duration (500-s burn) inductively-sustained pulses at 24 MA are also possible. Table 1 summarizes key plasma-performance-related parameters. The emphasis here is to show the as-designed capability of ITER to conduct DT-burning plasma experiments and operation over a finite range of plasma operational conditions and hence over a credible range of foreseeable plasma performance outcomes. The availability of a finite operation capability domain is a necessary prerequisite of being able to then control the plasma performance within this domain. The range of plasma operation capability provided in ITER coupled with the availability of a comprehensive suite of plasma diagnostics will also facilitate the conduct and control of first-of-kind scientific and performance optimization experiments with GW-power-level DT-burning plasmas.

The specific means provided for the control of ITER plasmas are presented below. The presentation is organized into four topical Sections: Section 2, plasma operation scenario; Section 3, plasma magnetics control; Section 4 plasma kinetics control; and Section 5, prospects for the production and control of steady-state-capable plasmas with modified current profiles and internal transport barriers. The overall prognosis for ITER plasma operation and control is summarized in Section 6.

2. PLASMA OPERATION SCENARIO AND CONTROL CONCEPTS

Plasma operation in ITER will be conducted within the framework of an inductively-driven and controlled plasma operation scenario [1],[4],[5]. The scenario concept is identical to that employed in the present generation of shaped-cross-section divertor tokamaks. Figures 1 and 2 illustrate the scenario concept and show the plasma current/shape/configuration evolution that the scenario will incorporate.

TABLE I. ITER TOKAMAK AND PLASMA OPERATION CAPABILITIES

Parameter or Capability	Symbol	Units	Nominal	Expected Range[a]
Plasma major radius	R_o	m	8.14	8.1-8.5[b]
Plasma minor radius	a	m	2.80	2.8–2.4[b]
Vertical elongation (95% flux)	κ_{95}	–	1.63	1.55-1.70
				1.6–2.0[b]
Toroidal field (at R = 8.1 m)	B	T	5.7	4.0–5.7[c]
Safety factor (95% flux)	q_{95}	–	3.0	2.4-5
Plasma current	I_p	MA	21	12-24
Fusion power	P_{fus}	GW	1.5	1.0–1.8[d]
Auxiliary power	P_{aux}	MW	100	0-100
Fusion power gain (P_{fus}/P_{aux})	Q	–	∞	$5-\infty$

[a] Applies for 'full-performance' 1-1.5 GW DT plasma operation. Additional range may be available or applicable for low-power plasma operation.
[b] Plasma R, a, κ_{95} variations are correlated: R_o + a and divertor strikepoints remain approximately fixed.
[c] Toroidal field magnet range is 0-5.7 T; 4-T limit is set by ECH plasma start-up assist frequency, also possibly by rf heating/CD means and source frequency range(s).
[d] Nominal sustained maximum fusion power is 1.5 GW; up to 1.8 GW transient for 10 s is allowed.

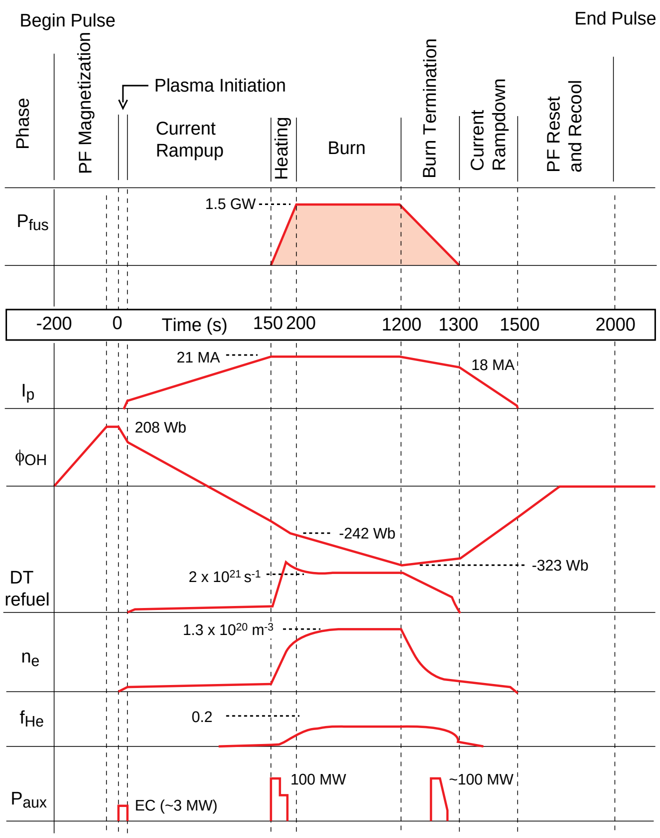

FIG. 1. PF and plasma parameter waveforms for the nominal 21-MA plasma operation scenario.

