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Physics fundamentals for ITER

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Abstract. The design of an experimental thermonuclear reactor requires both cutting-edge technology and physics predictions precise enough to carry forward the design. The past few years of worldwide physics studies have seen great progress in understanding, innovation and integration. We will discuss this progress and the remaining issues in several key physics areas.

(1) Transport and plasma confinement. A worldwide database has led to an 'empirical scaling law' for tokamaks which predicts adequate confinement for the ITER fusion mission, albeit with considerable but acceptable uncertainty. The ongoing revolution in computer capabilities has given rise to new gyrofluid and gyrokinetic simulations of microphysics which may be expected in the near future to attain predictive accuracy. Important databases on H-mode characteristics and helium retention have also been assembled.

(2) Divertors, heat removal and fuelling. A novel concept for heat removal—the radiative, baffled, partially detached divertor—has been designed for ITER. Extensive two-dimensional (2D) calculations have been performed and agree qualitatively with recent experiments. Preliminary studies of the interaction of this configuration with core confinement are encouraging and the success of inside pellet launch provides an attractive alternative fuelling method.

(3) Macrostability. The ITER mission can be accomplished well within ideal magnetohydrodynamic (MHD) stability limits, except for internal kink modes. Comparisons with JET, as well as a theoretical model including kinetic effects, predict such sawteeth will be benign in ITER. Alternative scenarios involving delayed current penetration or off-axis current drive may be employed if required. The recent discovery of neoclassical beta limits well below ideal MHD limits poses a threat to performance. Extrapolation to reactor scale is as yet unclear. In theory such modes are controllable by current drive profile control or feedback and experiments should be forthcoming soon. Recent results on JET and TFTR have confirmed qualitative understanding of alpha particle driven toroidal Alfvén eigenmodes (TAEs). Present predictions for TAE effects in ITER are favourable, but require further work. The large stored energies in ITER have focused attention on disruption physics. Databases for thermal and current quenches, vertical displacement events (VDEs) and halo currents have enabled thermomechanical design. Some questions remain open as to the production, confinement and localization of runaway electrons in potentially unstable plasmas and mitigation strategies have been proposed. Other crucial ITER needs such as diagnostics, control and heating appear to have acceptable solutions.

All this rich physics requires experimental validation by a reactor-scale plasma and care has been taken to provide sufficient flexibility for ITER to cover a wide range of scenarios.

1. Introduction

The studies of ITER physics issues have been carried out primarily by the four Parties' national programmes, monitored by the ITER Expert Groups composed of physicists from the four parties working together with the Joint Central Team. The results will be published in the *ITER Physics Basis* [1], a multi-hundred page document to appear in *Nuclear Fusion*, to which the reader is referred for more details and caveats, as well as extensive references. In this paper, personal views of the critical physics issues are given.

The objective of ITER's engineering design activity phase has been to produce a design for a long-pulse (1000 s) ignited, buildable, operable and maintainable tokamak plasma which would provide the physics and technology basis for a subsequent fusion reactor. Such a 'practical' viewpoint imposes many constraints on the design—shielding, cooling, stresses, vertical control, AC heating of superconductors, etc. Some features which are very attractive from the physics point of view, for example, large elongation, are limited by these engineering constraints. For example, a possible design for a $Q = 12$ ITER 'LITE' to cost about 50% of the full ignited ITER can be found by reducing the major radius by about 25%. However, since the shielding thickness and coil build are fixed, most of the shrinkage is in plasma size so that further size reduction rapidly leads to unacceptable performance. In this paper, all specific examples will refer to the full-size 21 MA design as described in the ITER Final Design Report [2].

A major benefit to the world's fusion community has been the assembling of empirical databases for the purposes of extrapolating to predict ITER behaviour. Some topics in which databases have been assembled are: energy-confinement time, L–H power threshold, β limits and disruption characteristics (thermal and current quench times and halo currents and their asymmetries). Wherever possible, an attempt has been made to compare with theory, at least to the extent of expressing results, in terms of the plasma dimensionless parameters of size, pressure and collisionality

$$\rho^* = \frac{\rho_\mathrm{i}}{a} \qquad \beta = \frac{2nT}{B^2} \qquad \upsilon_* = \frac{\upsilon_{\mathrm{ii}} R}{\varepsilon^{3/2} V_\mathrm{i}}. \tag{1}$$

Here ρ_i is the ion gyroradius, υ_{ii} is the ion collision frequency, V_i is the ion thermal velocity, R is the major radius and $a \equiv \varepsilon R$ is the minor radius.

A number of special experiments have been done in ITER-like geometry generally with ν^*, β and the profiles as those projected for ITER so that the issue is to understand extrapolation to large size (small ρ^*). In passing, a satisfying result obtained was a comparison between JET and DIII-D discharges with all three dimensionless parameters closely matched. The discharges proved indeed to be nearly identical in their behaviour and transport.

Of course, while the design was proceeding, tokamak experiment and theory in the home teams, sometimes as a result of ITER urging, was making progress in a number of directions. Notable among these areas are the following.

(1) 'Advanced tokamaks' (meaning hollow or very flat central q profiles) which could lead to high bootstrap-current fraction steady-state reactors.

(2) A vast increase in computer power and simulation sophistication. We are on the verge of complete fully nonlinear resistive three-dimensional (3D) magnetohydrodynamic (MHD) simulations and of being able to determine self-consistently the electrostatic turbulence (ion temperature gradient mode) and transport by following tens of millions of particles through thousands of time steps.

