Scientific challenges and opportunities to studying burning plasmas

The ITER experiment will answer open research questions to enable the production of fusion energy.

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Burning plasmas, in which the energy from charged particles created by fusion reactions compensates for heat loss from the plasma, power the Sun , other stars, and could provide abundant energy for humankind. For fusion in the Sun, proton fuel is used for a successful fusion reaction. But on Earth the highest performance fuel option is composed of deuterium and tritium ions . There are different ways of achieving fusion energy in the laboratory, most notably magnetic and inertial confinement.

Magnetic fusion has made major progress, culminating in the planning and construction of the ITER experiment. Significant fusion power up to 10 and 16 MW for a little less than a second has been achieved in magnetically confined plasmas in the Tokamak Fusion Test Reactor (TFTR) in the US [1] and in the Joint European Torus (JET) in the UK [2], respectively. In those fusion reaction experiments, alpha-particles self-heat the deuterium-tritium fuel, which is a modest fraction (<13%) of the total plasma heating. (See also the article by David Pace, Bill Heidbrink, and Michael Van Zeeland, Physics Today, October 2015, page 34.). Whereas the original ITER design was based extensively on empirical results, designs for a tokamak power plant such as those performed by EUROfusion in Europe and in other countries increasingly utilize theoretical and computational research coupled with results from ITER.

The construction of the international fusion research facility, ITER, in France will provide the opportunity to study burning plasmas in which at least two thirds of the total heating is from fusion reactions, producing 500 MW of fusion power for more than 300 seconds. The facility also will enable the first in-depth study of burning plasmas in a magnetic confinement configuration.[3] The promising possibilities of ITER will make it one of the largest scientific experiments ever undertaken, with participation by China, the European Union, India, Japan, Russia, South Korea, and the US. The high cost of the project and the advantage in sharing the worldwide community's scientific and technical knowledge has motivated the breadth of the international collaboration. Although project costs increased and its construction schedule was delayed early on, ITER has been successfully adhering to the current schedule in recent years after a major reorganization. It aims to begin experiments with hydrogen in the mid 2020s and to perform burning plasma experiments in the mid 2030s.

ITER will create an axisymmetric toroidal plasma, whose cross-sectional view is shown in figure 1. For deuterium–tritium plasmas with ion temperatures in the range of 10-20 keV, the fusion power density is approximately proportional to the ion pressure squared. The core of the plasma has the highest plasma pressure and reactivity. The temperature there is determined by the balance of plasma heating and heat loss from the core to the edge of the plasma. The heating balance is mostly affected by small turbulent fluctuations in the density on the order of a few percent or less in amplitude. During the initial phase of the project before the use of deuterium-tritium fuel, the plasma will be mostly heated by an external source.

Once the plasma pressure is sufficiently large in a plasma fueled with deuterium and tritium, the self-heated, or burning, phase begins. During that phase, fusion reactions generate suprathermal alpha particles, which are confined by the magnetic field and heat the plasma as collisions with the background plasma slow them down. The alpha particles self-heat the plasma by about twice that of any external heating source. Eighty percent of the fusion power is released as energetic neutrons, which do not interact with the plasma. Their energy is transferred as heat to the surrounding blanket which would then convert the heat to electricity in power plants. Twisting magnetic fields that lie on nested toroidal magnetic flux surfaces confine the plasma in the core. But outside of the last closed-flux surface called the separatrix, the magnetic field lines are no longer closed and intersect the first wall components, preferentially in the so-called divertor region seen in figure 1. The separatrix is formed by the plasma field current interacting with the current that's carried by the large superconducting divertor coil below the plasma chamber. The large heat flux from the plasma core impinges on the divertor surface and needs to be carefully controlled.

Producing a burning plasma requires the simultaneous achievement of plasma parameters that are different from those in current experiments. Similar to fluid dynamics studies, dimensionless parameters for the core plasma can be identified based on theoretical arguments that distinguish different physics regimes and scaling the plasma's behavior from one set of experiments to another.[4] ITER will have to simultaneously achieve several conditions. Large values of system size will be required, which is related to how many ion Larmor radii, r_L ,fit along the plasma minor radius a, $(1/\rho^* = a/r_L)$. The frequency of plasma collisions will need to be low, such that the mean free path between Coulomb collisions is much longer than the characteristic distance around the plasma along magnetic field lines. Finally, ITER will also need to produce modest values of the plasma pressure p that is normalized by the magnetic field pressure B, $\beta \propto p/B^2$.