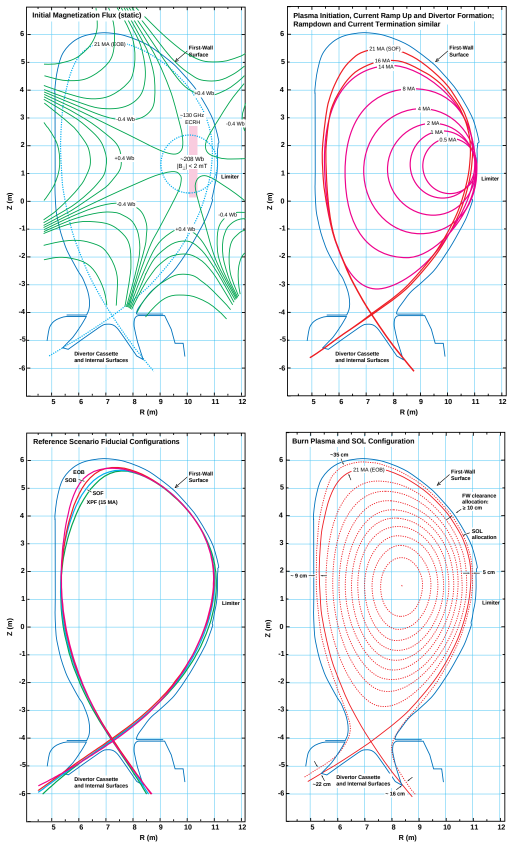

FIG. 2. Plasma equilibrium evolution and features for 21-MA plasma scenario.

Features of the scenario include *i)* a 530 Wb PF system flux swing, *ii)* inductive plasma initiation (Townsend avalanche breakdown with EC assist) in a high-order multipole field null positioned near the outboard port-mounted limiter, *iii)* minor radius and elongation expansion of the startup plasma on the limiter prior to divertor formation at I_p ~15 MA, and *iv*) maintenance of a well-controlled single-null divertor configuration during the heating/burn/burn-termination phases. Current termination is effected following burn termination with a minor radius and elongation contraction on the limiter.

Simulations of the plasma startup and shutdown dynamics show that MHD stability (trajectory in the q-l_i domain) and the edge plasma power balance needed to avoid a density-limit disruption are satisfied with acceptable margins. The resistive flux (volt-second) consumption during the startup and current rampup phase falls within the physics design basis of $0.45\mu_0 R_o I_p$ ($\cong$ 100 Wb) and $\geq$ 80 Wb of PF system flux swing will be available for sustaining the 21-MA plasma current during fusion burn. For the estimated burn-phase plasma resistive voltage, this flux swing will provide a 1300-s burn.

The scenario design basis is predicated upon a ‘reference case’ plasma with I_p = 21 MA, poloidal beta β_p = 0.9 and dimensionless internal inductance $l_i(3)$ = 0.9. The sizing of the PF coils and power supplies allows equilibrium control and in most cases ≥1000-s inductively-sustained burn to be obtained for 21-MA plasmas with $0.7 \leq \beta_p \leq 1.2$ and $0.7 \leq l_i \leq 1.1$. Operation at 24 MA ($q_{95} \cong 2.6$) with $\beta_p \cong 0.8$ and $l_i \cong 0.8$ (1.5 GW fusion power) is also feasible. Inductively-sustained burn at 24 MA is about 500 s. The scenario will also support operation with ohmic and auxiliary-heated DD plasmas during initial plasma commissioning, and extended-pulse inductively sustained driven-burn operation with reduced plasma current (e.g., 6000 s burn at ~1 GW with I_p = 17 MA and 100 MW H/CD power). True steady-state operation may also be possible with a reversed-shear plasma at I_p ~12 MA. Possibilities and open physics and plasma control issues for such operation are discussed in Section 5.