(3) Energetic particle experiments have studied toroidal Alfvén eigenmodes (TAEs) driven by neutral beam injection, radio frequency (RF) produced ion tails and finally, the fusion alphas on JET and TFTR. Large codes have been developed and qualitative agreement with experiment obtained.

(4) The production of large runaway electron currents remains an important issue for ITER in view of the large avalanching of runaways to be expected at high current. Experiments, particularly on JT60U, show qualitative agreement with theoretical models. Of particular importance, these experiments demonstrated that if a high level of MHD activity can be maintained during a disruption or vertical displacement event, then runaways were not produced. Several in principle proposals now exist for suppressing runaways.

(5) It was predicted almost 15 years ago that the property of bootstrap current being excluded from a magnetic island O-point due to pressure flattening should lead to a growth of resistive tearing modes. Experiments in the last few years have confirmed this, shown β

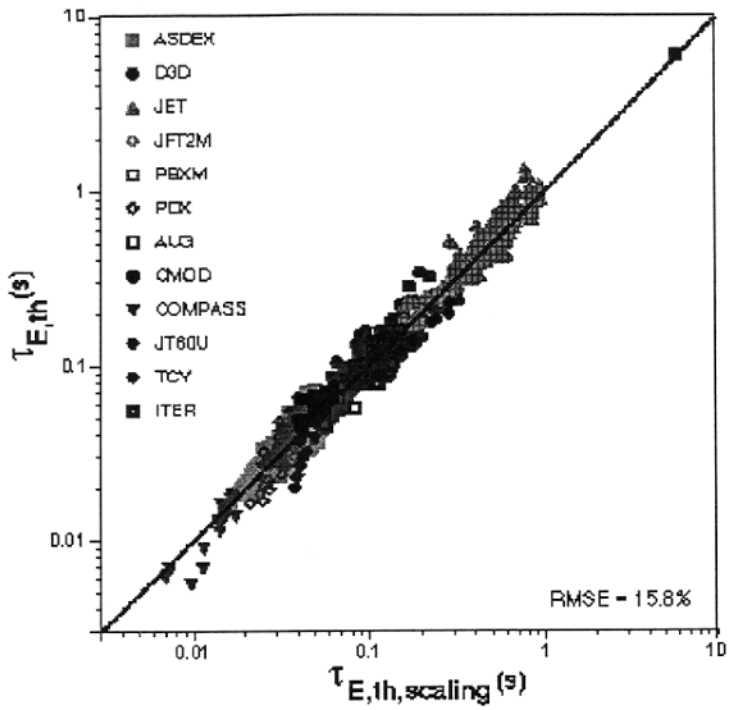

Figure 1. Confinement time for world tokamaks compared to the empirical scaling of equation (2).

limited below ideal MHD values and revealed a vital part of the picture—namely that there is a minimum size 'seed' island which can grow. This seed appears to be formed as a result of sawteeth or edge localized modes (ELMs). The good news for ITER is that island growth is very slow and that modest amounts of localized current drive should suppress these β-limiting modes. Experimental results on mode suppression should be forthcoming in a year. Preliminary successful results have just been reported by ASDEXU [3].

(6) A major conceptual breakthrough has been the development and partial testing of a workable divertor concept to accept the enormous heat leaking out near the separatrix. This can be handled only by a high neutral density, $Z > 1$ region near the strike point leading to radiative distribution of the energy over the whole divertor box. To maintain the proper low neutral density near the plasma, a baffled design has emerged through 2D modelling of plasma and neutral flows.

2. Confinement and simulation

Let us now turn now to a necessarily superficial discussion of progress in some of these areas. The most obvious requirement for a fusion experiment is adequate energy confinement. In figure 1 [4] is shown the world tokamak database for H-mode confinement time with extrapolation to ITER. This data is fitted by the empirical scaling law [5]

$$\tau_E = 0.0365 I^{0.97} B^{0.08} P^{-0.63} n^{0.41} M^{0.20} R^{1.93} \varepsilon^{0.23} k^{0.67}$$
$$[\text{in s, MA, T, MW, } 10^{19}\ \text{m}^{-3}\text{, AMU, m}]. \qquad (2)$$

This scaling can also be expressed in terms of the three dimensionless constants referred to earlier, and predicts 6 s confinement time for ITER. Purely statistical uncertainties amount to $\pm 15\%$, while satisfactory behaviour of ITER requires more than 3.5 s confinement, judged to be at the 95% confidence level, on this purely statistical basis. However, the choice of a power law form for extrapolation is arbitrary and it is difficult to rule out different functional forms, leading to uncertainties in extrapolation to ITER.

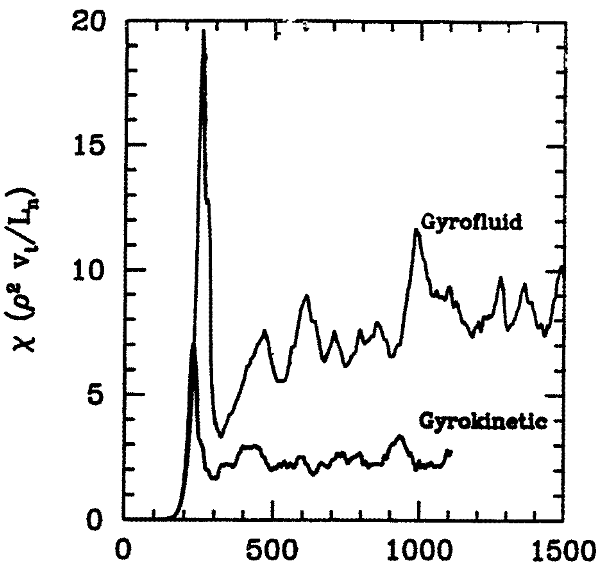

Figure 2. Evolution of turbulent diffusivity against time for gyrofluid (GF) and gyrokinetic (GK) simulations.