In addition, another parameter has been found to describe various limitations to the plasma density n_e achieved in experiments. The so-called Greenwald density n_e^{GW} is proportional to the plasma current and inversely proportional to the square of the plasma minor radius. Although it does not come from idealized equations often used for plasma dynamics, it successfully describes the upper limits of the density in the region near the edge but where closed magnetic lines exist, the beginning of H-mode confinement degradation at high separatrix density, and the onset of magnetohydrodynamic instabilities. The instabilities may arise from atomic-physics effects in the edge region and electro-magnetic radiation from impurities, which are not captured by the above mentioned dimensionless plasma physics model.

Present day experiments can achieve the parameter values typical of ITER individually but not simultaneously. ITER, therefore, will allow scientists to study burning plasmas and the unique dimensionless parameter regime that's important for ITER and other deuterium–tritium, magnetic fusion facilities that will produce electricity from burning plasmas.[5]

Transporting heat and particles in the plasma core

ITER was designed mainly using empirical projections of the confinement of heat in the plasma based on experimental results from around the world. The results come from a regime in which there is reduced turbulent transport in the edge region. [5] The so-called H-, or high-confinement, mode[6] is a consequence of sheared plasma flow that decorrelates turbulent eddies. They generate a narrow zone of steep radial gradients in the edge that corresponds to the pedestal region shown in figure 2.

Since the beginning of the ITER facility's design, there has been substantial progress in understanding the turbulent processes in the core and comparing detailed experimental results with sophisticated theoretical models. One such model has shown that an experimental heat flux result is not simply proportional to the temperature gradient but can be highly nonlinear. The heat flux strongly increases above a threshold value for the normalized temperature gradient R/L_T, where R is the major radius of the torus and the gradient scale $1/L_T=|\nabla T|/T$. The value R determines the curvature of the magnetic field lines, which is responsible for the micro-instabilities that give rise to strong turbulent transport once the threshold value of R/L_T is exceeded. [10] Experimentally, the occurrence of strong turbulent transport implies that R/L_T is relatively constant for parameters of interest. It also means that the shape of the radial temperature profiles $T(r) \sim C\ exp(-r/L_T)$ is approximately self-similar to the plasma edge at the top of the pedestal. These arguments are supported by theoretical calculations predict both the critical normalized temperature gradient for the increase in heat flux and the dependence of the heat flux on the gradient. The latter is also affected by additional plasma parameters, including the normalized gyro-radius.
ITER is predicted to be where these turbulent processes are important, so the temperature in the plasma core is approximately proportional to the temperature at the top of the pedestal. The sensitivity of the predicted fusion power to the pedestal temperature is illustrated in figure 3. Whereas the transport model predictions of power amplification Q versus pedestal temperature T_{ped} were based on quite sophisticated turbulence simulations using some of the largest supercomputers a decade ago, there were some simplifications. Higher performance is expected from estimates that include various stabilizing effects—density peaking, magnetic components of turbulence, and beam-driven rotation—that were neglected in figure 3.

The modelling of the nonlinear, turbulent plasma state has become increasingly more sophisticated. Now it simultaneously considers turbulence at both the ion- and electron-scale lengths, the role of shear in the profile of plasma rotation, and the effect of fast ions, which affects the relationship of the heat flux to the normalized temperature gradient. [11] Experiments have shown regimes of operation in which the heat flux may decrease because of an increase in the plasma rotation or the presence of fast ions. Experiments show that the rotation in the plasma arises from a combination of externally applied torque from the injection of neutral beams and from turbulence-induced processes, a phenomenon known as Reynold's stress in fluid dynamics. So the same turbulence that mainly determines the loss of heat can enhance the plasma rotation and play a major role in determining the saturation level of turbulent transport. Even for plasmas with no applied torque, significant rotation is observed. Whereas the applied torque for ITER is smaller than in current experiments, it may still drive enough rotation to improve performance, and additional rotation may be driven by turbulent processes.

