ITER PHYSICS BASIS

CHAPTER 9: OPPORTUNITIES FOR REACTOR-SCALE EXPERIMENTAL PHYSICS

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ABSTRACT. A reactor-scale tokamak plasma will exhibit three areas of physics phenomenology not accessible by contemporary experimental facilities. These are: 1) Instabilities generated by energetic α-particles, 2) Self-heating phenomena, and 3) reactor-scale physics, which includes integration of diverse physics phenomena, each with its own scaling properties. In each area, selected examples are presented which demonstrate the importance and uniqueness of physics results from reactor-scale facilities for both inductive and steady-state reactor options. It is concluded that the physics learned in such investigations will be original physics not attainable with contemporary facilities. In principle, a reactor-scale facility could have a good measure of flexibility to optimize the tokamak approach to magnetic fusion energy.

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1. INTRODUCTION

The emphasis of the ITER Physics Basis Articles is on the physics basis for projections reactor-scale plasma properties and their attendant uncertainties. It is evident from the scope of this report that the overall performance of a fusion reactor depends on the integration of many separate physics and technology phenomena. Uncertainties arise from three major sources: (1) scaling of individual phenomena from present devices to a reactor scale facility; (2) new physics and technology phenomena inaccessible to experimental investigation by contemporary facilities, and (3) integration of the various elements which must be studied separately in today's facilities. Differences in scaling among various physics and technology elements lead to changes in their relative importance in the transition from present devices to a reactor so that the resulting integrated system is scale-dependent. Experiments on reactor scale facilities can diminish these uncertainties and establish fundamental scalings and phenomenology. It is the goal of this Chapter to provide at least a partial answer to the question: What will be learned from operation of a reactor-scale experimental tokamak facility?

However one evaluates the uncertainties in the context of projecting reactor-scale plasma performance, it is clear that the present uncertainties are sufficiently large as to preclude an effective design of a demonstration tokamak fusion power reactor (often called DEMO) on the basis of data currently at hand. DEMO must rest on an experimental basis and projection methodology that has far less uncertainty than those associated with the scale-up from present experiments to a reactor-scale device. For example, factor-of-two uncertainty in the maximum reliable value of β^2 and thus fusion power output, will preclude an efficient power reactor design. Actually, given the magnitude of the scale-up to ITER — for example, a factor-of-20 in energy confinement time at ITER-like β and collisionality — in a system with fundamentally turbulent physics, it is a testimony to the ability of fusion scientists to limit the uncertainties to within a factor-of-2. It follows that physics experiments and technology demonstrations are needed on a reactor scale device to serve as a design basis for DEMO. With such a testbed, extrapolation of plasma performance to DEMO is foreseen to be minimal: One can develop control and operational scenarios — either inductive or advanced — based on reactor-relevant sensors and control actuators. And, the mechanical, thermal, and erosion design requirements will rest on technology data that needs little extrapolation. In this sense, the ongoing debate regarding uncertainties in ITER plasma performance projections actually strengthens the case for a reactor-scale experimental facility. The uncertainties we now face in ITER projections can be considered as opportunities for experimental plasma research in a reactor-scale facility and must be resolved before DEMO is undertaken.

What our present knowledge does support is our ability to define the major parameters and flexibility requirements for a reactor-scale experiment so that the information returned by operating the device will be sufficiently definitive to identify the optimum reactor operating mode, to quantitatively specify the design requirements of such a reactor, and generally to assess whether tokamaks are a viable route to magnetic fusion power reactors. It is a goal of this section to argue that an experimentally-based scientific understanding of the physics of a tokamak reactor needs reactor-scale experiments; it can not be established by experiments with present facilities. Moreover, plasma-wall interaction and tritium retention data from an ITER-class facility will be generic to other toroidal fusion reactor schemes.

This Section describes representative issues for experimental physics on an ITER-class device. Comprehensiveness is not a goal, and many more examples could advanced as appropriate for a program of reactor-scale plasma physics. Burning plasma physics issues are first examined for a nominal, inductive discharge, followed by a discussion of exploitation of how the generic flexibility features of a reactor-scale device can be utilized to investigate advanced tokamak modes and to optimize a reactor-scale device in general. The Section Summary concludes that a reactor-scale experimental facility is an essential element in the development of the tokamak approach to fusion power generation.

2. BURNING PLASMA PHYSICS

The physics part of an ITER experimental program has become known as the physics of burning plasmas. While burning plasma physics is often identified with the physics of modes excited (or stabilized) by energetic α-particles, it is important to recognize that there are actually three elements to burning plasma physics.

2.1. Energetic Particle Effects

The first element is the by-now classic physics of a dilute species of super-Alfenic particles (i.e., α-particles or ion-cyclotron energized tail) whose pressure gradient excites discrete modes of the stable MHD spectrum known as Alfven Eigenmodes [1]. Chapter 5 reviews our understanding of these modes. Energetic particles also can act either to stabilize (m,n)=(1,1) modes, leading to the "monster" sawtooth phenomenology of JET [2, 3] or to destabilize fishbone modes [4]. These effects depend in turn on the spatial distribution of energetic particles which differs for α-particle, NBI, or ICRF heating sources. Finally, the simple fact that α-particles principally heat electrons has the important consequence of eliminating confinement enhancement arising from $T_i >> T_e$ characteristic of many internal transport barriers observed in present experiments [5, 6].

2.2. Self-Heating

The second element of burning plasma physics is the self-heating property, which taken together with the experimentally-documented spontaneous formation of transport barriers at the edge (H-mode transition [7]) or internally [5, 8-10], leads to fuel dilution, thermal stability, and control issues [11]. If steady-state plasmas are to be realized, further self-consistency issues arise regarding pressure profiles, bootstrap current, confinement, and the role of external sources of current drive as well as angular momentum input to control plasma rotation.