Control of the overall ITER plasma operation scenario and control of the plasma magnetic and kinetic attributes during the various phases of the scenario will be effected by a Plasma Control System (PCS) that will be a subordinate control system embedded within the overall ITER facility-wide Control and Data Acquisition (CODAC) System [3]. The precise hardware details and configuration of the PCS remain to be defined, but it is certain that the PCS will, like the control systems of the present generation of large tokamaks, be implemented primarily by computer software and will configured in a modular and upgradeable fashion, with independent tokamak and plasma sequencing (scenario), plasma magnetics (plasma current, shape and divertor configuration) and plasma kinetics (density, fusion power and divertor power) control algorithm modules [4]. The PCS will also control the means (e.g., impurity gas or pellet injection) for effecting a fast fusion power and plasma current shutdown in circumstances where plasma operation anomalies develop and/or where at-risk tokamak components (e.g., the divertor targets) require prompt plasma shutdown for protection. These capabilities will provide a plasma-operation-related ‘machine protection’ function that will be integrated with plasma operation and control.

The PCS will also incorporate ‘plasma operation regime’ identification and reaction capabilities. The regime-identification capability, which will use plasma diagnostic data to ascertain the plasma operation regime or state (ohmic, L-mode, H-mode, full-attached or partially-attached divertor, etc.) and energy content, will be used to enable (activate) plasma control algorithms that are optimized for the corresponding plasma operation conditions. The PCS will also incorporated event-triggered plasma operation abort and machine component protection capabilities. These capabilities will facilitate more-robust and reliable control the sequence of plasma states and control procedures that will be required to effect an ITER fusion burn pulse of specified power and duration and to take appropriate action if plasma or machine operation anomalies occur. Additional details of the PCS regime-identification and event-reaction capabilities are described below.

Plasma control systems that include plasma-regime-identification and event-triggered plasma operation abort and machine component protection logic are presently being implemented in a number of tokamaks, and there have been significant successes with such systems in providing more reliable control of highly-optimized plasma operation regimes (e.g., the high-radiation H-mode in ASDEX-Upgrade [5,6]) and in minimizing the adverse effects of disruptions in vertically-elongated plasmas in JET [7]. In these ‘modern’ plasma control systems, the traditional separation between tokamak system operation and protection and plasma operation and control is becoming increasingly blurred (in part owing to the increasing levels of plasma magnetic and kinetic energy that modern tokamaks can obtain), and we anticipate that by the time that ITER operation commences that a comprehensive physics and plasma control engineering experience basis for the implementation of such ‘integrated’ and ‘intelligent/adaptive’ tokamak and plasma control systems will be available.

III. PLASMA MAGNETICS CONTROL

Magnetic control comprises quasi-static and dynamic control of the global parameters of the tokamak plasma equilibrium — current, shape and position — by magnetic means, effected through control of the currents in the array of Poloidal Field (PF) coils that modern tokamaks incorporate. For ITER, there are 9 independently-controlled PF coils (PF1-PF9) plus a separate central solenoid (CS), all located outboard of the Toroidal Field (TF) coil [1],[3]. Figure 3 illustrates the plasma and tokamak cross-sections of ITER and ASDEX-Upgrade, which has a plasma and PF geometry that is very similar to ITER [5]. Figure 3 shows the use in each experiment of fiducial plasma boundary position control points (g_1-g_6 for ITER) to specify and control the plasma shape and position within the first-wall (FW) plasma-facing-surfaces and the divertor. A similar ‘FW gap + strikepoints’ control strategy has recently been implemented in ASDEX-Upgrade [8]. The control points used are shown in Fig. 3.

The underlying physical basis of plasma current, shape, and position control in tokamaks is ideal MHD equilibrium, as embodied in the Grad-Shafranov plasma equilibrium equation. For ITER, validated Grad-Shafranov equilibrium models have been applied for assessments of both ‘slow’ quasi-static equilibrium control during the course of the plasma operation scenario (e.g., Fig. 2.) and ‘fast’ dynamic control of the plasma configuration in response to ‘plasma disturbances’ that include sawteeth, minor disruptions and Type I ELMs. Table II summaries the disturbances specified for ITER control system evaluation purposes. The most serious disturbance is a minor disruption during the burn phase, with 20% beta loss and 0.1 l_i drop. The plasma shape and current response to such a disturbance (plus the other disturbances given in Table II) has been used to evaluate available PF control power and assess whether adequate plasma shape control (avoidance of appreciable separatrix-to-FW contact) will occur.