Concerning particle transport, experiments and theory predict that turbulent plasma processes will result in a peaking of the density profile in low-collisionality discharges. [12] That is advantageous for fusion reactivity because turbulence result in peaking of the density profile in the high temperature part of the plasma. In present day experiments, low collisionality is only achieved at low normalized density. ITER and future power-producing experiments, however, will operate at high normalized density. Such experiments will therefore have an important role in validating the theoretical models in new regimes where the fast ion effects are due to alpha particles operating at very low collisionality and large values of system size. In addition to transporting the main fuel species, transporting the alpha-particle ash from the deuterium–tritium reactions to the edge and subsequent pumping of the ash is important to avoid accumulating impurities and diluting the fuel. Turbulent processes are expected to dominate collisional processes and will help determine the fuel dilution and the presence of impurities in the core from plasma material interaction at the edge.

Pedestal performance

The reduced transport in the edge region associated with H-mode discharges results in steep temperature and pressure gradients, as shown in figure 2, and has many ramifications in addition to the confinement in the core. The steep pressure gradient in the edge region generates parallel currents through a mechanism similar to thermoelectric-driven currents. Large pressure and current gradients affect the magnetohydrodynamic (MHD) stability of the edge region and can trigger edge instabilities, so-called edge-localized modes[13] (ELMs)] that can be seen in figure 4. The resulting operating boundaries due to ELMs are relatively well understood by MHD models that simulate the stability of the edge.

Whereas the global pedestal stability is described well by linear, ideal MHD modelling, the parameters at the top of the pedestal are determined by a combination of MHD stability and the transport mechanisms between the edge of the plasma and the top of the pedestal. A model (EPED) that successfully predicts the width and height of the pedestal pressure combines linear MHD stability analysis of the pedestal with a simplified assumption about the transport in the region.[9] However recent experiments using tungsten as a wall material in the Axially Symmetric Divertor Experiment Upgrade (ASDEX Upgrade) and the Joint European Torus (JET) tokamaks have shown a degradation that could not be explained by the MHD model. Since the temperature and density profiles can respond differently partly because of an influx of recycled neutral particles from the wall into the region between the plasma edge and the pedestal top. Those effects mean

that additional transport physics will have to be incorporated into the EPED model to improve predictions.
ITER will be operating in a different system-size regime, so the role of neutrals may change. Such a combination of low collisonality and large system size may alter the turbulence characteristics in the pedestal region and affect the height of the pedestal. As the first device to combine a pedestal with low collisionality parameters and operate at high densities relative to the Greenwald density, ITER provides a unique opportunity to validate the understanding of the edge pedestal.

For ITER, the large ELM instabilities could damage divertor components. To mitigate those instabilities, techniques have been developed to suppress them, such as by operating within the stability boundary using additional coils that give rise to small 3D perturbations of the tokamak axisymmetry on the order of $\tilde{B}/B \sim 10^{-4}$, where B is the magnetic field.

Plasma-boundary interactions

A major challenge for ITER will be controlling the heat and particle exhaust. The sharp pressure gradients in the edge region extend to just beyond the last closed-flux surface from where heat and particles are rapidly transported along the open magnetic field lines to the divertor plates. Recent experiments on a variety of devices have shown that the heat flux width at the outboard midplane (where the plasma has its largest radial extension) is narrow and would extrapolate to as low as 1–2 mm in ITER. Advances in transport modeling across the magnetic field on the open field lines describe the heat flux width in current experiments reasonably well. When the same model is applied to ITER it indicates that turbulence changes in the edge may actually increase the heat flux width to up to 5 mm. That prediction requires stringent experimental tests that may only be possible using ITER.

To illustrate the magnitude of the heat-exhaust challenge, consider that the heat flux into the ITER boundary will be 150 MW if the power lost by radiation is ignored. Taking into account the extrapolated experimental heat flux width range, that heat exhaust would far exceed the approximately 10 MW/m^2 power-handling capability of steady-state, high-heat flux components with tungsten armor that will be used in the ITER divertor.