2.3. Reactor-Scale Physics

The third element, and arguably the most important, is the scale of the device needed so that α-particle heating will balance transport losses arising from microinstabilities and other causes. Scale enters tokamak physics in many important ways and much of this section will focus on specific examples of this: TAE modes, disruptions, divertor power dispersal and fuelling, operational β-limits, density limit, confinement scaling, and core-edge integration.

3. EXPERIMENTS WITH INDUCTIVE DISCHARGES

Operation of a reactor-scale facility in a nominal ELMy H-mode state will provide opportunties to generate data of essential importance to DEMO in all three elements. This section sketches nine selected examples.

3.1. Energetic Particle Effects on TAE Modes and Sawteeth

It is interesting that even the first of the three elements, Alfven eigenmode instabilities, is predicted to be qualitatively different in a reactor than in present experiments. There are two reasons for this: 1) the relative concentration of destabilizing energetic particles is less in a reactor-scale device and 2) the size of the plasma.

The relative concentration of destabilizing energetic particles follows from the power balance relation

$$\frac{n_{fast}}{n} = \left(\frac{3T}{E_{fast}}\right)\left(\frac{\tau_s}{\tau_E}\right) \qquad (1)$$

where n_{fast} is the density of fast (super-Alfvenic) particles which transfer their energy on a slowing down time-scale τ_S to thermal electrons. In present experiments, super-Alfvenic particles are created by minority ICRF or NBI heating. In this context, we note that particles with $v \approx 0.3 \cdot V_{Alfven}$ also can have a (weaker) stabilizing or destabilizing interaction with Alfven Eigenmodes [12]. The relative energy density of energetic particles is much larger in present experiments than that anticipated in a reactor-scale device because τ_S/τ_E is much larger (by about a factor-of-10). This difference is directly attributable to the scale of the device via τ_E. Thus, even in the present round of DT experiments, energetic particles created by auxiliary power systems dominate Alfven Eigenmode stability physics. Direct effects of fusion α-particles were negligible in the JET DTE1 experiments [6]. TFTR Alfven Eigenmode experiments [13] did show effects of fusion $\alpha-$particle drive, but just in special circumstances. Only in a high-Q reactor-scale experiment will fusion α-particles dominate the stability of Alfven Eigenmodes. Since $\tau_s/\tau_E \quad T^3$ for gyroBohm confinement scaling, variation of temperature in a reactor scale facility will permit investigation of Alfven Eigenmode physics over a range of energetic particle concentrations.

Scale also enters the nonlinear physics of the TAE modes. Theory predicts that the low-toroidal-mode-number-n modes, which dominate present experiments, will be stable in a reactor-scale device. Instead, any instabilities which arise will have moderate-to-high modes numbers $10 \leq n \leq 30$, which opens the possibility of a turbulent mixture of modes and a generalized nonlinear island overlap interaction rather than the isolated low-n modes of present experiments. Section 4.2 of Chapter 5 summarizes the arguments. One can conclude that present experiments have supported the conceptual theoretical framework developed for Alfven eigenmodes modes and have validated sophisticated linear stability analyses [13], but are not able to test theory in nonlinear, reactor-scale situations where the possibility of a turbulent mixture of modes arises.

Turning to sawteeth, α-particle stabilization is predicted to increase the sawtooth period up to ~100 sec on ITER-scale device [2], in spite of the lower relative fast particle concentration anticipated. In this case, the theory is well-developed and validated via comparison to JET "monster" sawteeth [3]. For high temperature ignition, α-particle concentrations anticipated in reactor-scale facilities are predicted to exceed the fishbone threshold. Theory predicts that, because of the large plasma size (relative to a banana orbit), fishbones will cause only a mild spatial redistribution of the α-particles [4]. Reactor scale experiments will test these projections as well as provide an experimentally based sawtooth phenomenology for DEMO design. Research is just starting on whether energetic particles affect neoclassical tearing modes and ELMs. Observation on JET (see Fig. 10 of ref.[14]) indicate quite different electron pedestal pressures depending on whether heating is NBI (energetic particles at edge) or ICRF (no energetic particles at edge).

3.2. Self-Heating and Thermal Stability

By definition, in a burning plasma physics experiment transport losses approximately balance heating from fusion α-particles. In the ultimate case of ignition, attainment of a thermal steady-state results from the fact that transport losses, which according to Appendix D scale as P_{loss} $n_e^{1.6}$ $(T/H_H)^{2.8}$, increase faster with temperature than fusion power P_{fusion} $d(n_eT)^2$ where d is a fuel dilution factor $d = n_T n_D/n_e^2$. A balance gives P n^3 $(H_H)^7 d^{3.5}$. This simple exercise demonstrates that control of core density or dilution jprovides a mechanism for controlling the fusion power output of the device. It also demonstrates the fusion power output is very sensitive to the confinement multiplier H_H, which can undergo abrupt changes in the event of transport barrier formation and to fuel dilution which can be controlled by pellet injection. PRETOR modeling studies for the ITER/FDR design found thermal stability for ignited plasmas within the context of a model wherin a diffusion equation with particle diffusivuity $D \approx \chi_{th}/3$ governed evolution of the density. On the other hand, for an ELM-free, outside-pellet-launch experiment, DIII-D observations [15] indicate nonuniform particle diffusivities with very low core values implying that control of core densities could be difficult. There are presently no ITER Density Evolution Demonstration Discharges which directly measure response of the central plasma density to peripheral pellet fueling nor a database regarding controlling central plasma density via peripheral fueling and the characteristic times scales on which control could be effected. In the case of an abrupt formation of an internal transport barrier for both particles and heat, it remais to demonstrate whether density or fuel dilution changes within the barrier could be carried out sufficiently rapidly to prevent a temperature increase to the β-limit. While one can conceive of burn control simulation experiments in present facilities using feedback from measurements of $T_i(0)$ or neutron rate to control heating power, demonstration of thermally stable operating modes for DEMO must be based on data from a reactor scale facility heated by fusion reactions. In the case of steady-state operation, the simple global issues discussed above are replaced by complex, mutual diffusive evolution of profiles as described in Section 4.2.