TABLE II. PLASMA DISTURBANCES FOR ITER CONTROL SYSTEM EVALUATION

Disturbance	Δl_i (decrease)	$\Delta\beta_p$ (decrease)	Waveform/frequency/recurrence
Vertical drift	0	0	0.1-m ‘control-off’ drift phase
Minor disruption (A)	0	≤ 0.2 (decrease)	Step/0.2 Hz/< 10 per pulse
Minor disruption (B)	≤ 0.1	≤ 0.2 (decrease)	Step/0.2 Hz/< 10 per pulse
Sawtooth	≤ 0.05	≤ 0.05 (decrease)	Sawtooth: 0.1-0.01 Hz
ELM	≤ 0.05	≤ 0.05 (decrease)	Sawtooth: 2-0.2 Hz

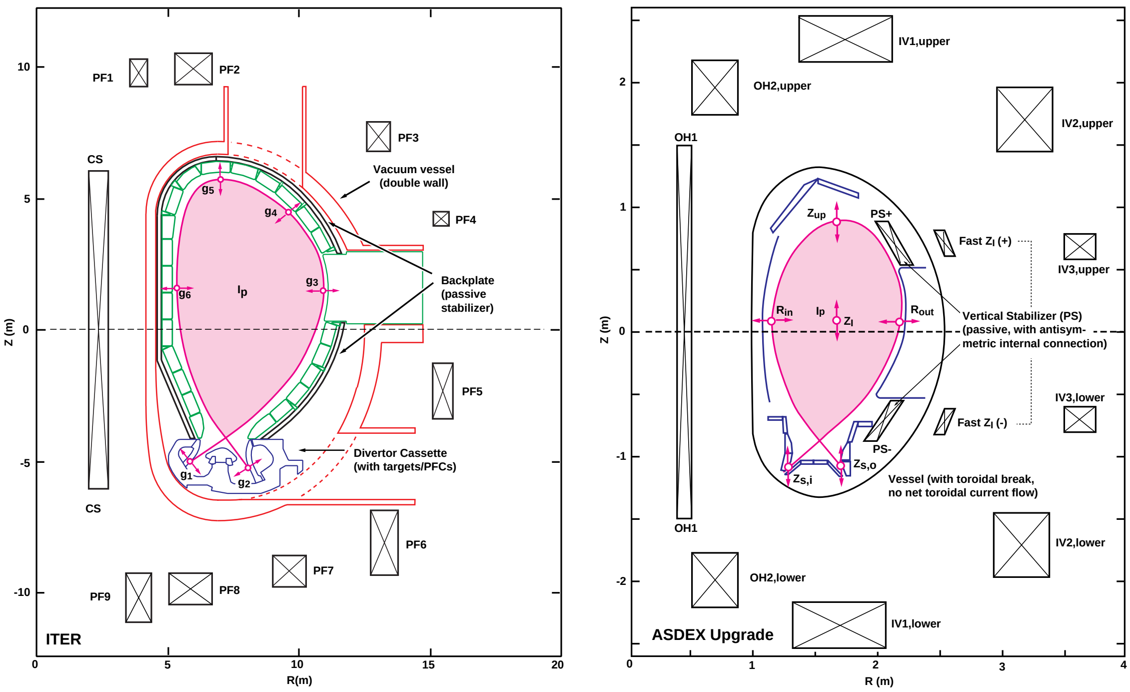

FIG. 3. Comparison of the plasma and PF coil configurations of ITER and ASDEX-Upgrade. The plasma configuration boundary control points for the two experiments are indicated. The bi-directional arrows show the magnetic control action for the corresponding point.

A simulation using a Tokamak Simulation Code (TSC) model of response to a minor disruption (B) disturbance is shown in Fig. 4. The gap data show that the immediate response of the separatrix is to shrink away from the first wall. The maximum excursion of the gap between the separatrix and the first wall or the position of the divertor leg strike point is 17 cm. The power required to bring the plasma back to the nominal separatrix location is ~80 MW, well within the ITER line-power demand and rate-of-demand-variation specifications. There is no plasma (separatrix) contact with the wall. The rf antenna-separatrix separation (gap g_3) does not decrease by more than 5 cm. All nominal ITER PF/plasma control requirements are met. Similarly favorable results are obtained for the other disturbances given in Table II.

The control algorithm used for the simulation shown in Fig. 4 is derived using the 'H-infinity' control design methodology applied to a linearized Grad-Shafranov model (the CREATE-L model). The H-infinity method allows the internal parameters of the controller elements (gain, frequency response, and coupling of the feedback loops that control the PF power supplies) to be optimized in a mathematically defined manner. Limitation of the plasma control gap variances and limitation of the peak power demand and rate-of-demand enter into the controller optimization procedure. The plasma response simulation with a full non-linear TSC model confirms both the validity of the design optimization procedure and the ability of this type of power-demand and gap-variance-optimized controller and PF design to meet ITER axisymmetric (n = 0) magnetic control requirements.

