PPPL-3246 - Preprint: April 1997, UC-420

THE PHYSICS ROLE OF ITER

Paul H. Rutherford
Princeton Plasma Physics Laboratory

ABSTRACT

Experimental research on the International Thermonuclear Experimental Reactor (ITER) will go far beyond what is possible on present-day tokamaks to address new and challenging issues in the physics of reactor-like plasmas.

First and foremost, experiments in ITER will explore the physics issues of "burning plasmas" -- plasmas that are dominantly self-heated by alpha-particles created by the fusion reactions themselves. Such issues will include (i) new plasma-physical effects introduced by the presence within the plasma of an intense population of energetic alpha particles; (ii) the physics of magnetic confinement for a burning plasma, which will involve a complex interplay of transport, stability and an internal self-generated heat source; and (iii) the physics of very-long-pulse/steady-state burning plasmas, in which much of the plasma current is also self-generated and which will require effective control of plasma purity and plasma-wall interactions.

Achieving and sustaining burning plasma regimes in a tokamak necessarily requires plasmas that are larger than those in present experiments and have higher energy content and power flow, as well as much longer pulse length. Accordingly, the experimental program on ITER will embrace the study of issues of plasma physics and plasma-materials interactions that are specific to a reactor-scale fusion experiment. Such issues will include (i) confinement physics for a tokamak in which, for the first time, the core-plasma and the edge-plasma are simultaneously in a reactor-like regime; (ii) phenomena arising during plasma transients, including so-called "disruptions", in regimes of high plasma current and thermal energy; and (iii) physics of a "radiative divertor" designed for handling high power flow for long pulses, including novel plasma and atomic-physics effects as well as materials science of surfaces subject to intense plasma interaction.

Many of the physics issues of burning plasmas, as well as issues of intense plasma-materials interactions, are generic to any magnetic confinement approach, not just the tokamak.

Experiments on ITER will be conducted by researchers in control rooms situated at major fusion laboratories around the world, linked by high-speed computer networks -- thus extending further what is already a much-acclaimed paradigm for international collaboration in science.

DISCLAIMER

I. INTRODUCTION

The ITER plasma will provide a unique opportunity for reactor-scale plasma physics research. First and foremost, experiments in ITER will explore "controlled ignition and extended burn of a deuterium-tritium plasma" [1], which will involve fundamentally new effects in the plasma physics of magnetic confinement. The successful achievement in ITER of "steady state as an ultimate goal" [1] will require a far higher level of plasma optimization than has been achieved to date experimentally. In addition, the large size and high energy content of the ITER plasma, as well as its very long pulse length, give rise to important plasma-physical effects and plasma-material interactions in presently-inaccessible regimes.

The purpose of this paper is to describe several areas in which the plasma physics of ITER will be fundamentally different from that accessible in present-day experiments, identifying which areas are generic to all magnetically-confined fusion plasmas, not just the tokamak. Effects specific to burning plasmas -- ITER's main role -- are described in Section II. Effects arising from ITER's large plasma size, high energy content and long pulse length -- essential for accessing and studying the burning-plasma regime in a tokamak -- are described in Section III. We also comment on the potential of ITER's operational phase to demonstrate a unique form of international partnership in the conduct of experimental research (Section IV).

II. PHYSICS OF BURNING PLASMAS

The overarching physics role of ITER is to realize, for the first time in controlled fusion research, magnetically-confined plasmas that are self-heated by the fusion reactions themselves. Fusion reactions provide the energy which sustains the sun: it is ITER's goal to create a "man-made sun" in the laboratory.

1. Alpha-Particle Effects in a Magnetically-Confined Plasma

Fusion reactions in a deuterium-tritium (D-T) plasma create a population of extremely energetic alpha particles (nuclei of helium atoms) with energies ranging up to their birth energy of 3.5 MeV. To achieve ignition or near-ignition in ITER, it is essential that these alpha particles be very well confined by the magnetic field.

The "single particle" trajectories of alpha particles in the toroidally symmetric magnetic field of a tokamak remain confined essentially indefinitely; in ITER, their radial excursions away from birth radii are at most about 20 centimeters -- a small fraction of ITER's 2.8-meter plasma radius. Although the small non-symmetric "ripple" in the toroidal field that is unavoidable in any practical tokamak causes the trajectories of some alpha particles to migrate to the vessel wall, the alpha-particle losses due to this process can be kept very small by appropriate magnetic design: for ITER's full-size plasma, less than 1% of the alpha particles can be lost in this way.

A fusion plasma such as ITER's, however, produces such a copious population of alpha particles that "collective effects" can arise, by which the energetic alphas introduce new types of micro-turbulence in the plasma. This topic has been an active area of theoretical research since the mid-1970s, when it was first pointed out that the birth speed of alpha particles exceeds somewhat the Alfvén speed -- the speed at which an important class of naturally occurring waves in a magnetized plasma travel -- thereby allowing the possibility of unstable excitation by alpha particles of certain modes within the Alfvén-wave spectrum. Theoretical work over the past two decades has now led to the identification of one particular mode of this type -- the so-called "toroidal Alfvén eigenmode (TAE-mode)" which depends on the spatial gradient of the alpha-particle population -- as the most dangerous in most practical situations. The threshold for onset of TAE-mode instabilities depends on the density of the alpha particles (usually measured by their contribution to the plasma beta-value -- the ratio of the plasma pressure to the pressure of the confining magnetic field, which in ITER is about 3.0% averaged over the entire plasma and about 10% at the center of the plasma), the steepness of the spatial gradients in the alpha-particle population, and the strength of various damping mechanisms, especially "Landau damping" by the main plasma ions.