In particular, theory and modelling of the ion temperature gradient mode suggest that gradients in ion temperature are limited to lie close to critical $(1/T)(\mathrm{d}T/\mathrm{d}r) < (4/R)$. In such a theory $T_0 \propto T_{\mathrm{ped}}$ where T_{ped} is the edge temperature. Experimentally, the pressure gradient in the pedestal is about equal to the ballooning limit while demanding microstability would limit the width to a few ion gyroradii. Experimental data is quite scattered but suggests $(\rho^*)^\alpha$ dependence for the pedestal width with $\alpha = 0.5 \pm 0.5$. More data is needed. Clearly, a low pedestal temperature such as would be predicted for $\alpha = 1$ and a profile clamped to marginal stability would lead to pessimistic size scaling, At present, different models, while agreeing on linear stability thresholds, differ widely in their predictions of nonlinear behaviour, i.e. the dependence of thermal diffusivity on the ratio of temperature gradient to the critical gradient.

The importance of this issue has stimulated a large computational effort in an attempt to quantify the predictions. Early gyrofluid simulations (IFS-PPPL) indeed predicted very ‘stiff’ behaviour with the profile locked close to marginal and very poor ITER performance. However, it has become clear in the last year through the US cyclone project benchmarking efforts that the gyrofluid moment equations overpredict diffusion as compared to the more accurate gyrokinetic equations which actually follow tens of millions of ions through thousands of steps.

A typical run showing thermal diffusivity evolving as a function of time is shown in figure 2 [6] where it can be seen that after an initial transient, the gyrofluid (GF) predicted flux is about three times larger than the gyrokinetic (GK). As shown in figure 3, the discrepancy becomes still larger if one compares 'flux-tube' fluid models, such as IFS-PPPL, with large-scale full-torus GK simulations. For this case, the overall discrepancy between flux-tube GF and more accurate full-torus GK may be a factor of eight in thermal diffusivity for the particular case examined. The reason for the discrepancy between GF and GK simulations is not yet clear, although it has been noted that sheared poloidal rotation is an important saturation mechanism, and that the GF equations introduce non-physical damping of rotation. Also shown in figure 3 is the MMM–Weiland theory based on zero-dimensional (0D) analytic saturation estimates. In spite of its relatively crude theoretical basis, this theory matches closely the full-torus GK results, plausibly fits the limited experimental database to which it has been applied and has the same ITER prediction (ignition) as the 'empirical scaling' given by equation (2).

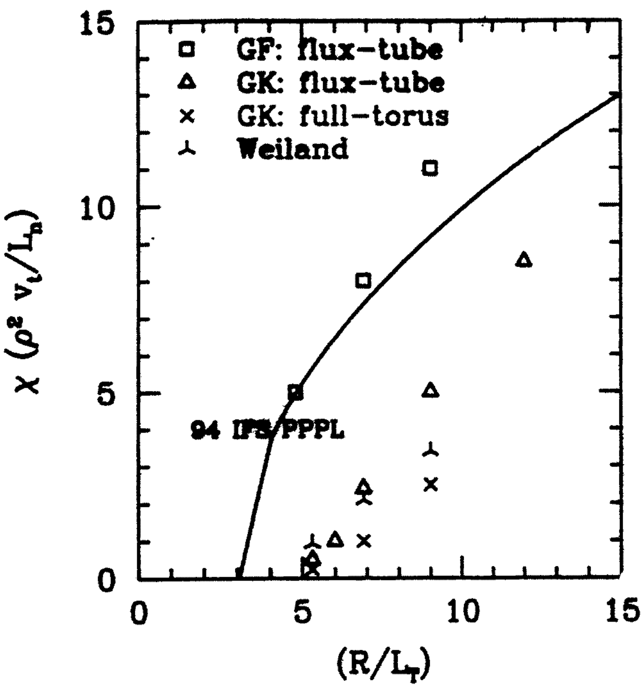

Figure 3. Comparison of turbulent diffusivities for various models as a function of the ratio of major radius to temperature gradient length. Note that the most accurate (full torus gyrokinetic) predictions are considerably smaller than the gyrofluid prediction.

Nonetheless, a real first principles theory is needed. The strong motivation for the gyrofluid modelling is that the codes are fast enough to be able to run full profiles while only a few gyrokinetic simulations can be afforded. Thus, a process of 'renormalizing' the gyrofluid models is under way, while the gyrokinetic codes are being updated to include trapped-electron effects and collisions. A particular need is for better determination (probably empirical) of pedestal temperature.

In spite of complexity of the problem, the rapid growth of computer power and computational sophistication makes for optimism about the progress of the numerical tokamak. A believable first principles transport code could be of great use eventually in assessing the relative characteristics of different confinement schemes and geometries. However, at this time it appeared to the Expert Group to be premature, in view of deficiencies in the present gyrofluid model, to use the simulations as a quantitative predictor of ITER behaviour.