The ITER magnet system and the ITER PCS will incorporate provisions for the avoidance of MHD instabilities caused by non-axisymmetric (n > 0) field errors that arise from small deviations of the ITER TF and PF coils from ideal symmetry. Here small ($B_{error}/B \cong 10^{-4}$) levels of low m,n non-axisymmetric error fields are projected to result in plasma mode rotation locking and subsequent disruption in low-density startup plasmas ($n_e \sim 2 \times 10^{19}$ m^{-3}) and possibly also in high-density high-beta burn phase plasmas [1]. The ITER magnet system will include three n = 1 error-field correction coils capable of reducing the rms-weighted sum of the 1,1, 2,1 and 3,1 error fields to levels ~2 x 10^{-5}. The magnitude and toroidal phase of the currents in the three correction coils will be controlled —in response to changes in the PF coil currents—to maintain an acceptable level of error field throughout the plasma operation scenario.

3. PLASMA KINETICS CONTROL

Kinetics control comprises the establishment and sustainment of the kinetic attributes of the core and divertor plasma regions of the tokamak discharge. At the most elementary level, the key kinetic attributes of the core are density, temperature, impurity content and, for a DT plasma, fusion power. At the same elementary level, the key attributes of the divertor plasma are temperature, density, impurity content and ionization state and the resulting levels of radiated power and power conducted to the divertor targets. In ITER, an integrated DT-fueling/impurity-injection/auxiliary-heating plasma kinetics control system that can simultaneously control the fusion power level and limit the amount of power to be exhausted by the divertor system to acceptable levels (~ 50 MW) will be provided [1]. Figure 5 illustrates how the kinetics control system maintains control of fusion and divertor power during a severe plasma power balance transient (100 MW power added). Fusion and divertor target power control during power ramp up and rampdown are also maintained (high-frequency power transients are likely modelling artifacts and are within the transient capabilities of the affected components).

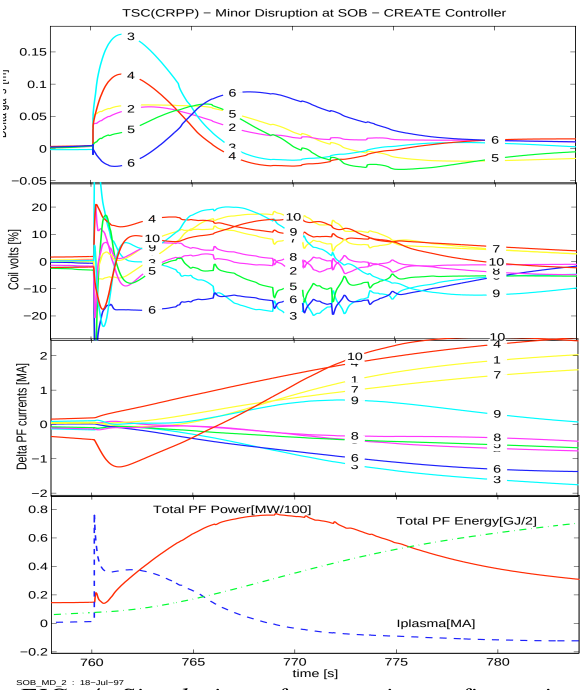

FIG. 4. Simulation of magnetic configuration control following a minor disruption.