One of the primary objectives of the D-T experiments in the Tokamak Fusion Test Reactor (TFTR) was to identify unstable TAE-modes of this type. Since the alpha-particle population in TFTR is quite dilute -- contributing only about 0.03% to the central plasma beta-value -- it was necessary in these experiments to take special measures to weaken the TAE-mode damping mechanisms, for example by making observations only after turn-off of the ion beams used for plasma heating. Weakly unstable TAE-modes with toroidal mode numbers in the range 2-4 were indeed observed [2], but their amplitude

was too small to cause measurable alpha-particle loss. [Deuterium-tritium experiments to be conducted in the Joint European Torus (JET) later this year -- and more intensively in 1999 -- are expected to see similar "incipient" TAE-mode activity.]

The alpha particle population in ITER will be about thirty times more intense than that in TFTR -- contributing as much as 1.0% to the central plasma beta-value, i.e., about a tenth of the main plasma contribution. Moreover, depending on plasma density and temperature profiles, alpha-producing fusion reactions will occur over almost all of the plasma cross section, in contrast to TFTR where they are limited to the central core of the plasma. In these circumstances, theory predicts that many TAE modes may be driven unstable [3], with toroidal mode numbers typically in the range 20-50. Moreover, the unstable modes could be distributed throughout a large fraction of the plasma radius, and the mode amplitudes could become significantly larger. The differences between the TAE-modes observed in present-day experiments such as TFTR and those possible in ITER are summarized in Table I.

	TFTR	**ITER**
Alpha contribution to central beta	0.03%	1.0%
Number of unstable TAE modes	few	many
Unstable toroidal mode numbers	2-4	20-50
Amplitude of the mode's magnetic field perturbation	0.001%	0.01%
Radial extent of possible unstable modes	local, no overlap	broad, with overlap
Alpha particle losses	insignificant	possibly large

Table 1: Comparison of alpha-driven TAE modes in TFTR and ITER

Recent computational studies of the turbulent stage of unstable TAE-modes exhibit many of the features found in contemporary theories of the onset of stochasticity in nonlinear dynamical systems. For an isolated TAE-mode,

saturation occurs due to particle trapping in a "resonant" drift-orbit "island", and the overall outward transport of alpha particles is very small; this effect is predicted to be dominant in present-day experiments such as TFTR. In ITER, if a wide range of higher-mode-number TAE-modes does prove to be unstable, the amplitudes could in some cases become sufficiently large for stochastic diffusion due to "overlapping resonances" to occur, in which case there would be significant transport of alpha particles out of the plasma [4].

The ITER plasma will provide the first-ever opportunity to study experimentally the physics of these alpha-particle effects -- effects which are generic to all magnetically confined D-T fusion plasmas, including most "alternate concepts" as well as the tokamak. By varying the plasma parameters and profiles, it should be possible to access regimes where TAE modes cause significant outward transport of alpha particles, as well as the more favorable standard ITER regime where alpha-particle losses are expected to be insignificant. The ITER program is presently sponsoring the development of suitable "lost alpha" detectors and is planning to deploy infra-red cameras to observe regions of the vessel inner wall on which lost alphas could impinge, as well as sophisticated diagnostics to identify and measure alpha-driven internal turbulence (for example, by microwave reflectometry). The study of alpha-particle effects will be one of the most exciting elements of ITER's experimental physics program.

2. Confinement Physics with a Self-Generated Heat Source

Confinement physics in the ITER plasma will differ from that in present-day tokamak plasmas in that the dominant plasma heat source in the plasma will be internal and self-generated, i.e., from the alpha particles produced by the fusion reactions. Internal self-generated heating introduces a new "feedback loop" into the coupled physics of plasma transport, stability and heating.

The dominance of the internal self-generated heat source over externally applied heating is achieved in ITER in many different experimental conditions and for a wide range of plasma performance. The fusion performance of a D-T plasma is usually measured by the Q-value -- the ratio of fusion power output to heating power input. Plasma "ignition" occurs when the plasma is sustained entirely by self-generated heating, and the external heating power can be turned off altogether, at which point the Q-value becomes infinite. Moreover, since the total fusion energy produced in a D-T reaction (17.6 MeV) is

approximately five times the energy of the confined alpha-particle (3.5 MeV), the self-generated heat source exceeds externally applied heating whenever $Q > 5$. On the basis of scalings derived from tokamak experiments worldwide, the ITER plasma is projected to achieve an "energy confinement time" of approximately 6.0 seconds, which would be sufficient for ignition. Even if some degradation of the confinement scaling were to result in an actual energy confinement time of only 4.0 seconds, it will still be possible in ITER to realize regimes with $Q > 10$, which is sufficient for the self-generated heat source to exceed externally applied heating by a factor of two, or more.

Plasmas in which the internal alpha-particle-generated heat source is dominant are said to be "burning". Since fusion reactivity varies with the square of the plasma density and varies even more strongly with the plasma temperature, the intensity of the self-generated heat source will change as the plasma parameters and profiles evolve. In addition, the plasma's internal stability will be influenced by the evolution of the profile of plasma pressure (product of density and temperature), although the standard ITER plasma is predicted to be comfortably within gross stability limits. By contrast, in present experiments, the intensity and profile of externally-applied heating is largely controllable. The unique feature of a "burning plasma", which cannot be realized on any present-day experiment, is that the time-evolution of the plasma profiles, in particular the density and temperature profiles, will be determined by the self-consistent interplay of self-generated heating, transport and stability.

3. Steady-State with a Self-Generated Heat Source

Confinement physics in the ITER plasma will differ from that in present-day tokamak plasmas also because the plasma pulse length will be much longer than all time-scales characteristic of plasma-physical effects or plasma-vessel interactions.