3. Alpha physics

The essence of 'burning' plasma physics is to understand the behaviour of a plasma with a large energetic (super Alfvénic) population. It was predicted 20 years ago that the free energy inherent in the gradient of alpha particle energy density could destabilize shear Alfvén waves, particularly those existing as isolated modes in the gaps of the continuum spectrum. Figure 4 [7] shows the types of modes that can, in principle, exist in such a structure. The original predictions concerned the toroidal Alfvén eigenmodes (TAEs), while energetic particle modes (analogous to fishbones with drift resonance of the alphas) are also a matter of concern.

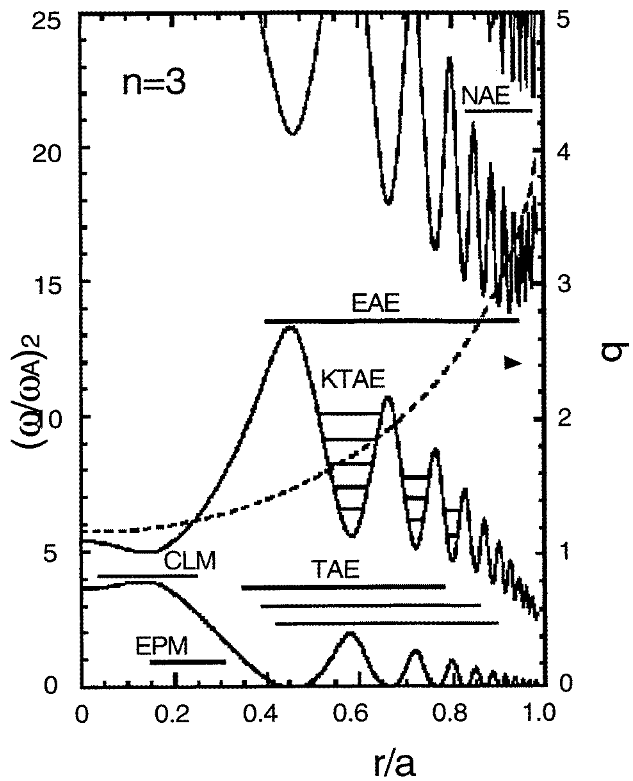

Figure 4. Schematic showing representative shear Alfvén frequency continuum curves as functions of minor radius r for $n = 3$, with horizontal lines indicating the approximate frequency, radial location and mode width for the toroidicity Alfvén eigenmode (TAE), kinetic TAE mode (KTAE), core-localized TAE mode (CLM), ellipticity Alfvén eigenmode (EAE), triangularity Alfvén eigenmode (NAE) and energetic particle continuum mode (EPM).

During the past years, many such modes have been seen experimentally, driven by NBI, ICRF tails and external antennae. A great deal of theoretical work has looked at energetic particle drive and competing damping mechanisms (ion and electron Landau damping, continuum damping and absorption and reflection of mode converted electrostatic modes). In general, reasonable agreement with experimental thresholds and mode structure is found. A real demonstration of understanding was reached with the prediction and subsequent observation in the DT experiments (figure 5) on TFTR and JET of the so-called core localized modes. Since the alpha drive was necessarily small, it was necessary to look in the afterflow of the neutral beam injections (NBI) when the beam damping had disappeared and with a low magnetic shear configuration.

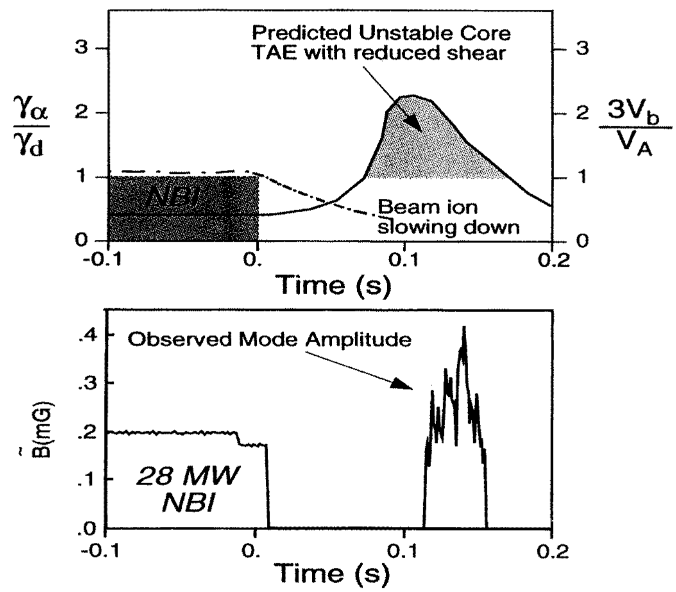

Figure 5. α driven core localized TAE in TFTR.

Thus, there is now a fair degree of confidence in the linear codes for small and moderate values of toroidal mode number n. For ITER only modes with $n > 5$–10 ($\omega^* > \omega$) can be destabilized. According to these codes, the standard ITER profile is close to marginal stability for TAEs, due to the relatively low energetic particle population. However, not all stages of the dynamic profile evolution have been studied, nor have the low shear 'advanced' scenarios.

An interesting issue of principle remains. There are a large number ($\sim$100) of TAE modes of differing toroidal and radial structure with nearly the same threshold and close to marginal as we have seen. Hence, the potential exists that many waves will be excited to a level where the Chirikov overlap criterion for particle orbits is violated. A 'reasonable' assumption is that this would lead to a quasilinear flattening of the alpha density profile. Such a flattening could reduce the gradient to a factor more than five below the critical gradient without loss of alphas. Thus, it would appear that alpha-driven TAEs should not be a problem for ITER 'standard' profiles. However, as pointed out by Berk and Breizman [8], the nonlinear behaviour is very complex and as yet there are no nonlinear codes adequate to rule out destructive bursting rather than the desired diffusive flattening.