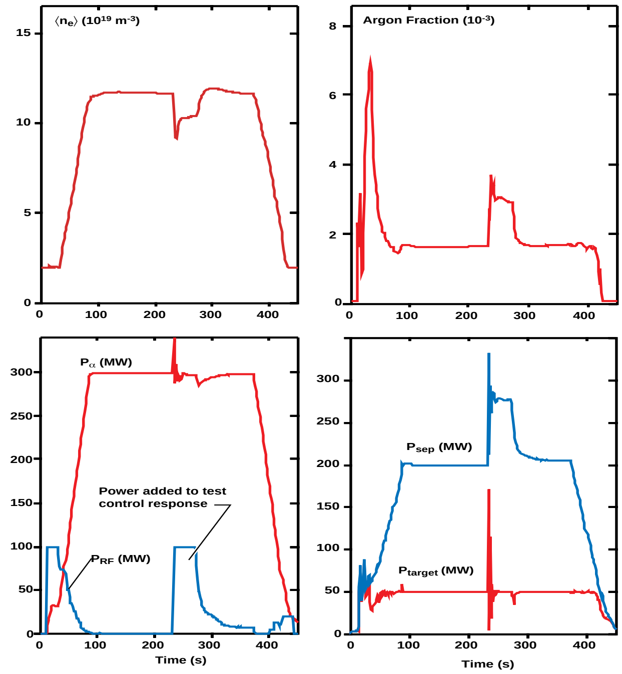

FIG. 5. Simulation of burn and divertor power control during a plasma power input transient.

The physics basis for the ITER kinetic control design derives from the same control techniques — fueling control, auxiliary power control and impurity injection control — that are successfully used in present tokamaks for control of plasma density, plasma temperature and beta and the amount of power reaching the divertor targets. In present tokamaks, these control functions are largely independent. In ITER, ignited or high-Q DT plasmas will be wholly or largely self-heated, and impurities in the plasma core will affect core radiation (bremsstrahlung) and fusion reactivity, so the plasma kinetic control problem becomes more highly coupled as far as the interaction among fueling, heating and impurity injection divertor power control is concerned. In addition, for ITER operation with an ELMing H-mode, the conducted power at the plasma boundary must be sufficient to maintain the H-mode. This H-mode threshold requirement introduces yet another coupling into the kinetic control problem.

The complexities of the overall plasma core, edge (H-mode), and divertor control problem and the importance of profile effects make 1.5-D plasma modeling (transport code simulation) methods the best basis for assessing control feasibility and for developing candidate control approaches and control system parameters. Studies of the feasibility of ITER kinetic control during the fusion burn phase have been examined through the use of time-dependent 1.5-D simulations of the plasma core and edge that incorporate a 1-D divertor model abstracted from more-elaborate 2-D simulations of the full ITER divertor geometry. [1] Application of this self-consistent core/edge/divertor model to ITER shows that for reference assumptions (ELMy H-mode confinement, H-mode power threshold scaling and uniform mixing of impurities (Ar or Ne) injected to control divertor power), control of the fusion power level while limiting the total divertor target power to 50 MW or less is possible.

The kinetic control system concept derived from these studies is shown in Fig. 6. There are three control ‘modules’ for DT fueling (gas or pellet injection), impurity fueling (Ar gas injection) and auxiliary power. The DT module controls either plasma density or fusion power. Density control overrides power control if the requested density exceeds an allowable maximum (limit) density. The Ar module limits the divertor target power to less than a specified allowable power. The auxiliary heating module is configured to control the amount of auxiliary heating, in order of decreasing priority, to *i)* maintain the plasma in H-mode, *ii)* maintain the plasma density below a maximum value, and *iii)* assist in the control of the fusion power. Within each module, control of the respective ‘actuator’ (DT fueling rate, Ar fueling rate or P_{aux}) is effected by a PID (proportional/integral/differential) feedback loop that compares the measured value of the plasma kinetic attribute with a setpoint, or reference waveform. This setpoint is either specified externally by the PCS or (for the cases of plasma-parameter dependent quantities (e.g., H-mode threshold power) calculated in real time from plasma diagnostic data.

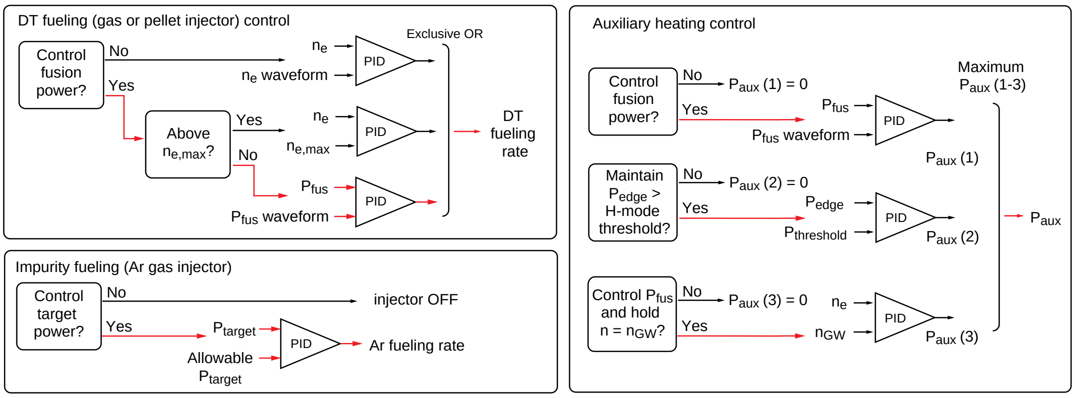

FIG. 6. Concept for the ITER plasma kinetics control system (fusion and divertor power).