The plasma pulse length achievable in a tokamak experiment (which is generally determined by engineering considerations, such as the maximum pulse lengths of the magnets and heating systems and the number of "volt-seconds" available in the transformer for "inductive" drive of the plasma current) should be compared with a hierarchy of time-scales characteristic of various relevant physical processes. In ascending order, the most important time-scales characteristic of plasma-physical processes in a D-T tokamak are (i) the energy confinement time, (ii) the time for significant build-up of the "helium ash"

arising from thermalized alpha particles, and (iii) the "skin time", which characterizes the slow evolution of the current profile within the plasma. The various interactions between the plasma edge and the vessel wall (including the "limiter" and "divertor", etc.) introduce another hierarchy of characteristic time-scales; some of these are extremely short (such as the time for reaching ionization equilibrium between the plasma and the surrounding neutral gas), while others are extremely long (such as the time for reaching equilibrium conditions on the inner surface of the vessel wall itself, especially in regions away from the most intense plasma interaction). The longest of these time-scales, which we call the "plasma-wall equilibration time", is the most challenging to the demonstration of true steady-state-like conditions.

Approximate values for the various characteristic time-scales are given in Table 2, both for ITER and for a typical present-day large tokamak experiment of the JET/JT-60U/TFTR class. It is apparent that only ITER is able to explore two key "long-pulse" physics effects in burning plasmas, namely (i) build-up to saturation of the helium ash, and (ii) evolution of plasma profiles over several "skin times" with an alpha-particle internal heat source. In addition, only ITER will have a sufficient pulse length to approach plasma-wall equilibration. Most of these effects are generic to all long-pulse magnetically-confined burning plasmas: they are not specific to the tokamak.

	Present-day large tokamak	**ITER**
Energy confinement time (secs)	0.5	6
Helium "ash" build-up time (secs)	NA	20
Plasma "skin time" (secs)	20	500
Plasma-wall equilibration time (secs)	100-1,000	100-1,000
Plasma pulse length (inductive) (secs)	10	1,500

Table 2: Time-scales relevant to long-pulse/steady-state plasma operation

The conditions on the inner surface of the vessel wall in ITER will be affected not only by the very long pulse length, but also by the high "duty factor", or pulse repetition rate (one 1,000-second pulse every 2,200 seconds). Moreover, since ITER's superconducting magnets will remain energized during the interval between pulses, certain techniques used in tokamaks for between-pulse wall preparation will not be feasible in ITER. Overall, it seems likely that wall conditioning, which plays a major but not-well-understood role in optimizing confinement in present-day experiments, will be different in a very-long-pulse/high-duty-factor tokamak such as ITER. Issues of wall conditioning are generic to most magnetic-confinement fusion reactor concepts.

By introducing "non-inductive" current drive, ITER can achieve plasma pulse lengths far longer than the maximum 1,500-second purely inductive pulse [5]. Non-inductive current drive has been successful in sustaining high-performance plasmas in JT-60U [6]. All of ITER's candidate neutral-beam and radio-frequency heating systems are being designed to provide current-drive capability. However, since the efficiency of neutral-beam and radio-frequency current-drive is relatively poor, the realization of fully-steady-state plasmas (in practice, pulse lengths up to 10,000 seconds) in ITER will require that a large fraction of the plasma current be provided by the "bootstrap" effect. Reference [5] gives the results of a calculation of the current profile in a 12-MA steady-state plasma in ITER, showing that the entire current can be provided by the bootstrap contribution (70%) together with a non-inductive contribution (30%) produced by neutral-beam or radio-frequency techniques.

The "bootstrap current" is a self-generated contribution to the plasma current which arises spontaneously in tokamak plasmas with sufficiently high beta-values and sufficiently low inter-particle collisionality. In ITER, the plasma temperature is easily large enough to provide the needed low collisionality, but to achieve the needed beta-values, it will be necessary to operate the plasma much closer to its stability limit. Global stability depends on the beta-value and on the shape of the plasma pressure and current profiles. Local stability depends on the local pressure gradient and on the local "magnetic shear" -- a property of the current profile shape. At any location in the plasma, the density of bootstrap current depends on the local density and temperature gradients.

Thus, just as the time-evolution of the <u>two</u> plasma profiles (density and temperature) relevant to moderate-pulse-length burning plasmas is determined by the self-consistent interplay of <u>three</u> processes, namely self-generated

heating, transport and stability, the time-evolution of the three plasma profiles (density, temperature and current) relevant to steady-state burning plasmas is determined by the self-consistent interplay of four processes, namely self-generated heating, transport, stability and self-generation of bootstrap current. Experiments on ITER will determine whether some form of active "intervention" will be needed to adjust plasma profiles to desirable shapes. Fortunately, the current-drive system provides a means for such intervention; special types of highly-localized current drive can also be used to improve local stability at higher beta-values.

Extensive experimental studies will be needed to attain and understand this highly optimized and complex steady-state plasma regime. By utilizing more strong-shaped plasmas at reduced current, such studies are well within the capabilities of ITER, but go beyond what could be done on any existing tokamak -- or indeed on any of the previously-proposed short-pulse burning-plasma experiments.

III. PHYSICS OF REACTOR-SCALE PLASMAS

Achieving and sustaining burning plasma regimes in a tokamak necessarily requires plasmas of large size, high energy content, and long pulse length. Although scalings and computational models derived from present experiments are sufficient to provide a reliable basis for the design of ITER, many important plasma-physical effects, as well as plasma materials interactions, will be combined together at reactor scale for the first time on ITER. Accordingly, the physics role of ITER must include the experimental study of issues specific to its reactor-like size, energy content and pulse length.

1. Integration of Core and Edge Reactor-Scale Confinement Physics

It is not possible with present tokamak facilities to produce plasmas which permit the study and optimization of ITER-like core plasma physics and ITER-like edge plasma physics simultaneously.