3.3. Reactor-Scale Operations – Disruptions and Runaway Electrons

The first experimental objective of a reactor-scale device will be to establish reliable operations. These experiments will utilize hydrogen plasmas to eliminate activation and permit hands-on maintenance and repair. Disruptions will be the among the first issues encountered. It is expected that the evolution of disruptions in a reactor-scale device will follow the same two-phase pattern established in present experiments [16]. A rapid thermal quench phase (~1-10 ms) in which the plasma thermal energy content is deposited on the divertor strike plates and nearby plasma facing components leads to a subsequent, slower (more than 50 ms) current quench phase in which the plasma current is transferred to the surrounding vessel. But, within this conceptual framework, physics phenomena will arise in a reactor scale plasma that are inaccessible to present facilities.

Turning first to the current quench phase, we note that in its hydrogen plasma stage, a reactor-scale device can reach its full poloidal field and plasma current. This large scale and the correspondingly large plasma current bring the phenomena of runaway electron avalanche during a disruption current quench to the fore. Put simply, a large-angle scattering of a runaway electron from a small seed population energizes the secondary electron sufficiently so that it too will runaway, leading to an avalanche of runaway electrons [17]. Scale enters because theory shows that the number of e-foldings Γ the avalanche mechanism will support is proportional to the plasma current and is given by $\Gamma \approx I_p \left(I_{Alfven} \ell n \Lambda \right)^{-1}$, where $I_{Alfven} = 4\pi mc / \mu_o e = 17 kA$. For a reactor-scale plasma, $\Gamma \approx 50$. This is sufficient so that the plasma will transfer its current to a runaway population, in contrast to present experiments. The question becomes: Will each disruption transfer the bulk of the plasma current to runaway electrons, which will eventually impact on plasma facing components? The answer to this question is best determined by experimentation and will depend on plasma impurity content, plasma density, and magnetic surface configuration. The following paragraph makes clear that impurity content will depend on ablation in the thermal quench phase. With regard to magnetic surfaces, it is difficult to theoretically address whether nested toroidal surfaces, which are evidently lost in the thermal quench phase, will reform during the current quench, thereby creating the confinement needed for the runaway avalanche. Experimentally, JT-60U observes that the presence of magnetic fluctuations suppresses runaway generation [18]. Runaway electrons are also potentially unstable to velocity-space instabilities. Reliable experimental results are essential for DEMO design.

Plasma scale also leads to a qualitatively different response to a disruption thermal quench. Based on a constant-β estimate of the plasma energy content, the energy/area striking the divertor target plates in a disruption thermal quench scales roughly according to

$$W/R^2 \quad \beta B^2 R \tag{2}$$

increasing approximately a factor-of-15 from JET to a reactor-scale device. For a nominal discharge with an energy content of 1 GJ, this translates into a divertor target plate loading of roughly 10 MJ/m^2. While for JET the energy associated with a disruption thermal quench can be accommodated by the heat capacity of solid material, the increase of W/R^2 from JET to a reactor implies routine melting and vaporization of the divertor target area and possibly a good fraction of the divertor walls [19]. Section 5.4 of Chapter 4 discusses this phenomenon which is absent in present tokamak operations. Data in this new regime will bear on impurity concentration and runaway electron production in the current quench phase of the disruption as well as providing technology data regarding erosion and redeposition of the ablated/melted plasma facing component material, tritium codeposition, and the effect of disruption debris on diagnostics. Taking the ITER/FDR design as exemplary, a disruption thermal quench will vaporize more than 1μ of tungsten plasma facing material from the divertor chamber walls. This serves as a strong impurity source for the subsequent current quench plasma. Even for initial operation in hydrogen plasmas with 100 MW of auxiliary heating, the energy content of an L-mode plasma will be 0.3 GJ, which induces a heat load of 3 MJ/m^2 — still sufficient to cause vaporization of the divertor target plates and to allow for an initial assessment of the role of impurity production during the thermal quench phase influencing the physics of the subsequent current quench stage.

3.4. Reactor Scale Operations – Power Dispersal and Fuelling

Reliable operations will require effective power dispersal via core-mantle impurity radiation as well as radiation from Scrape-Off-Layer (SOL) plasmas. Even with only 100 MW of auxiliary power in the initial proton plasmas, a completely attached divertor would give power flux to the divertor strike points of approximately 10 MW/m^2 — just at the limits of technological feasibility. While present experiments indicate that tokamak divertor plasmas can fulfill power dispersal goals, the scale of a reactor plasma introduces new features. Three examples can be given. First, because of plasma size, the SOL electron temperature needed to conduct heat from the midplane to the divertor region will be appreciably higher in reactors than in present devices. One can compare a separatrix $T_{e,s} \approx$ 50-100 eV measured in present experiments [20], [21], [22] against the value $T_{e,s} \approx$ 200 eV projected for the ITER FDR design [23]. Since impurity radiation depends on absolute temperature, this will change the radiation and MARFE patterns and has lead to the projection [21] that an ITER-scale device will not exhibit MARFEs inside the separatrix. Second, in present experiments there is a disjointness, discussed in Chap. 4, Sec. 3.3 , between the high density plasmas needed to realize detached divertor operation and the lower densities characteristic of ITER Demonstration Discharges which have the β and ν^* values anticipated for a reactor. The combination of confinement scaling and 2D divertor modeling codes indicates that the disjointness will disappear in reactor-scale experiments. Experimental demonstration is, of course, needed. Third, cross field diffusivities for heat and particles constitute another source of uncertainty and govern the thickness of the SOL. Present experiments can be modeled by two dimensional codes utilizing an ad-hoc but reasonable value for cross field diffusivity [23]. Since theory has yet to provide unequivocal guidance on extrapolating these values to reactor-scale plasmas [24], experiments are needed to fully characterize and demonstrate effective power dispersal from the divertor of a reactor-scale device. Because of the importance attributed to self-regulating carbon radiation by codes [23], divertor power dispersal may well depend on the choice of plasma facing materials.