Studies of the stability and robustness of the type of controller shown in Fig. 6 have shown that it can maintain fusion power and divertor target power control (power limitation) during both the fusion burn startup and termination phases of the scenario and during the burn 'flattop' despite foreseeable plasma power balance or external actuator disturbances. The ability to maintain control is, of course, dependent on achievement of a satisfactory static plasma burn operating point in which there is finite margin with regard to maximum allowable density and also finite margin for sustained ignition (with $P_{aux} = 0$) or, for driven burn, finite margin for auxiliary power control. Given such margins in the operation point, simulations of the type shown in Fig. 5 demonstrate that reliable and robust control is obtained over a credible range of underlying plasma modelling assumptions (basis for transport coefficient, variation in convective terms, variation in inherent impurity levels, etc.). The ability of the control system to maintain fusion and divertor control when a simulation of a candidate divertor target surface temperature measuring system with finite response time and noise is introduced into the control loop has also been confirmed [9].

The elementary control logic functions incorporated in the plasma kinetics control concept in Fig. 6 anticipate what will undoubted be a more comprehensive implementation of plasma-operation-regime identification and control system reaction capabilities. Here ITER plasma-regime or state-identification will encompass not only the basic operation regime (ohmic, L-mode, H-mode, etc.), but also more subtle distinctions as to plasma core MHD activity (sawtooth characteristics, presence of neoclassical tearing modes, etc.), radiation characteristics (high localized radiation in the divertor or MARFE onset) and proximity to empirical or calculated operation (beta and density) limits. The kinetic control system and the PCS in general will incorporate capabilities to modify the control algorithms of the subordinate control modules on the basis of both plasma state data (including indications of proximity to impending disruption) and the status of the tokamak and its ancillary support systems. This capability will allow the PCS to effect disruption avoidance, controlled plasma shutdown in a normal manner (in ~100 s) for tokamak component protection, and if necessary, fast (in ~ 1 s) shutdown of fusion power and plasma current by impurity injection for component protection if a loss-of-plasma-equilibrium-control or loss-of-[divertor target]-coolant-flow event occurs. As has been noted above, the implementation of plasma-state-cognizant control and the melding of plasma control and plasma-operation-related machine component protection functions is already occurring in present tokamak operation practice.

The ITER PCS can also incorporate a capability to use radially-localized current drive (by EC or possibly LH waves) for control or suppression of the growth of slowly-growing MHD instabilities e.g., neoclassical tearing mode instabilities, that may lead to a confinement limitation on achievable beta. Here modelling [10] of the effect of static or time-modulated localized ECCD on mode growth shows that this control measure can be effective in avoiding what may otherwise be a 'soft' beta limit deterioration of achievable kinetic performance. If on-going experimental tests of such control are successful, the method can be readily added to the functions of the kinetics module of the ITER PCS.

The kinetics control concept embodied in Fig. 6 illustrates that plasma kinetics control in ITER makes widespread use of data available from a suite of plasma diagnostics [11]. These data will provide not only the primary controlled-parameter measurements (density, fusion power and divertor power) but also supporting data on radiated power fraction, plasma core and edge MHD, q(r) profile, etc. This latter class of data, which will contribute to plasma-regime identification, will be essential for the kinetics control of ITER plasmas. Comprehensive data will also be necessary for implementation of pro-active 'before-incident' protection of the ITER plasma-facing-components that are at risk during plasma operation. In this sense, we anticipate that the distinction between the utilization of plasma diagnostic for plasma control, machine protection and the acquisition of scientific data will, like the partitioning of the plasma control and machine protection functions among the ITER control system units, become increasing blurred as the final design of the ITER plasma control system is fully defined.