4. MHD and neoclassical islands

An important new area of tokamak physics, and an area of some concern to ITER, has opened up in the last few years [9]. It was pointed out 15 years ago that if a magnetic island were formed, then the pressure in the island would flatten via parallel transport and bootstrap current would cease to flow in the island, representing a helical destabilizing current. Including this current, the evolution of the island width, (w/a), is described by a modified Rutherford equation, which for tearing parameter $\Delta' \sim -(m/a)$ with m the helical mode number, would give a saturated island width, w, of order $\beta_{\rm p}\sqrt{\varepsilon}/m$. Considering the large number of possible helicities, this would imply an extremely low β limit and it remained a curiosity that experimental β seemed to be limited only by ideal stability as given by the Troyon limit, $\beta = C(I/aB)$ with $C = 3.5$. ITER needs to operate with β above about 0.6 of this limiting Troyon value.

However, as experiments accessed higher temperatures where bootstrap currents flowed and pulses became long enough to be susceptible to slowly growing phenomena, it began to be observed that there was an effective β limit lower than the ideal MHD value. Islands (3, 2 or 2, 1) were observed and experimental measurement of the island evolution (figure 6) fit very well with theory. There was one further observation—that the islands only appeared above a certain minimum width and hence required a seed event such as a large ELM or sawtooth to form. Several explanations have been given for the breakdown of theory at very small island width, for example, the very long connection lengths for pressure flattening and the importance of kinetic effects for widths comparable to an ion banana orbit. As yet, there is no generally accepted theory, but sparse experimental data indicates a decrease in effective critical beta for island onset as the plasma becomes collisionless and as ρ^* becomes smaller as shown in JET, DIII-D and ASDEXU. If further experiments validate this unfavourable size scaling, then the maximum β achievable in ITER would be considerably less than the necessary value. The improvement in 3D resistivity MHD simulation capability should soon add to our understanding of seed island formation.

$$\tau_{tear}\frac{dw(t)}{dt} = r_s\Delta' + r_s\beta_p(t)\left[(f)\,\frac{w(t)}{w(t)^2 + w_d^2} - \frac{a_{pol}}{w(t)^3}\right]$$

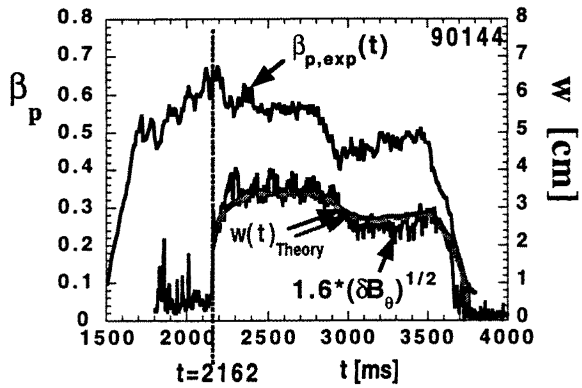

Figure 6. Comparison of neoclassical theory and an experimental (3, 2) island on DIII-D. W is island width, f is a constant of order unity, and w_d expresses the effect of crossfield transport. Note that the island above a certain width grows and saturates in accord with theory.

In principle, there is an easy solution to this problem. The modes grow on the resistive time scale, many minutes for large islands in ITER. Hence, if current can be driven by RF in the O-points of the island, the mode can be stabilized by replacing the 'missing' bootstrap current. In fact, from the geometry of the flux lines, it follows that only radial (and not poloidal) localization is required. If islands can be detected and RF current drive radially localized within a distance d, then the required driven current is $I \approx I_{\rm bs}d/a$. Ray tracing calculations have indicated that d of 10 cm is feasible for ITER and electron cyclotron current drive (ECCD) power of

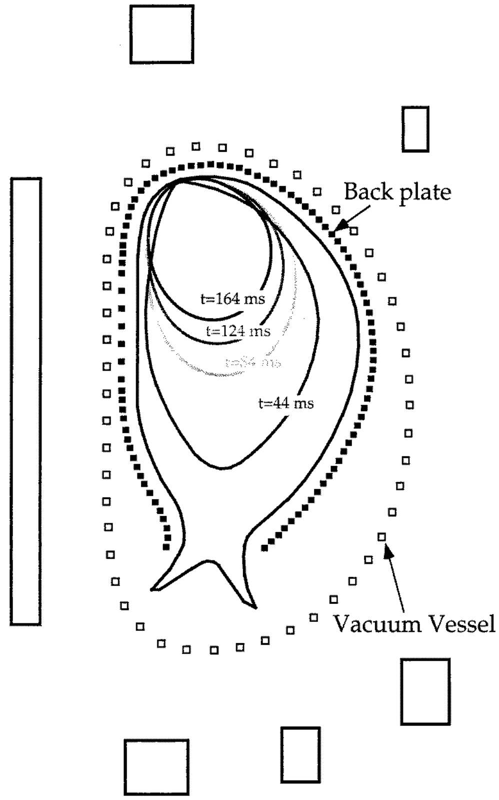

Figure 7. Axisymmetric DINA simulation of motion of a runaway disruption discharge. Times are in ms.

about 20 MW required to stabilize the island chain. Critical experiments are now in progress on ASDEXU and DIII-D to test whether such stabilization works in practice. Preliminary results on ASDEXU are encouraging. At small ρ^*, as in ITER, it may be necessary to stabilize several island chains. Other strategies involve reducing the seed island drive by operating with $q(0) > 1$ or operating in the 'advanced' mode, since with reversed shear the bootstrap current becomes stabilizing.