Two approaches are used to reduce the heat flux on the divertor plates. The first reduces the incident angle of the magnetic field lines carrying the heat flux by expanding the magnetic flux surfaces in the vicinity of the divertor plates and tailoring the angle of the plates so that they are nearly tangential to the magnetic field. That approach demands accurate alignment of individual target plates to ensure that the leading edges do not overheat. The heat flux on the target would be substantially reduced to about 40 MW/m^2 in the ITER geometry.[15] The decrease depends on the heat-flux-width assumptions, but the value is comparable to the heat flux on the surface of the Sun, which amounts to 60 MW/m^2.

The second approach dissipates part of the heat flux using seed impurities intentionally injected into the divertor plasma. The impurities will emit line radiation to distribute the heat load homogeneously in all directions. At constant-impurity concentration, the radiation losses increase with the square of the electron density. A high density at the separatrix is also required for effective power dissipation. Both techniques have been successfully demonstrated for various experiments. Figure 5 demonstrates how the radiation losses from the divertor and X-point regions can be enhanced by adding nitrogen.

The role of ITER is to test ideas to stably exhaust power and particles in a future fusion power plant at relatively high separatrix densities while simultaneously optimizing the fusion power. Among the questions to be addressed are: Will confinement be degraded because of the high density in the scrape-off plasma as seen in present day experiments? And will the fuel concentration in the core be decreased by impurity ions penetrating the core from the divertor plasma, which reduces the plasma reactivity?

Control of Burning Plasmas

Perhaps the most important question is whether the high-energy alpha particles that sustainably heat the burning plasma deposit their energy to the ions and electrons from Coulomb collisions or whether their interaction with electromagnetic waves results in a spatial redistribution or loss of alpha particles.[2] Deuterium–tritium experiments on both JET and the TFTR found initial indications of heating by the alpha particles generated from the fusion reactions (alpha-heating). The slowing down of the 3.5MeV alpha particles to energies comparable to the core ion temperature is in good agreement with classical calculations in plasma conditions that are most similar to what's planned for ITER's first set of high fusion power experiments.

However, in those experiments, the ratio of intrinsic alpha-heating to external heating was small, about 10% or less. ITER will be the first device to conduct experiments in which the alpha-heating dominates the heating power, i.e. that ratio will be large, and can hence address the nonlinear wave–particle interaction to decrease the uncertainty associated with the earlier low-power alpha-heating experiments. ITER will be able to test whether electromagnetic waves generated by the energetic alpha particles are more unstable than in current experiments due to the large system size. Those results will enable researchers to determine optimal operating conditions and minimize or even avoid such instabilities.

Controlling fusion power in a burning plasma would appear to be straightforward. The fusion reactivity in ITER decreases with increasing temperature, and the confinement time of the plasma energy decreases with heating power, which enables a stable operating point. The problem becomes more interesting when the consequences of changes in thermal confinement are considered. In future power plants, a 10% increase in confinement can roughly double the fusion power. The increased fusion power could be feedback control because it occurs over the timescale of several energy confinement times. Large concentrations of fast ions, can both reduce the plasma turbulence enhancing the plasma confinement and also decrease the efficiency of the alpha-particle heating by interaction with electromagnetic waves that can lead to loss of the fast ions. Hence, there is significant uncertainty about how the fast ions will affect the burn-control dynamics and whether the system will respond nonlinearly. Only ITER, with its dominant self-heating by alpha particles, will enable a comprehensive exploration of burn control.

Researchers do know that burn control is affected by additional operational constraints related to the maximum density, the MHD stability of the plasma, which is strongly related to the plasma

pressure, and the heat flux to the plasma-facing components. Consider the effect of increasing the plasma density: In principle, it's advantageous because it increases plasma reactivity and the power radiated. However as the density approaches its maximum, observations show that the confinement time degrades. Experimental physicists are using theoretical models to help fine tune the experimental conditions to optimize the plasma's performance and prevent it from touching the operational boundaries. Sophisticated plasma control systems are used to control various actuators, such as fueling by gas valves, frozen deuterium pellets, and the auxiliary heating power.
For ITER, the simulations used to predict the discharge performance will need to take into account the alpha-heating dynamics and the plasma control systems to avoid breaching an operational boundary. From a control perspective, the challenge is that the control actuators, such as for auxiliary heating, have a smaller impact on burning plasmas than in present experiments since alpha heating will play a larger role in the power balance.[17]