5. STEADY-STATE AND ENHANCED-PERFORMANCE OPERATION

The design basis plasma operation scenario for ITER is based on sawtoothing ELMy H-mode operation [3] where the 21-MA current flattop during the 1000-s burn is sustained by inductive current drive. Driven-burn variations of this scenario with reduced plasma current and continuous auxiliary power and current drive will allow burn pulses of up to 6000-s duration. The ITER design also incorporates the hardware provisions — including sufficient PF system flexibility and plasma magnetics control capability and also various options for radially-localized heating and current drive — that are anticipated to be necessary to support steady-state operation sustained by non-inductive current drive and bootstrap current. Assessments of the feasibility of achieving such operation in ITER confirm that the capabilities of the design are consistent with known requirements for the reversed-shear (RS) plasma operation modes that are now obtained (mostly on a transient basis) in present tokamaks [12]. However, since the physics understanding of these modes and the operation features required to sustain and control them on a steady-state basis are still subjects of on-going physics R&D, at the present time the degree to which steady-state operation can be achieved in ITER and the details of how such operation will be controlled remain as research to be undertaken in the future.

A number of considerations related to ITER plasma operation and control in an RS mode have emerged. First, weak or negative magnetic shear ($s = r/q\ dq/dr$) plasmas can be produced in ITER by the same method of current and/or shape ramping combined with early auxiliary heating that has been used to obtain enhanced performance RS plasmas in present tokamaks. Second, the ITER PF system and divertor system are compatible with a high-q, high-elongation, high-triangularity plasma that can be obtained by shifting the plasma radially outward with decreased minor radius ($R_o \cong 8.5$ m, $a \cong 2.4$ m, $\kappa_{95} \cong 2.0$, $I_p \cong 12$ MA, $q_{95} \cong 5$). Third, non-inductive sustainment of a reverse-shear current profile and 12-MA current (~80% bootstrap-driven) is consistent to first approximation with 100 MW of current drive power apportioned between on-axis and off-axis deposition. Fourth, the ideal MHD stability of such current profiles and the plasma pressure profile required for up to 1.5 GW fusion power is adequate if stabilization of external kink modes by an ideally-conducting wall located at $r/a \cong 1.3$ is assumed. However, considerations 3) and 4) are based upon *ad hoc* assumptions that it will be possible to simultaneously obtain near-optimal plasma pressure, current density and safety factor profiles, MHD wall stabilization and also reduction in plasma energy transport relative to transport observed in positive-shear H-mode plasmas. In contrast, simulations that self-consistently examine the RS plasma performance attainable with candidate current drive efficiencies plasma densities and various degrees of transport reduction inside an internal transport barrier located at the zero-shear radius of candidate RS plasmas show that attaining $Q > 5$ (> 0.5 GW power) will require obtaining a challenging combination of plasma performance 'enhancement' characteristics including beta levels that will require wall stabilization [9].

There are also significant uncertainties as to how well RS plasmas can be controlled in a steady-state regime. Magnetic control of the high-elongation plasmas needed for RS operation will be less robust than magnetic control of full-bore plasmas. Control of the current profile is also more problematical in the physics sense, since the bootstrap current profile is determined by the pressure profile (which in turn may be determined by the shear and/or rotation profiles in a manner that is not yet fully understood), and also in the practical sense that the ability to arbitrarily control on-axis and off-axis current drive profiles is subject to physics and technological limitations. In addition, how plasma rotation affects wall stabilization and whether 'active' stabilization of the resistive wall MHD mode (the result of resistance in the kink-stabilizing wall) will be required to obtain adequate beta in RS plasmas both remain as open physics issues. These considerations make design of a RS plasma control system and drawing conclusions about its adequacy premature. It is clear, however, that at a minimum, accurate real-time measurement of the q(r) or j(r) profiles and also of the electron and ion p(r) profiles of ITER RS plasmas will be critical to their successful control.

6. SUMMARY

The ITER design defined during the EDA incorporates provisions needed for the operation and control of ignited and/or high-Q driven-burn DT plasmas with fusion powers in the 1-1.5 GW range and fusion burn durations of ≥1000 s. Extension of the burn duration to ~ 6000 s in a reduced-current driven-burn operation mode is feasible. Methods used in present tokamaks for magnetic and kinetic control of present H-mode plasmas are applicable and adequate for similarly robust control of ITER plasmas. Simultaneous control of fusion power and limit of divertor power to acceptable levels appears achievable. The incorporation of regime-identification and control reaction capabilities into the plasma control system is anticipated to make ITER plasma control highly reliable and robust. Steady-state plasma operation with appreciable fusion power may also be possible in a reverse-shear operation mode, but detailed understanding of the physics basis for the attainment and control of such plasmas remains as an on-going task for the world magnetic fusion program.

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