5. Disruptions

Plasma disruptions are a very large concern for a tokamak with a high stored energy [10]; there is about 1 GJ in plasma thermal energy and 1 GJ in plasma driven magnetic energy in ITER. The phenomenology of disruptions is complex but qualitatively understood—a change in the plasma brought about by impurities, or too high a pressure or density, or secondary reactions to sawteeth or ELMs, leads to a MHD instability which by opening up magnetic field lines, causes a large drop in plasma temperature and high thermal energy deposition, particularly in the divertor. This 'thermal quench' in ITER is expected to be about 1 ms in duration and can lead to vapourization and erosion of the C divertor wall to a depth of a few microns. For metallic divertors, the thermal quench could lead to melting and larger loss. The coupled issues of melting in metals and tritium retention in C need further work. Following this drop in

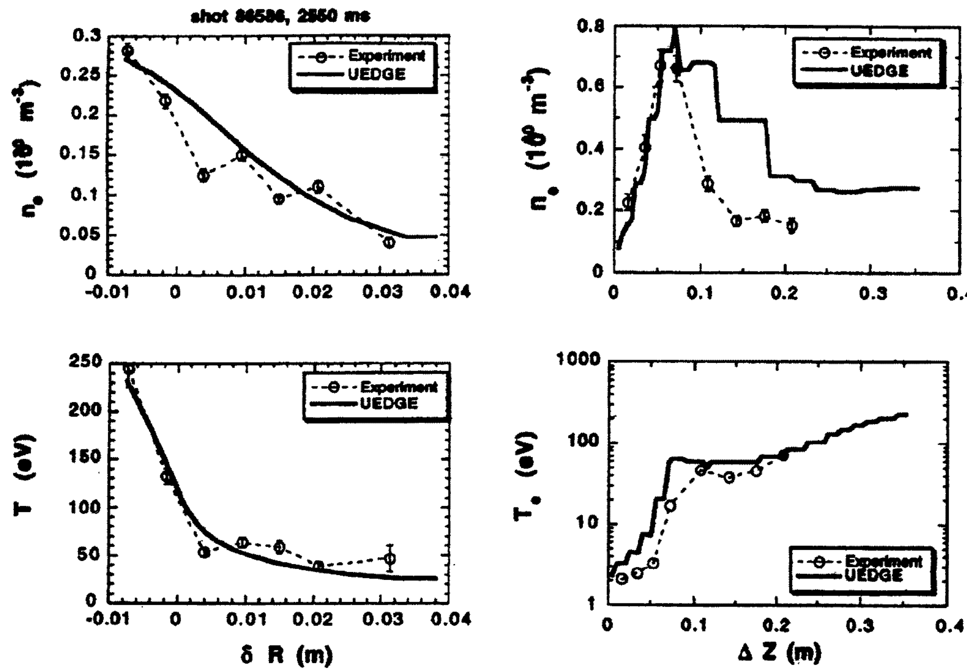

Figure 8. Agreement between 2D code predictions and DIII-D measurement of temperature and densities in divertor as a function of position.

plasma temperature, the resistivity becomes large leading to rapid current decay, the 'current quench'. The change in plasma pressure and inductance may lead to a loss of vertical stability and a vertical displacement event. As the plasma strikes the wall, currents begin to flow in the halo region outside the closed flux surfaces. As first shown in Alcator C-MOD, kinking leads to toroidally asymmetric halos. ITER has been designed to withstand the electromagnetic forces generated by these currents.

There remains the further issue of runaway electrons. This seems likely to be a severe problem for high current tokamaks, where a new mechanism may come into play—the avalanche. For electric fields larger than the minimum drag force, i.e.

$$eE > eE_c \equiv 4\pi n_e \left(\frac{e^2}{mc^2}\right)^2 mc^2 \ln \Lambda$$

high-energy electrons continually gain energy. Here, E is the electric field and $\ln \Lambda \approx 20$ is the usual Coulomb logarithm. Moreover, knock-on secondaries resulting from these primaries can lead to a growth in the runaway current given roughly by [11]

$$\frac{1}{j}\frac{\partial j}{\partial t} = \frac{e(E - E_c)}{2mc \ln \Lambda}.$$

The number of e-foldings of runaway current are thus proportional to the volt-seconds or pre-quench current in the tokamak. $N \approx 2I_{\mathrm{Ma}}$, meaning that even very small seed currents can lead to runaway current of order $(1/2)$ the original plasma current which then remains constant until with $E = E_c$ until the inductive flux (volt-seconds) is exhausted. Such long runaway current plateaux have been observed on many large tokamaks. Careful voltage and current measurements at JT60U showed that the voltage is indeed held to keep $E = E_c$ at the magnetic axis [12].