As reactor scale experiments progress from proton to deuterium to DT plasmas, divertor and edge plasma physics acquires additional objectives beyond power dispersal which are associated with particle and thermal control – the second element of burning plasma physics already discussed in Section 3.2. If ignition occurs, then it is obvious that the plasma must find a thermal equilibrium where transport and radiation losses balance α-particle heating and the self-heating aspects of burning plasma physics come to the fore. Fusion power production in this state depends sensitively on the core plasma density, which is controlled via a combination of (inside pellet) fuelling, particle transport in the edge/SOL plasma, and pumping. Two experimental questions are: 1) What will be the resonse of central density to a inside-pellet-lauch fueling capability? And 2), to what extent will the SOL layer plasma density be an adjustable parameter available to satisfy power dispersal requirements? At present, the most likely core fuelling scenario will be shallow inside-launch pellet fueling, which has a much higher fraction of the pellet fuel retained in the plasma than does outside pellet fuelling [25]. This scenario rests on the assumption that high baffling diminishes the relative role fuel sources from the wall or divertor-target/private-flux regions play in core fuelling compared to pellet sources. Since ITER-like designs differ from present devices in terms of its highly baffled divertor as well as its very long pulse, which will affect wall and recycling fuel sources, this assumption can not be investigated in present devices. It follows that experiments at the reactor-scale are needed to ascertain both the sensitivity of separatrix plasma density to core density and the ability to independently control separatrix density by private flux gas puffing. It also follows that the major control “knobs” for power dispersal will be separatrix density and the radiating impurity species concentration in the SOL and mantle. Reactor scale experiments will determine the required impurity species and concentrations for power dispersal and whether these concentration will adversely affect the fusion reaction rate.

3.5. Operational β-Limit

The overall level of fusion power for a given facility depends sensitively on the β-value which can be supported. Planned nominal, inductive operating scenarios are well below ideal MHD β-limits so those are not of concern. (This is not the case for high-bootstrap-fraction steady-state scenarios, see Section 9.4.2.) The ITER/FDR design is counting on operating at $\beta_N = 2.3$, which is consistent with long-pulse discharges in JET and DIII-D. The expected operational β-value for DEMO is a key design input, and reactor scale physics experiments will be needed to determine this value as well as the physics which governs it. Indeed, there would be quite a difference in DEMO fusion power between $\beta_N = 3.3$, as attained on JET long-pulse discharges [26], and $\beta_N \approx 2.5$ discharges, which on DIII-D and ASDEX-Upgrade develop β-limiting (3,2) and (2,1) MHD modes thought to result from neoclassical island physics [27]. Scale-dependent physics, for example differences in seed island physics resulting from higher plasma conductivity and size, could well be at work. Experimental long-pulse β-limits will be an essential physics output of reactor scale experiments.

Reactor scale experiments will also have the flexibility to pursue experiments aimed at increasing the β-limit via Electron Cyclotron Current Drive (ECCD) stabilization of magnetic islands – a much-discussed technique [28-30] with encouraging initial experimental rersults [31].

The situation regarding n=1,2 neoclassical tearing modes is roughly analogous to the stabilization and control of axisymmetric n=0 modes. The principal is the same: A highly conducting media slows down the growth of unstable mode, so that control systems of reasonable power can be brought to bear. For slowly growing n=1,2 neoclassical tearing modes, the growth rate is determined by magnetic topology changes at rational magnetic surfaces within the plasma and thus is governed by the conductivity of the plasma. A growth time of 10-30 sec is representative — much longer than the growth time of axisymmetric modes. The controller mechanism is modulated electron cyclotron heating power, with the modulation phase controlled by a feedback signal derived from poloidal magnetic field fluctuations. Neoclassical island modes often rotate and 500 Hz is an representative frequency for an ITER-class device with NBI injection. A magnetic control system could not penetrate the ITER conducting vacuum vessel and backplate at this frequency.

3.6. Reactor-scale Fuelling and Core Density

Section 3.2 has identified the key role which core density plays in regulating fusion power. By effective coincidence documented in Section 3.9, the optimum density for a reactor-scale device lies close to the empirical Greenwald density limit value $n \leq n_{GR} = I_{MA}/\pi a^2$ [32]. With gas-puff fuelling, both JET and JT-60U suffer appreciable confinement degradation as $n \rightarrow n_{GR}$ [7, 33, 34]. However, there is not yet any physics argument that would indicate that the Greenwald value is a fundamental limit on core density. Indeed, pellet-fuelled experiments have attained core densities exceeding the Greenwald value by 50% or more and the prevalent opinion is that any density limit is an edge, not a core, density limit [25, 35]. A new round of experiments with efficient inside pellet launch fuelling, which was shown on ASDEX-Upgrade [25] to maintain core density above edge density, will be particularly interesting, both from a physics perspective where toroidal plasma drifts play a role and from reactor-system considerations. Section 3.9 addresses the core-edge integration issues associated with such experiments on contemporary devices. Reactor scale experiments with inside pellet fuelling and a highly baffled divertor, which minimizes main chamber neutral pressure, could maintain a difference between core density and edge density and transcend the Greenwald limit appreciably. The potential increase in fusion power is remarkable: If $n = 2.0 \cdot n_{GR}$ and $\beta_N = 3.0$, then fusion power for the ITER/FDR design would increase from 1500 to 3000MW — a value that would be attractive for an inductive tokamak DEMO. (We note that, for ITER/FDR design, the blanket cooling system could tolerate this value only for approximately 100 s.) Since the high degree of divertor baffling is a result of plasma size (compared to atomic process mean-free-paths), reactor-scale experiments are needed to ascertain the core density limit — if any — as well as differences between gas-puff and inside-pellet fuelling appropriate for DEMO design.