The plasma physics of a tokamak core involves such issues as energy transport, stability limitations on the plasma beta-value, and (in the case of D-T plasmas) confinement of energetic alpha particles and transport of the thermalized helium. The plasma physics of a tokamak edge involves such issues as the plasma density limit, the power required to effect a transition to the

so-called "H-mode" of confinement with its favorable edge "transport barrier", the "edge pedestal" pressure-values which such a barrier can sustain, and the physics of relaxation cycles called "edge-localized modes" which destroy the transport barrier and provide periodic releases of energy from the edge plasma. All of these issues bear importantly on the performance of ITER -- and indeed of any reactor-scale tokamak. However, the core physics issues and the edge physics issues must be studied separately on present-day tokamaks, and separate projections to ITER must be developed and then combined into an overall assessment of ITER plasma performance This is because the fundamental scaling relationships used to establish the similarity, from the viewpoint of plasma physics, between ITER and specific experiments on present-day tokamaks are different for core physics and edge physics.

Projections of core physics performance in ITER are based on the concept of scaling from present experiments to ITER by using the appropriate "non-dimensional parameters". Such methods are employed commonly in other fields of continuum physics, especially where turbulence plays a role as it does in the tokamak core. The non-dimensional parameters appropriate for describing transport in a high-temperature fully-ionized magnetized plasma were identified many years ago by Kadomtsev: they are the plasma beta-value, an appropriately normalized measure of the plasma inter-particle collisionality, and the number of ion gyro-radii (the radii of the small circular orbits which charged particles make in a strong magnetic field) that will fit into the plasma (minor) radius.

Experiments have confirmed Kadomtsev's thesis that different tokamak plasmas with the same non-dimensional parameters will have the same confinement time (appropriately normalized to make it, also, non-dimensional) even though the actual experimental facilities are of different size and have quite different dimensional plasma parameters. To carry out these experiments, it was necessary to ensure that other intrinsically-dimensionless parameters, such as the ratio of the major radius of the toroidal plasma to its minor radius (aspect ratio), the degree of vertical elongation of the plasma cross-section, and the "pitch" of the helical magnetic field formed by combining the toroidal and poloidal components, were also the same.

The use of non-dimensional parameters in scaling core physics from present-day tokamaks to ITER rests on the ability to produce plasmas with ITER-like beta-values and inter-particle collisionality, as well as ITER-like values of

the intrinsically-dimensionless parameters (aspect ratio, elongation of the plasma cross section, and helical pitch of the magnetic field), in both DIII-D [7] and JET [8]. In addition, these experiments are able to match heating and plasma profiles reasonably well. Comparison of the DIII-D and JET experiments determines the scaling of confinement time with the one remaining non-dimensional parameter, namely the number of ion gyro-radii in the plasma radius, and this scaling can then be used to project to ITER. The non-dimensional parameters of these so-called "ITER demonstration discharges" are given in Table 3: it is seen that the extrapolation from JET to ITER is a factor-of-three in the number of ion gyro-radii (from 300 to 900) -- to be compared with a factor-of-two variation in this same parameter between DIII-D and JET. (The JET team has also carried out a similar scaling study within JET itself by varying the number of ion gyro-radii in the range 150-300 while other non-dimensional parameters are kept fixed [8] -- these experiments confirm the scaling obtained from the DIII-D/JET comparison.) The confinement time projected for ITER by this method is slightly more than the 6.0 seconds required for plasma ignition.

	DIII-D	**JET**	**ITER**
Similar Parameters			
Aspect ratio R/a	2.7	3.1	2.9
Plasma elongation	1.7	1.7	1.6
Edge q-value (measures helical pitch of field lines)	3.8	3.2	3.1
Plasma thermal beta value	2.3%	2.4%	2.4%
Plasma collisionality relative to ITER	1.0	1.5	1.0
Varied Parameter			
Number of ion gyro-radii in plasma radius	150	300	900

Table 3: Parameters of ITER demonstration discharges in DIII-D and JET

The principle difficulty with this approach is that the edge physics in a tokamak appears to have different scalings from the core transport physics. It turns out that the ITER plasma will operate at, or close to, two edge-dominated empirical limits on good tokamak behavior, namely (i) an empirical upper limit on the plasma edge density, usually called the "Greenwald limit", and (ii) an empirical lower limit on the heating power flowing across the edge needed to access the favorable confinement regime, usually called the "H-mode power threshold". However, because the core physics and edge physics scalings are different, the "ITER demonstration discharges" in DIII-D and JET have densities well below the Greenwald limit and heating powers easily exceeding the H-mode power threshold.

Indeed, the fundamental scalings of plasma physics derived by Kadomtsev imply that it is impossible to produce a plasma with ITER-like beta-value and inter-particle collisionality and a density at the Greenwald limit on any of today's-size tokamaks. Typically, any present-day plasma at the Greenwald density limit will have a core-plasma inter-particle collisionality greater than in ITER.

Although the H-mode power threshold is not yet described by any well-validated empirical scaling, similar considerations apply here also. Present-day plasmas in which the heating power far exceeds the H-mode threshold generally exhibit a more violent form of edge-localized modes compared with the more benign form characteristic of ITER-like operation just above the H-mode threshold. Conversely, present-day experiments with power only marginally exceeding the H-mode threshold do not attain ITER-like non-dimensional core plasma parameters.

The edge physics of the density limit has been analyzed in one model [9] that is able to derive the scaling of the Greenwald limit by the hypotheses (i) that the edge pressure gradient is at the local stability limit, (ii) that the width of this region is approximately one ion gyro-radius evaluated with the poloidal (rather than toroidal) magnetic field, and (iii) that the ion temperature in this extreme edge-region is limited to about 100 eV, perhaps by atomic processes such as radiation and charge-exchange -- which lie outside Kadomtsev's universe of purely plasma-physical efforts.