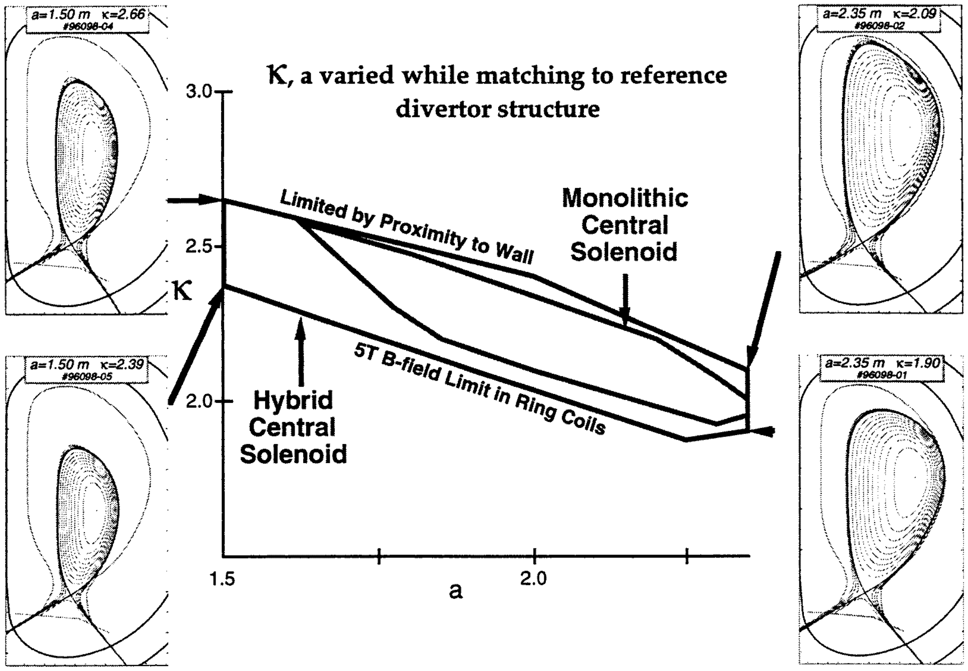

Figure 9. The accessible region of (κ, a) space for ITER advanced steady-state operating mode for both monolithic and hybrid central solenoid design variants. The equilibrium plasma configuration is shown at the four corners of the hybrid operating space.

In order to determine the effects of these electrons, the DINA code was used to calculate the motion of the discharge on the axisymmetric approximation as shown in figure 7 [13]. Included are axisymmetric MHD, halo currents and runaway sources and avalanche. When a flux surface containing runaway electrons hits the wall, the runaways are absorbed. However, the avalanche process replaces them by a skin current of new runaways. Thus, the total energy of runaways striking the wall can be increased from about 25 MJ to 250 MJ. However, the greatest part of these skin current runaways is produced after q_{95} has fallen below one so that, in fact, flux surfaces are probably destroyed by MHD. One concludes that the total energy of runaways deposited on the wall is perhaps of order 50 MJ. The DINA results suggest that these are deposited fairly locally since there is little poloidal motion of the strike point and unknown toroidal localization. We estimate a barely tolerable peak depositions of about 10 MJ m^{-2} but a 3D simulation is needed.

The production of runaway electrons obviously depends on the existence of good flux surfaces. An interesting experiment on JT60U showed that in VDEs, runaways were not seen for initial $q < 2.5$–3 where MHD is likely to be quite intense. This suggests that a strategy for avoidance of runaways would be to deliberately destroy flux surfaces during the current quench. The direct application of helical fields seems impractical for ITER. A more practical strategy would be to trigger successive, small disruptions by inputs of low-Z (to avoid large electric fields) gasses or pellets. Indeed, even without MHD activity, massive deuterium injection can prevent runaway formation. A first step in confirmation of this strategy was obtained in a recent massive He injection into DIII-D which led to a ‘normal’ disruption with no runaways [14].

Table 1.

Topic	Issue	Resources for resolution	Mitigation technique if needed
Sawteeth	α particle effects Neoclassical seed island	3D MHD simulation Comparison of experiment and models	Operation with $q(0) \geqslant 1$
Divertors	Interaction of divertor and core confinement Erosion and melting in disruptions	Experiments and 2D models	Lowering of power Evolving divertor design More frequent change of divertor cassettes
Energetic-particle modes	Non-linear effects in large machines	3D codes	Flatten α deposition profiles lower T_e, higher n
Confinement scaling	Pedestal width Nonlinear stiffness	Experiments, gyrokinetic simulation	Edge shaping, heating, or profile control 'Advanced tokamak' operation
Neoclassical islands	β onset scaling	Theory, experiments	Use of ECCD for profile control or feedback
Disruptions	Runaway electron energy deposition	3D simulations Dedicated experiments	Massive H or He injection Induced kinking or other MHD
Advanced tokamak scenarios	Internal transport barrier power threshold β limits	Experiments + theory	Shaping, profile control, rotation External feedback Reduced β operation

6. Divertors and other issues

An important element of EDA success has been the development of a scenario for a divertor [15] to handle the high heat loads. With the rather narrow expected scrape-off layers, the localized heat flux at the divertor strike points is very large, $\sim$20 MW m^{-2}. Thus, the strategy is to have enough neutral pressure in the divertor to radiate the energy and spread it over the walls of the whole divertor 'box', leading to so-called 'detached (or semi-detached) divertors'. At the same time, neutral pressure must be kept low near the plasma to prevent edge density disruptions. 2D simulations, including Monte Carlo neutral transport, indicated that a closed heavily baffled structure could maintain such a large differential. Recent C-Mod experiments have verified these expectations and shown that an adequate 'compression ratio' can be obtained with a closed bypass structure.