3.7. Confinement Scaling

Confinement projections for reactor scale plasmas rest, in part, on results obtained from ITER Demonstration Discharges which are prepared to be as nondimensionally identical to ITER/FDR design as possible. Nonetheless, reactor size and magnetic field strength combine to make the crucial nondimensional parameter of core physics ρ^* a factor-of-5 smaller than the values found in JET , given comparable values of β and ν^*. From a reactor-performance point-of-view, the key issue is: Does the ρ^*-dependence of confinement scale according to a Bohm, gyroBohm, or some intermediate scaling relation as one progresses from JET to a reactor-scale device? The equivalent physics question is: Is there a separation of scale between the microinstabilities causing the turbulent transport and the device size which governs the gradients which create turbulence in the first place? ITER projections are based on an almost gyroBhm scaling, as discussed in Sec. 7 of Chap. 2. It should be remarked that most first principle transport simulations assume scale separation and are intrinsically gyroBohm in nature. However global gyrokinetic simulations can support large-scale modes [36]. Present experiments find that confinement scaling varies from Bohm for L-mode to gyroBohm for H-mode. But, there is concern regarding whether differential toroidal rotation or energetic particle content in present experiments would bias confinement properties. Reactor-scale experiments will resolve the confinement scaling issue and by extension the scale separation issue with reactor-like levels of toroidal rotation and E_r/B_θ shear, eliminating bias concerns. Extrapolation to DEMO will be minimal.

A second confinement physics issue of importance to reactor plasmas is what has become known as "stiffness." This issue arises in the context of transport resulting from ion-temperature-gradient microinstabilities, which become unstable if the ion temperature gradient exceeds a critical value. Stiffness is, in effect, the rate at which turbulent heat flux increases with ion temperature gradient, once the critical ion temperature gradient has been exceeded. In a stiff model, the increase is rapid, forcing the temperature gradient to remain close to the critical gradient. In its extreme, marginal-stability form, a stiff ion-temperature-gradient confinement model produces a central temperature that is related to the edge temperature by a numerical multiplier. The core temperature is thus determined by the physics determining the edge temperature, whose scaling and scale-separation physics can be quite distinct from that of the core. Flexible ion-cyclotron and electron-cyclotron heating facilities can control power deposition profiles and examine how core temperature responds to edge power deposition and temperature variations.

A reactor scale plasma will provide a new capability to investigate stiffness via transient ion transport physics. In present tokamaks, it is very difficult to induce ion temperature perturbations via modulation of auxiliary heating power because $\tau_s/\tau_E \sim O(1)$, where τ_s is the slowing down time during which an energetic ion transfers its energy to the thermal ions. Because of its size, we expect in ITER that $\tau_s/\tau_E << 1$, and that one can rapidly induce ion temperature perturbations (relative to the energy confinement time) and observe their evolution. Stiff systems will evolve much faster. In present experiments, only electron temperature can be so modulated (by ECH), and investigation of stiffness of ion heat transport must proceed through the intermediary of T_i/T_e ratios. An ion temperature modulational capability, coupled with the ability of ICRF to locally heat ions, will give ITER a qualitatively superior capability to investigate microinstability transport via transient techniques.

Many tokamaks have found that wall conditioning is essential for high performance plasmas. Various techniques — boronization, lithium pellet injection, bakeout of graphite, etc — are successfully employed. The physics mechanism remains to be identified however, but neutral pressure in the main chamber is an evident candidate and inversely correlated with confinement as well as controlled by wall conditioning [37]. Of course, reactor scale facility could inject boron, beryllium, or LiD pellets as present experiments do. But the size of such a facility, coupled with the divertor geometry, creates a very high degree of baffling that can act to provide a new operational regime with trans-Greenwald core plasma densities, yet low main chamber neutral pressure. This demonstration, which is not accessible to present experiments, would most definitely impact operating scenarios for a DEMO.

3.8. H-mode Power Threshold and Pedestal

The nominal operating mode for ITER is the ELMy H-mode, where a transport barrier forms just inside the magnetic separatrix and remarkably steep density and temperature gradients arise [7]. Experiments indicate that a threshold power-across-the-separatrix is needed to effect a transition of the edge plasma from a turbulent L-mode state to the H-mode transport barrier with appreciably reduced turbulence. Regression analyses of existing data indicate that this power will be of order 100 MW for a reactor scale device but with an uncertainty of a factor-of-2 both upward and downward [38, 39]. Since this is comparable to planned auxiliary heating power, the H-mode power threshold is of evident importance to reactor projections.

H-mode physics is an area of fusion science where a qualitative, phenomenological understanding has been attained, but where complexity has prevented development of theoretical expressions appropriate for scaling present experimental results to reactor-scale devices. For example, strong radial electric field (E_r/B_θ) shear is observed to develop in the transport barrier region and can act to stabilize microinstabilities there [40]. But the mechanism which produces the E_r-shear in the first place is not clear and this conceptual approach has yet to produce a scaling relation for the power threshold. In addition to the power threshold, the H-mode is characterized by pedestal temperature and density values just inside the transport barrier, the width of the high-gradient, transport-barrier region, and the extent and consequences of Edge Localized Modes (ELMs) [41-43] destabilized by the steep gradients. Since all these issues bear on how a DEMO device will function and validated theoretical scalings are not available, reactor-scale experiments will be needed to establish a reactor-scale phenomenological understanding to support the DEMO design. In the meantime, a vigorous experimental and database campaign should be maintained in the Parties' base programs to foster the theory of H-mode physics and to create databases which will both guide the theory and support empirical regression projections of H-mode physics for the design of reactor scale experiments.