The edge physics of the transport barrier, which forms over a somewhat broader edge region when the H-mode power threshold is exceeded, is currently the subject of intense theoretical, computational and experimental

research. The reduced energy flow through this transport barrier leads to an edge "pedestal" on the ion temperature profile, the height of which will be important in determining overall energy confinement in ITER [10]. There is now considerable experimental evidence [11] in support of theories [12] which explain the transport barrier in terms of a reduction in turbulent transport caused by "sheared" flow within this broader plasma edge region. Sheared flow arises from plasma rotation in the toroidal direction, typically as a consequence of momentum injected in neutral-beam heating and, even in the absence of rotation, from radial electric fields which arise to compensate the so-called "diamagnetic drift"; the latter effects are often dominant and are proportional to the plasma pressure gradient. The toroidal rotation in present tokamaks tends to be larger than is projected for ITER, with the result that the so-called "ITER demonstration discharges" in DIII-D and JET are not well-matched to ITER in regard to an appropriate dimensionless measure of sheared toroidal flow.

The suppression of turbulent transport by sheared flow has been seen in computational simulations using a "gyro-kinetic" model [13]. However, in the "strong turbulence" regime predicted for the edge region of ITER, the turbulence can itself produce a sheared flow in the poloidal direction, which can be the dominant effect in suppressing turbulent transport, depending on the rate of plasma damping of poloidal flow. The theory of these complex effects, which could determine the width of the transport barrier and the height of the temperature pedestal (and, thereby, the overall confinement) in ITER, is still in an evolutionary stage. If the width of the transport barrier can be extended over several ion gyro-radii evaluated with the poloidal magnetic field, then an adequate temperature pedestal should be achievable in ITER.

Experimental work on present tokamaks can investigate separately various aspects of core and edge plasma physics. Indeed, present projections of ITER performance are firmly rooted in the physical understanding and the empirical databases developed from these experiments. However, an integrated demonstration of reactor-like physics in both the core and edge plasma regimes requires an ITER-scale experiment. Ultimately, the plasma physics experiment which will determine the validity (or otherwise) of any theory or model that purports to describe overall transport in a reactor-scale tokamak will be ITER itself!

2. High-Current Plasma Disruption Phenomena

All tokamak plasmas are subject to occasional rapid termination events, called "disruptions". Disruptions impose challenging design issues for ITER, which are being resolved as the present engineering design phase progresses. Although much of the physics of disruptions is tokamak-specific, similar plasma/electromagnetic transients are likely to occur in other magnetic-confinement concepts -- especially those involving high-beta and high-power-density plasmas; solutions being developed for ITER may be applicable to these other concepts also, once they reach reactor-like parameters and size.

During a typical disruption, most of the plasma thermal energy is lost almost instantaneously, and the plasma current rapidly decays. Plasmas with strongly shaped cross sections, such as in ITER, are also subject to a particular kind of disruption, called a "vertical displacement event (VDE)", in which the entire plasma column moves vertically (often toward the divertor, i.e., downward in ITER) as the current decays.

Disruptions are "off-normal" plasma events, and they are normally avoided by operating away from known plasma limits and by appropriately programming the plasma start-up phase to provide satisfactory profiles. However, even if disruptions occur only infrequently, they could pose a threat to the practicality of a tokamak reactor, because of the sudden deposition of the plasma thermal energy onto plasma-facing components of the vessel wall and the rapid transfer by electromagnetic induction of much of the plasma current to nearby conducting structures; these effects produce high thermal and mechanical stresses and can also result in significant surface erosion or damage of plasma-facing components.

Fortunately, ITER is being designed to accommodate even "worst case" disruptions and VDEs; the vessel and the shield-blanket structure will both withstand the maximum projected electromagnetic forces, and the plasma-facing components will survive the heat loads from disruptions for lengthy operating periods before their replacement will be required.

Since disruptions were first observed on small tokamaks in the mid-1960s, their underlying physics has been the subject of extensive theoretical, computational and experimental research. Disruptions are generally preceded by the growth of internal plasma instabilities in the vicinity of "resonant" surfaces where the magnetic field lines close on themselves after a small number of transits around the torus. The magnetic configuration then changes slightly,

because the magnetic field lines "reconnect" to form thin "magnetic islands" at the resonant surfaces. The energy source for the underlying instabilities can be the magnetic energy in the (poloidal) component of the magnetic field that is created by the plasma current itself, or the thermal energy of the plasma, or a combination of the two. When the instabilities are sufficiently strong and arise over a sufficient fraction of the plasma radius, the phenomenon of "resonance overlap" will occur, and magnetic islands can begin to fill most or all of the plasma cross section. At this point, global "magnetic reconnection" has occurred, and the thermal energy can be transported directly along field lines out of the plasma. Magnetic reconnection is a pervasive phenomenon in plasma physics: it occurs in astrophysical and space plasmas (for example, in "solar flares" of the sun's corona and in the earth's magnetosphere), as well as in laboratory experiments, and it remains an active area of contemporary plasma research [14].

	C-Mod	JET	ITER
Plasma thermal energy (megajoules)	0.2	15	1,000
"Specific energy" (megajoules per square meter)	0.03	0.1	1.0
Thermal quench time in fast disruptions (milliseconds)	~1	1-10	1-50
Maximum halo current in VDEs (megamperes)	0.4	0.8	10
Available volt-seconds	3	15	80
Potential for runaway avalanche	No	No	Yes

Table 4: Disruption-relevant parameters in present-day tokamaks and ITER

Disruption-relevant parameters in ITER are compared with those in JET (a large tokamak where severe disruptions have already caused significant machine damage) and those in Alcator C-Mod (a moderate-size tokamak which

is used extensively for disruption studies in the US), in Table 4. Although disruptions in ITER will have the same underlying causes as those in present-day tokamak experiments, the relevant scalings [15] imply that their consequences will be different in three important respects.