A large research effort has been mounted on the divertor topic. Here we can only mention that remarkable new diagnostics and new 2D simulations now seem in reasonable agreement (see figure 8), and predict satisfactory performance in ITER. A remaining issue is the effect of the divertor on plasma edge pedestals, particularly if enhanced radiation at the edge is needed to reduce heat loads. Preliminary modelling is encouraging that a useful operating space

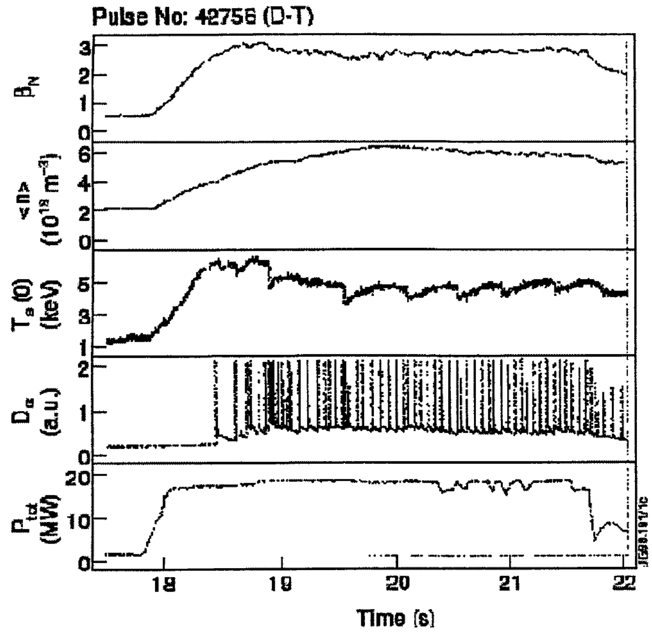

Figure 10. A JET ITER-simulation discharge. β, ν^* and profiles are the same as for ITER.

exists, but more needs to be understood about the type I ELMs which coexist with good core confinement and their divertor interaction.

An issues frequently cited is that the nominal operating point for ITER lies at densities above the empirical Greenwald limit. We note that modelling shows that if the density is constrained to lie below the Greenwald limit there is little loss of ignition margin, although wall loading is decreased somewhat. Recent successes in high density operation in ASDEXU with inside (high-field) pellet launch have led to incorporating such a launcher in the ITER design.

Recent measurements on JET of the L–H mode transition power threshold show an improvement as atomic mass increases. Though there remains large scatter in the data base, it now appears that ITER power is a factor of two above the threshold.

Finally, we can only mention that in the crucial enabling areas of diagnostics [16], neutral beam and RF power [17], the strong JCT-Expert Group effort appears to have found satisfactory design solutions.

7. Advanced tokamaks

It is often argued that an attractive tokamak reactor should be steady state, i.e. with non-inductive current drive. Since RF or NBI current drive is quite expensive, the only strategy is to rely heavily on bootstrap currents with seed RF currents. This implies hollow current profiles. There has been much recent research and experimental activity which indicates that greatly improved confinement can also exist in such ‘advanced’ regimes with the formation of ‘internal transport barriers’ near the zero magnetic shear point, The power threshold scaling for this transport barrier remains to be determined. To date, the fusion production of major tokamaks is about the same (albeit transient) in the ‘advanced’ mode as in the standard mode of operation.

Studying such ‘advanced’ operating modes is an important part of the ITER mission [18]. Figure 9 shows MHD ‘advanced’ equilibria of various shapes which can be produced with ITER shaping coils. These have about 0.6 of full ITER current. With optimized profiles, only a few percent of the current need be RF-driven. At this time, more experiments are needed to understand the causes and cures of the termination of many reversed shear discharges. This seems likely to involve neoclassical islands in the outer discharge regions and possible external kink resistive-wall modes. These factors will determine whether attractive betas can be confined. Probably current drive feedback and possibly ‘rotating shells’ may be required. In any event, retaining and if possible increasing, the flexibility for shaping and profile control in ITER design is essential.

8. Conclusions

A paper such as this, focused on interesting open physics issues, inevitably concentrates on unknown or partially unknown areas and may leave an unwarranted overall negative impression. For many of these physics issues, there is partial understanding, but not yet real predictive capability. We can expect great progress in computational resolution in the coming years and more information from present experiments, but an experiment at reactor scale is ultimately needed.

In table 1 are listed the critical physics topics that have been discussed, the issues, the resources for resolving them and possible mitigation techniques or alternative scenarios. It is important that the ITER design has provided adequate flexibility to implement these measures, for example, sufficient plasma shape control and RF power sources. The issues of neoclassical island stabilization and runaway electron effects appear most likely to require mitigation.

In spite of these uncertainties, the ITER demonstration programme has produced discharges very similar to those required. For example, in figure 10 a JET ELMy H-mode discharge in the ITER single null shape is shown [19]. The discharge is long lasting and has β and confinement required for successful ITER operation. Sawteeth are fairly large (about as predicted for ITER) but non-disruptive. The dimensionless constants, β and ν^*, are the same as for ITER. Thus, we can point to such experiments as true scaled down demonstrations for ITER and focus our attention on the remaining ρ^* scaling. We can identify a probable need to stabilize several bootstrap island chains as size increases and possibly some degradation in confinement due to lower relative pedestal height. However, with experience, it would seen unwise to rule out the occurrence of new phenomena or new modes of operation as we enter the reactor scale regime. The key to success may well be the avoidance of regimes with unpleasant surprises and the exploitation of pleasant ones.

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