3.9. Integration of Core and Edge Physics

From a fundamental physics viewpoint, H-mode physics is an example of how different scalings between core and edge physics phenomena make integrated demonstrations of ITER-like core and edge physics possible only in a reactor-scale device. Experiments have determined the H-mode power threshold scaling sufficiently accurately so it is clear that ITER Demonstration Discharges prepared in present devices with ITER-like core values of β and ν^* will have transport losses considerably above the H-mode power threshold, assuring adequate power-across-the-separatrix and H-mode operation. But in an ITER scale device, the different scaling between these two physics processes imply that ITER will operate close to the H-mode power threshold [44], which experiments indicate could change the character of the ELMs associated with the transport barrier from Type I to Type III as well as degrade core confinement [41]. The uncertainties are such that reactor-scale experiments are needed to quantify the power threshold for DEMO as well as to guide theoretical efforts to derive a scaling expression for the power threshold. And, since present experiments near the power threshold do not have cores with ITER-like β and ν^*, reactor-scale experiments are needed to ascertain the consequences of operating near the H-mode power threshold on confinement in an ITER-like core.

Another illustrative example is the ratio of core plasma density to the so-called Greenwald limit value $n_{GR} = I_p/\pi a^2 \quad B/R$. Many plasmas exhibit an edge density limit at this value. But the fundamental reason is not known and atomic physics would have to enter if the Greenwald scaling were to be strictly true. For ITER Demonstration Discharges with constant β and ν^*, core density scales as $n \quad B^{4/3} R^{-1/3}$ so that

$$n/n_{GR} \quad B^{1/3} R^{2/3}. \tag{3}$$

One can argue that an optimized tokamak reactor will operate (coincidentally it seems) at $n/n_{GR} \approx 1.0$, based on a $\beta_N = 2.5$ and $<T> = 10$ keV. Scaling from these reactor parameters to JET via Eq. (3) leads to $n/n_{GR} = 0.4$ for JET. Thus present experiments with an ITER-like core will not be at the Greenwald density limit. An integrated demonstration of confinement in a ITER-like core operating at the Greenwald limit is simply not possible in present machines. It follows that, in present machines, ITER-like core and edge plasma physics must be studied separately; an integrated demonstration is possible only on a reactor scale experiment.

4. ADVANCED TOKAMAK OPERATIONS

Our overall goal is optimization of a reactor-scale tokamak, which will involve many parameters. Successful operation in steady-state or improved-confinement modes is an official goal of the ITER program, as set forth by Special Working Group 1 (see Appendix C). One can ask: What are the flexibility requirements to accomplish this goal? The present experimental status and prospects for advanced tokamak experiments on a reactor scale device are outlined below. In this introductory paragraph, we wish to stress that the operational techniques used to access advanced modes in present devices are, in good measure, available to reactor scale devices as well. Indeed, present experiments indicate that an ITER-like plasma shape is potentially capable of a variety of reverse-shear and internal-transport-barrier configurations as well as the conventional ELMy H-mode scenario. Differences with present machines arise more from reactor-scale constraints and cost issues and less because a particular flexibility feature has been omitted from a candidate design. For example, the poloidal field flexibility of the ITER/FDR design is governed by its monolithic central solenoid and its reactor-like set of poloidal field coils, which lie outside the toroidal field coils. These specific design choices result from tradeoffs among cost, feasibility, and reliability [45]. This contrasts with DIII-D [46] which has PF coils very close to the plasma (not reactor-compatible) and the TPX design study [47] which chose a pancake central solenoid (a different design tradeoff). Turning to divertor flexibility while, at any given time, the ITER/FDR single-null divertor configuration is evidently fixed, the design has retained the ability to change divertor hardware configuration and hence the divertor magnetic configuration and divertor leg lengths. Especially at the 12 MA plasma current levels typical of proposed steady-state discharges for ITER, the ITER PF system has the capacity to create a variety of plasma shapes and positions. Figure 13 of Chapter 1 illustrates one of these.

In common with present tokamaks, reactor scale plasma operations can adjust the relative timing of transformer ramping and auxiliary heating to create discharges with reverse-shear or $q(0) > 1.0$ for 50-or-more energy confinement times. ITER's planned 50 MW, 1 MeV neutral beam system is characteristic of a tokamak reactor system. Its low fuelling rate will likely eliminate improved confinement modes based on NBI fuelling and its lack of ion-heating will preclude modes based on $T_{\mathrm{i}} >> T_{\mathrm{e}}$. But this is a generic feature of toroidal fusion reactors not a lack of flexibility on the part of the ITER/FDR design. The NBI system does inject angular momentum and is expected to maintain toroidal rotation frequencies about 500 Hz. A 50 MW electron cyclotron heating and current drive system is being designed to have appreciable off-axis current drive capability to maintain high-bootstrap-current-fraction discharges as well as the possibility to stabilize neoclassical island modes by feedback-controlled modulation.

Arguably the most technically demanding aspect of tokamak optimization lies in triangularity. This is because changes in triangularity imply changes in internal hardware. For example, high-triangularity, double-null divertor configurations will not be possible on ITER/FDR. Using the ITER/FDR design and Fig.13 of Chapter 1 as a guide, reactor-scale optimization with respect to triangularity and elongation may be best carried out with discharges which do not completely fill the vessel.

At present it is not possible to justify an ITER-class device based on demonstrated steady-state or advanced modes common to at least several tokamaks. A potential overall strategy is to justify the ITER program on the basis of the ELMy H-mode and then exploit the flexibilityinherent in the ITER design to provide a reactor-scale test bed for advanced tokamak research.