First, we see that the plasma thermal energy is almost a hundred times larger in ITER than in JET and several thousand times larger in ITER than in C-Mod. More importantly, the "specific energy", i.e., the plasma thermal energy divided by the surface area of the plasma, which provides a measure of the severity of the plasma-wall interaction in disruptions, is about ten times larger in ITER than in JET, and about thirty times larger in ITER than in C-Mod. Although the time-scale for energy deposition in the rapid "thermal quench" phase of a disruption will be somewhat longer in ITER than in present-day tokamaks (see Table 4), it will still be short compared with time-scales for heat transfer through the material of plasma-facing components. Thus, although "worst case" disruptions in present-day tokamaks can sometimes produce melting or sublimation of material surfaces, the thermal energy deposited by disruptions on plasma-facing components in ITER could be sufficient to cause significant ablation of surface material, resulting in a substantial influx of impurities into the plasma. This combination of ablation and impurity-influx, which is expected to dominate the plasma-surface interaction in ITER's disruptions, is not energetically possible in present-day experiments: it will be seen first on ITER itself.

Second, vertical displacement events (VDEs) will produce much larger forces on vessel components in ITER than in present-day tokamaks. As the plasma column moves downward during a VDE, a substantial fraction of the plasma current is transferred to a "halo" region above the main plasma; since the halo intersects the vessel wall, the return path for this "halo current" passes through conducting components of the vessel structures. Time-dependent computations of the magnetohydrodynamical (MHD) evolution of a VDE in ITER typically include a full two-dimensional plasma simulation together with the electromagnetic effects of the vessel and conducting structures. Since the return current will flow through conducting vessel structures primarily perpendicular to the main magnetic field, it will exert a large mechanical force on these structures. The ITER design allows for halo currents that are approximately the same fraction of the total plasma current as is observed in present-day experiments (see Table 4). In addition, as the plasma's outer

layers are "scraped-off" by the vessel wall during the downward plasma movement in a VDE, another plasma-physical effect enters when the shrinking current-carrying plasma core becomes vulnerable to MHD instability in the form of toroidal "kinking" of the plasma column, and this is also taken into account in calculating the mechanical load on surrounding conducting structures in ITER. (The "kink" instability is another example of magnetic reconnection in a plasma [14] -- in this case involving the appearance of magnetic islands with a helical structure just outside the current-carrying plasma core). However, the higher specific energy in ITER may produce a more conductive halo region, which would inhibit magnetic reconnection and kinking, or the impurity influx from surface ablation could have exactly the opposite effect. Although present-day experiments will continue to provide essential design guidelines for disruption effects in ITER, the full interactive dynamics of VDEs at ITER-like plasma-physical parameters can be explored only on ITER itself.

Third, the rapid decay of the plasma current in a disruption makes available a much larger number of "volt-seconds" in ITER than in present-day tokamaks. (According to the laws of electromagnetism, the volt-seconds -- the product of the voltage which appears in the toroidal direction around the plasma and the duration in seconds which this voltage lasts -- is proportional to the magnetic flux associated with the decaying plasma current.) Since a plasma electron making a toroidal transit of a tokamak will gain an amount of energy essentially equal to the toroidal voltage, the number of available volt-seconds provides a measure of the total energy which can be imparted to an electron during the current decay phase of a disruption. The number of volt-seconds available in ITER is several times greater than in the largest present-day tokamak (see Table 4) and is sufficient to produce a physical effect not yet encountered in tokamak research, namely an exponential "avalanche" in the number of relativistic "runaway" electrons by "knock-on" electron-electron collisions [16]. Indeed, in ITER there is a potential for large-scale plasma-to-runaway current conversion [17], which would change the characteristics of the current-decay phase of all types of disruptions and could introduce severe heat-load problems on those parts of the vessel wall on which the runaway electrons finally impinge. No present-day tokamak can access this phenomenon: it will appear (if at all) first on ITER.

3. Physics of a High-Power-Flux Radiative Divertor

The power incident on "plasma-facing" components in ITER will be several times larger than in present-day experiments. Moreover, the plasma pulse length and the cumulative experimental run-time will be between a hundred and a thousand times longer in ITER. The solutions that are being developed for the problems posed by these requirements in ITER involve new physical concepts and innovative techniques, which go beyond what is needed for the successful operation of present-day experiments. This development is highly multi-disciplinary, involving plasma physics, atomic and molecular physics, computational physics, surface physics of the plasma-wall interactions, and materials science. Multi-dimensional computational models are being used to provide the extrapolation from present experiments to ITER. Although the techniques employed for power handling in the present ITER design are based on well-validated scalings and models, the design also incorporates considerable flexibility to accommodate unforeseen effects.

The high power and particle fluxes in ITER are directed away from the main plasma by magnetically "diverting" the outer layers of the plasma to a "divertor chamber". This has the advantage of moving the plasma-material interaction away from the vessel wall surrounding the main plasma, but it tends to concentrate the power on a relatively small area within the "divertor". The divertor in ITER is placed at the bottom of the main plasma vessel.

Parameters relevant to power handling in present-day tokamaks and in ITER are compared in Table 5. In present experiments, the power and particle loads are sufficiently small that they can be handled by making the "divertor plates" (on which the diverted field lines impinge) of materials such as graphite, and cooling these divertor plates between discharges. The longer pulse lengths for ITER require the use of active cooling during the plasma discharge.

In ITER, however, unless the power reaching the divertor plates is reduced, the maximum peak power loads are too large. It is therefore planned to reduce the peak heat fluxes in the divertor by using impurity radiation to transfer most of the power from the plasma to the walls, especially the walls of the divertor chamber, thus spreading it out over a much larger surface area [18,19]. For this purpose, impurity ions are deliberately introduced into the plasma in the divertor chamber; the impurity species is chosen so that the ions will radiate copiously (mainly by "line excitation" and "recombination" radiation) at the low plasma temperatures in the divertor, while causing minimum

degradation in the performance of the main plasma. While some of the physics involved in this technique is being tested in present experiments, the conditions required to achieve this "radiative divertor" regime with much larger volumes and with higher power fluxes can only be realized in ITER itself.