Demonstrating control of a burning plasma in ITER for both inductive and non-inductive regimes is essential before extrapolating to a power plant. There the required power amplification is expected to be higher, Q = 20-50, than ITER's Q = 10 and Q = 5 in a fully non-inductive, steady-state operation. For a non-inductive, steady-state tokamak power plant, a large fraction of the current would be self-driven by temperature and density gradients, such as the thermoelectric bootstrap effect. As alpha heating will be the dominant heating mechanism that determines the temperature gradient, controlling the heat becomes more challenging.

Touching an operational boundary can lead to the occurrence of MHD instabilities that can terminate the plasma discharge. In present experiments, the worst plasma confinement disruption occurs when a substantial plasma current of the order of 1 MA is lost rapidly. on a timescale of several milliseconds. In ITER, 15 MA of plasma current could be rapidly terminated over a somewhat longer time. A sudden termination of the plasma discharge generates large toroidal electric fields that can drive energetic runaway electron currents, transmit heat fluxes to the plasma-facing components, and apply large forces to the vacuum vessel and its components. Whereas mitigation systems are being implemented to ameliorate the disruptions if the operating boundaries are exceeded, active control systems will also be used to ensure that the plasma operates within safe operating boundaries. Furthermore, active control of MHD instabilities, including by driving an additional current in the plasma by the injection of radio-frequency waves or by applying three-dimensional magnetic field perturbations, has made a lot of progress. Research has demonstrated that a localized current of a few percent of the total plasma current through the injection of RF waves can be sufficient to remove magnetic islands that are responsible for some disruptions and will likely be used in ITER.

Experiments with the forthcoming ITER will provide a unique opportunity to study burning plasmas, validate outstanding predictions, and develop the tools needed to better understand burning plasmas in the future. The experiments will provide seminal answers to issues that are central to the prospects for fusion. ITER will be a major step in bridging the gap between current understanding and the knowledge needed to design and operate fusion power plants for a safe, sustainable energy source of the future.

Acknowledgement

The authors want to thank R. Goldston, G. Hammett, A. Kallenbach and T. Luce for their thorough review and for their valuable comments and suggestions. This work was supported by the U.S. DOE Award No. DE-AC02-09CH11466.

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Figure captions

Figure 1: The poloidal cross-section of ITER shows a magnetic separatrix that's introduced by creating a so-called x-point. Closed magnetic surfaces exist inside the separatrix; outside the plasma flows in the narrow, few-mm-wide scrape-off layer towards the divertor. The dashed line indicates the dimension of the major radius R measured from the torus' symmetry line (dash-dotted line) to the plasma center, and the dashed line defines the minor radius. (Credit © ITER Organization.)

Figure 2: A typical ion temperature profile, plotted versus normalized minor radius of the confined plasma, features a pedestal region of steep gradients at the plasma edge (example from [7]). In ITER the expected temperature values will be about five times higher. (Adapted from ref. 7.)

Figure 3: The fusion power P_{fus} and power amplification Q for ITER depend strongly on the ion temperature at the top of the edge pedestal shown in figure 2. (adapted from [8]). The box labeled EPED shows the range of pedestal temperature T_{ped} predictions from a leading model [9].

Figure 4: An edge-localized mode (ELM) instability appears on this fast camera image from the Mega Ampere Spherical tokamak. The wide-angle view shows the whole plasma, and the enhanced D_a line emission clearly shows plasma filaments, which are ejected from the edge by the ELM instability .[14] (Courtesy of Andrew Kirk)

Figure 5: Two tomographic reconstructions of impurity radiation from the divertor and X-point regions show clear differences in their enhanced radiation losses. For an experiment of the ASDEX Upgrade tokamak without additional impurities (left), about 50% of the heating power is dissipated by electromagnetic radiation. But when nitrogen seeding is added (right), a zone of high radiation occurs at the X-point, and the radiated power fraction rises to values in excess of 80%. The size scale in the ASDEX Upgrade is quite different compared to that in ITER (see the scale at the lower left). (Adapted from ref. 16.)