4.1. Internal Transport Barrier Modes

During the past several years, many tokamaks of a variety of shapes have found that internal transport barriers, which support very high gradients, can spontaneously arise in a tokamak core [48-50]. Frequently, these barriers are associated with minima in magnetic shear and/or maxima in radial electric field (E_r/B_θ) shear. Taken together, they indicate that, at least transiently, plasma confinement in tokamak cores can appreciably exceed that of the ELMy H-mode on which ITER performance is based. Although historically internal transport barriers have been transient, recent results are encouraging regarding their endurance for many energy confinement times [51] [52]. An ITER-like design can serve as a test bed for reactor-scaleinternal transport barrier research including the topic of power threshold, which experience with edge (H-mode) barriers suggests will be important. Given the variety of tokamaks that have found internal transport barriers, one can argue that it will be more important to have a transport barrier than to optimize it over plasma shape. As an example, the ITER/FDR design should be an effective facility for investigation of transport barrier physics as well as transient ignition and its thermal control for at least 60 sec.

4.2. Steady-State Tokamaks

It is well-known from simple power balance arguments that, for a viable steady-state tokamak reactor, most of plasma current must arise from bootstrap current and that the magnetic configuration should be that of a relatively high-q, reverse-shear plasma so that the bootstrap current can be the dominant source for the poloidal field. To generate appreciable fusion power, a steady-state discharge must have a toroidal β equal to or exceeding those planned for ELMy H-mode discharges ($\beta > 0.028$). Although reverse-shear steady-state tokamak discharges have been demonstrated [5], [51], their β-values lie apppreciably below reactor requiremments. Indeed, the MHD stability picture for steady-state, reverse-shear discharges differs appreciably from the ELMy H-mode plasmas. For the steady-discharges, a conducting wall close to the plasma is needed to assure stability against external kink modes at reactor β-values. In contrast, ELMy H-mode plasmas do not need a stabilizing conducting wall Theory goes on to predict that, for a realistic, resistive wall, “resistive wall mode” instabilities will grow unless plasma rotation lies in a given range [53]. The required amount of plasma rotation remains a subject of theoretical debate and experimental study [54, 55]. Experiments on DIII-D have produced rotating discharges which are computed to be ideally unstable in the absence of a conducting wall and which endure for many wall-penetration times. But, the observed rotation of plasmas in the “wall-stabilized” regime decays in spite of continued injection of angular momentum by beams. Eventually, a growing wall mode triggers a disruption [54]. The assessment is that active n=1 magnetic feedback saddle coils will be needed to stabilize kink modes. Whether such feedback measures will permit sustained operations at the required β-values is just now becoming the subject of experimental studies [56]. Candidate reactor scale designs possesses the poloidal field flexibility to create and control reverse shear discharges as well as a reasonable amount of NBI to induce plasma rotation rapid compared to wall penetration times. On a longer time scale, self-consistency must be attained regarding the α-particle heating profile, transport and the pressure profile, the bootstrap current profile, and the driven current profile. The ability of an ITER-like device to control these profiles is representative of reactor-scale facilities. For example, the planned auxiliary heating sources, particularly the off-axis Electron Cyclotron Current Drive installation, are capable of driving the currents presently estimated as needed for reverse-shear discharges.

Certainly to base a DEMO on internal transport barrier or steady-state tokamak discharges, robust experimental demonstrations will be required on a reactor scale experiment.

5. SUMMARY

A magnetic fusion reactor is characterized by many plasma, atomic, and material processes operating over a wide range of spatial and temporal scales, which combine to yield the total operational system. Success in developing a predictive description lies both in attention to each of the individual physics elements as well as to the overall computational scheme which integrates the combined effects of the individual processes. We have argued that the reactor-scale environment is such that many of the individual process must operate with parameters or constraints not available in present machines, so that definitive experimental investigations of individual processes must be done with an ITER-class facility. Moreover, differences in scaling principles of the various physics elements imply that the integrated operation of an ITER-class device can not be directly simulated by present facilities.

This Chapter has argued that obtaining experimental data which is sufficiently definitive to support the design of a demonstration fusion power station requires investigations on a reactor-scale, burning-plasma facility. It is clear that the physics learned in such investigations will be original physics not attainable with contemporary facilities. Data from present experiments does suffice to define a reactor-scale facility which will return the requisite data. In principle, this facility could have a good measure of flexibility to optimize the tokamak approach to magnetic fusion energy.

REFERENCES

[1] FASOLI, A. S., et al., Plasma Phys. Contol. Fusion, **39** (1997) B287.

[2] PORCELLI, F., et al., Plasma Phys. Control. Fusion, **38** (1996) 2163.

[3] CAMPBELL, D. J., et al., Phys. Rev. Lett., **60** (1988) 2148.

[4] CANDY, J., et al., in 24th European Physical Society Conference on Controlled Fusion and Plasma Physics, Berchtesgaden (European Physical Society, 1997, Vol. 21A), III, 1189.

[5] SHIRAI, H., et al., Phys. Plasmas, **5** (1998) 1712.

[6] KEILHACKER, M., et al., High Fusion Performance from Deuterium-Tritium Plasmas in JET (to be published, Nuclear Fusion, 1999).

[7] WAGNER, F., Plasma Phys. Control. Fusion, **36** (1994) A319.

[8] SYNAKOWSKI, E. J., et al., Phys. Plasmas, (1997) 1714.

[9] GREENFIELD, C. M., Phys. Plasmas, **4** (1997) 1596.

[10] JET TEAM, (presented by F. X. Söldner), Plasma Phys. Control. Fusion, **39** (1997) B353.

[11] WESLEY, J., et al., Fusion Technology, **32** (1997) 495.

[12] BETTI, R., FREIDBERG, J.P., Phys. Fluids, **B4** (1992) 1465.

[13] NAZIKIAN, R., et al., Phys. Rev. Lett, **78** (1997) 2976.

[14] JACQUINOT, J., et al., Overview of ITER Physics Deuterium-Tritium Experiments in JET, (to be published, Nuclear Fusion, 1999).

[15] BAYLOR, L. R., et al., in 22nd European Conference on Controlled Fusion and Plasma Physics, Bournemouth (European Physical Society, 1995, Vol. 21A), Part I, 113.