	Present-day large tokamak	**ITER**
Plasma heating power (megawatts)	40	400
Plasma surface area (square meters)	130	1,100
Plasma pulse length (seconds)	10	> 1,000
Cumulative run-time (seconds)	$< 5 \times 10^4$	3×10^7
Peak power load at divertor without radiation (megawatts/square meter)	10	30
Peak power load on divertor plate with radiation (megawatts/square meter)	5 - 10	5 - 10
Ratio of hydrogen neutral mean-free-path to plasma radius	0.1	0.001 - 0.03
Ratio of absorption length for Lyman-alpha radiation to plasma radius	0.01	0.0001 - 0.001

Table 5: Divertor-relevant parameters in present-day large tokamaks and ITER

The atomic physics of impurity behavior in the ITER divertor will be quite complex [20]. The impurity radiation emission rate will be proportional to the electron and impurity densities, but it will also have a complicated dependence on the electron temperature, the hydrogenic neutral density, and the impurity "recycling time" -- the characteristic time for impurities to complete one injection/exhaust "cycle". All of these quantities strongly depend on the power density and the size of the divertor. With higher power densities, the plasma

and neutral densities in the divertor will be larger in ITER than in present-day experiments, probably much larger. The ratio of a hydrogenic neutral's mean-free-path to the plasma radius, both for ionization and charge-exchange with the plasma and for neutral-neutral collisions in the divertor, will be as much as a hundred times smaller in ITER (see Table 5). This will make the transport of neutral atoms more collisional and will likely increase the impurity recycling time, reducing the net impurity radiation emission rate.

All of the relevant effects -- plasma physics, atomic physics, and surface-physics -- are included in computer simulations of the radiation losses in ITER [19], which are used to determine the operational conditions necessary to reduce the peak heat loads on the divertor plates to acceptable levels. A typical simulation calculates self-consistently the plasma parameters in the ITER divertor and the radiation losses due to hydrogen, helium (from the D-T reactions), neon (the deliberately-introduced impurity), and carbon (sputtered from graphite divertor walls) for the standard ITER case of 200 megawatts of charged-particle power flowing out of the main plasma (e.g., 300 megawatts of alpha-particle heating, less 100 megawatts of power radiated from the main plasma). The radiation levels from the divertor region for such a case are sufficiently high that the peak heat fluxes on the divertor plates are well below the maximum permitted levels. Impurities in these quantities produce only a small reduction in fusion reactivity in the core of the plasma. In this sense, the present ITER divertor design is based on a conservative application of well-validated physical models.

Hydrogen radiation losses and the balance between hydrogen ionization and recombination are also key physics determinants of the plasma conditions in the divertor [20]. The mean-free-path for absorption of the dominant hydrogen ("Lyman alpha") radiation for very high neutral densities can be substantially smaller than the size of the divertor plasma in ITER, more so than in present experiments (see Table 5), which can both alter the ionization balance and reduce the hydrogen radiation losses. In ITER, the effect should be an important factor in determining the divertor plasma conditions, particularly the "reference" divertor operational regime in which volume recombination and hydrogen radiation losses are very important.

In addition to handling the high power flux and controlling the concentration of the radiating impurity species, the divertor in ITER must accomplish three other functions simultaneously: (i) control the plasma density,

while exhausting a fraction of the particle flux coming from the plasma chamber; (ii) remove helium from the plasma; and (iii) demonstrate acceptable net erosion of material surfaces. In present-day tokamaks, each of these functions can be carried out essentially independently of the others: the ITER divertor will, for the first time, bring them together.

The particle balance in a tokamak is determined by absorption and outgassing by the wall and by the tokamak fuelling and divertor pumping systems. The time-scale associated with adsorption and outgassing of the wall -- called the plasma-wall equilibration time in Section II.3 -- is in the range 100 - 1,000 seconds for standard materials; because of its long pulse, ITER will be able to investigate these effects with very high power and particle fluxes in a way that no present tokamak can even approach.

The divertor pumping systems will also be required to remove thermalized helium, which is the residue ("ash") of D-T reactions. The helium density is expected to be about 5 - 10% of the plasma density in the ITER core, but its density in the divertor will depend on transport effects in the edge and divertor plasmas. Achieving a sufficient helium density in the divertor, as well as high neutral gas pressure, will be important for efficient pumping of the helium. The removal of helium "ash" is an essential part of the physics of "burning plasmas" to be explored on ITER.

The longer pulse length and cumulative run-time, together with the high heat loads and more intense disruptions discussed in Section III.2, mean that the depth of material eroded from the plasma-facing materials by sputtering, chemical reactions, ablation and melt-layer loss can be at least a thousand times greater for ITER compared to present-day tokamaks. For the most severe assumptions, the erosion lifetime of the plasma-facing components is sufficiently short that several replacements will be required during the lifetime of ITER, as is allowed in the present design. In present experiments, by contrast, the net erosion is barely measurable. Thus, for the first time in fusion research, the ITER experimental program must address the physics of the erosion mechanisms and the physics of how the eroded material is transported and redeposited.

The handling of large plasma power and particle fluxes and the associated issues of intense plasma-materials interactions are generic to all approaches to magnetic confinement at the reactor level. Much of what will be learned from ITER's divertor experiments will be directly applicable to

"alternate" confinement concepts, especially those utilizing a toroidal magnetic configuration.

In recognition of the uncertainties surrounding the operation of a high-power-flux radiative divertor, the ITER divertor is being designed to be very versatile, with components mounted onto removable cassettes so that they can be reasonably easily replaced. Studies of the plasma physics, atomic physics, plasma-surface interactions and material science of the divertor will be among the most exciting and challenging elements in ITER's experimental research program.