[16] SCHÜLLER, F. C., Plasma Phys. Control. Fusion, **37** (1995) A135.

[17] ROSENBLUTH, M. N., PUTVINSKI, S.V., Nuclear Fusion, **37** (1997) 1355.

[18] YOSHINO, R., et al., Plasma Phys. Control. Fusion, **39** (1997) 313.

[19] PETSTCHANYI, S., et al., in Proceed., in 24th European Physical Society Conference on Controlled Fusion and Plasma Physics, Berchtesgaden (European Physical Society, 1997, Vol. 21A), Part III, 981.

[20] MERTENS, V., et al., Nuclear Fusion, **37** (1997) 1607.

[21] MAHDAVI, M. A., et al., in Proceed., in 24th European Physical Society Conference on Controlled Fusion and Plasma Physics, Berchtesgaden (European Physical Society, 1997, Vol. 21A), Part III, 1113.

[22] MAINGI, R., et al., in 13th Plasma Surface Interaction Conf., San Diego (to be published, J. Nucl. Mater., 1999)

[23] PACHER, H. D., et al., in Proc., in 13th Plasma Surface Interaction Conf., San Diego (to be published, J. Nucl. Mater, 1999)

[24] CONNOR, J. W., et al., Comparison of theoretical models for scrape-off layer widths with data from COMPASS-D, JET, and Alcator C-MOD, Culham Report UKAEA FUS 396, (1998).

[25] LANG, P. T., et al., Phys. Rev. Lett., **79** (1997) 1487.

[26] HUYSMANS, G. T. A. e. a., in Proceed., in 24th European Physical Society Conference on Controlled Fusion and Plasma Physics, Berchtesgaden (European Physical Society, 1997, Vol. 21A), Part IV, 1857.

[27] LA HAYE, R. J., et al.,, Nuclear Fusion, **37** (1997) 397.

[28] HEGNA, C. C., et al., Phys. Plasmas, **4** (1997) 2940.

[29] ZOHM, H., Phys. Plasmas, **4** (1997) 3433.

[30] PERKINS, F. W., et al., in., in Proceed., in 24th European Physical Society Conference on Controlled Fusion and Plasma Physics, Berchtesgaden (European Physical Society, 1997, Vol. 21A), Part III, 1017.

[31] ZOHM, H., et al., in First Experiments on Neoclassical Tearing Mode Stabilization By ECCD in ASDEX-Upgrade, in Proc. 17 Intl. Conf. Fusion Energy, Yokohama (to be published, IAEA, Vienna , 1999)

[32] GREENWALD, M., et al., Nuclear Fusion, **28** (1988) 2199.

[33] SAIBENE, G., et al., in Proceed., in 25th European Physical Society Conference on Controlled Fusion and Plasma Physics, Praha (European Physical Society, 1998) **B026PR**.

[34] KAMADA, Y., et al. in Fusion Energy 1996, Proc. 16th Intl. Conf. Fusion Energy, Montreal (IAEA, Vienna (1997)) Vol. 1, 247.

[35] MAINGI, R., et al., Phys. Plasmas, **4** (1997) 1752.

[36] SYDORA, R. D., et al., Plasma Phys. Control. Fusion, **38** (1996) A281.

[37] SAIBENE, G., et al., J. Nucl. Mater, **241-243** (1997) 476.

[38] ITER CONFINEMENT DATABASE AND MODELING EXPERT GROUP, presented by T. Takizuka in Fusion Energy 1996, Proc. 16th Intl. Conf. Fusion Energy, Montreal (IAEA, Vienna, 1997) Vol. 2, 795.

[39] ITER H-MODE THRESHOLD DATABASE WORKING GROUP, Plasma Physics Control. Fusion, **40** (1998) 857.

[40] BURRELL, K. H., Phys. Plasmas, **4** (1997) 1499.

[41] ZOHM, H., Plasma Phys. Control. Fusion, **38** (1996) 105.

[42] CONNOR, J. W., Plasma Physics Control. Fusion, **40** (1998) 191.

[43] CONNOR, J. W., Plasma Phys. Control. Fusion, **40** (1998) 531.

[44] PETTY, C. C., LUCE, T.C., Nuclear Fusion, **37** (1997) 1.

[45] ITER JOINT CENTRAL TEAM, Technical Basis for the Final Design Report, IAEA, Vienna, (1998).

[46] LUXON, J. et. al., in Proc. 11th Intl. Conf. Plasma Physics and Controlled Fusion Research, Kyoto, 1986 (IAEA, Vienna (1987) Vol. 1, 159.

[47] NEILSON, G. H., et al., Fusion Technology, **26** (1994) 343.

[48] LEVINTON, F. M., et al., Phys. Rev. Lett., **75** (1995) 4417.

[49] FUJITA, T., et al., Nuclear Fusion, **38** (1998) 207.

[50] VOITSEKHOVITCH, I., et al., Nuclear Fusion, **37** (1997) 1715.

[51] IDE, S., et al., in Fusion Energy 1998, Proc. 17 Intl. Conf. Fusion Energy, Yokohama (to be published IAEA, Vienna, 1999)

[52] GRUBER, O., et al., in Fusion Energy 1998, Proc. 17 Intl. Conf. Fusion Energy, Yokohama (to be published IAEA, Vienna, 1999)

[53] WARD, D. J., BONDESON, A., Phys. Plasmas, **2** (1995) 1570.

[54] GAROFALO, A. M., et al., in Proceed., in 25th European Physical Society Conference on Controlled Fusion and Plasma Physics, Praha (European Physical Society (1998)) **B094PR**

[55] STRAIT, E. J., et al., Phys. Rev. Lett., **74** (1995) 2483.

[56] NAVRATIL, G. A., et al., Phys. Plasmas, **5** (1998) 1855.