IV. FULLY-INTERNATIONAL COLLABORATIVE CONDUCT OF RESEARCH

Presently, the technical work of the ITER EDA is being carried out at three "joint work sites" (located in Europe, Japan and the United States), linked to each other and to many other participating institutions by a high-speed computer network. Since the professional staff at each of the three sites is drawn approximately equally from all four ITER partners, the ITER EDA has proved conclusively that scientists and engineers from many different countries, cultures and scientific heritages can be melded together into a coherent and effective project team.

Perhaps even more remarkable is the level of scientific collaboration that has been achieved through the ITER process among the research groups working on tokamak experiments and related studies in the world's fusion laboratories. Working through "expert groups" drawn from the experimental and theoretical research programs of the four partners, the ITER process has involved the international coordination and prioritization of relevant physics research, the joint planning of experiments designed to address specific ITER physics issues, and the sharing of databases embodying the results of this and other ITER-relevant research.

In ITER's operational phase, the level of international collaboration will be extended even further and will include a fully-international central research team at the ITER site, together with comparable off-site research teams conducting experiments via remote control rooms at major fusion laboratories in each of the partners. It is expected that the use of high-speed computer networks will allow these remote control rooms to be essentially equivalent to the main experimental control room at the ITER site, at least to the extent

needed for carrying out most experimental studies. In addition, the experimental program will be planned jointly by the partners, and the experimental data will be analyzed jointly, often at remote sites.

Thus far, the key to ITER's success as an international undertaking has been that all four partners have been equally involved in major technical and management decisions from the outset; to the extent possible, they have also shared equally in the technical responsibilities and in the provision of resources. If a mutually acceptable framework for construction and operation can be found, it seems certain that the experimental phase of ITER will provide a further paradigm for international collaboration in science.

ACKNOWLEDGEMENTS

The writer has benefited greatly from contributions from ITER Joint Central Team (JCT) members F. W. Perkins (Sec. III.1), D. E. Post (Sec. III.3) and J. Wesley (Sec. III.2). Other valuable comments have come from D. Boucher, A. E. Costley, ITER JCT; C. Z. Cheng, G. W. Hammett, S. C. Jardin, N. R. Sauthoff, PPPL; H. L. Berk, J. Van Dam, Univ. Texas; R. O. Sayer, ORNL; D. E. Baldwin, R. E. Waltz, GA; H. H. Fleischmann, Cornell; W. Marton, DOE; and members of the ITER Steering Committee - US. This work was supported by U.S. Department of Energy contract DE-AC02-76-CHO3073.

REFERENCES

[1] ITER Engineering Design Activities (EDA) Agreement, July 1992.

[2] R. Nazikian, et al., paper CN-64/A2-4, Sixteenth IAEA Fusion Energy Conference, Montreal, Canada, October 7-11, 1996.

[3] C. Z. Cheng, et al., paper CN-64/FP-23, Sixteenth IAEA Fusion Energy Conference, Montreal, Canada, October 7-11, 1996.

[4] H. L. Berk, et al., paper CN-64/D2-5, Sixteenth IAEA Fusion Energy Conference, Montreal, Canada, October 7-11, 1996.

[5] D. Boucher, et al., paper CN-64/FP-22, Sixteenth IAEA Fusion Energy Conference, Montreal, Canada, October 7-11, 1996.

[6] K. Ushigusa for the JT-60U Team, paper CN-64/01-3, Sixteenth IAEA Fusion Energy Conference, Montreal, Canada, October 7-11, 1996.

[7] C. C. Petty, et al., Physics of Plasmas 2, p 2342 (1995).

[8] J. G. Cordey for the JET Team, paper CN-64/AP1-2, Sixteenth IAEA Fusion Energy Conference, Montreal, Canada, October 7-11, 1996.

[9] F. W. Perkins, paper CN-64/FP-24, Sixteenth IAEA Fusion Energy Conference, Montreal, Canada, October 7-11, 1996.

[10] M. Kotschenreuther, et al., paper CN-64/D1-5, Sixteenth IAEA Fusion Energy Conference, Montreal, Canada, October 7-11, 1996.

[11] K. H. Burrell, Review Paper, "E x B and Magnetic Shear Effects on Turbulence and Transport in Magnetic Confinement," Annual Meeting of the Division of Plasma Physics, American Physical Society, Denver, Colorado, November 11-15, 1996.

[12] P. H. Diamond, et al., Physical Review Letters 72, p. 2565 (1994).

[13] R. E. Waltz, et al., Physics of Plasmas 1 (7), p. 2229 (1994).

[14] "Symposium on Magnetic Reconnection in the Laboratory and in Space", Annual Meeting of the Division of Plasma Physics, American Physical Society, Denver, Colorado, November 11-15, 1996.

[15] J. Wesley, et al., paper CN-64/FP-25, Sixteenth IAEA Fusion Energy Conference, Montreal, Canada, October 7-11, 1996.

[16] R. Jayakuma, H. H. Fleischmann and S. J. Zweben, Physics Letters A 172, p. 447 (1993); H. H. Fleischmann and S. J. Zweben, Princeton Plasma Physics Laboratory Report PPPL-2914 (1993).

[17] M. N. Rosenbluth, et al., paper CN-64/FP-26, Sixteenth IAEA Fusion Energy Conference, Montreal, Canada, October 7-11, 1996.

[18] G. Janeschitz, et al., Journal of Nuclear Materials 220-222, p.73 (1995).

[19] A. Kukushkin, H. D. Pacher, et al., paper CN-64/FP-27, Sixteenth IAEA Fusion Energy Conference, Montreal, Canada, October 7-11, 1996.

[20] D. E. Post, Journal of Nuclear Materials 220-222, p. 143 (1995).