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Wilson, Howard orcid.org/0000-0003-3333-7470 (2019) The impact of plasma physics on the timescale to a tokamak fusion power plant. Philosophical Transactions of the Royal Society A: Mathematical, Physical and Engineering Sciences. 20170435. ISSN 1471-2962

https://doi.org/10.1098/rsta.2017.0435

The Impact of Plasma Physics on the Timescale to a Tokamak Fusion Power Plant

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Abstract Some of the main plasma physics challenges associated with achieving the conditions for commercial fusion power in tokamaks are reviewed. The confinement quality is considered to be a key factor, having an impact on the size of the reactor and exhaust power that has to be managed. Plasma eruptions can cause excessive erosion if not mitigated, with implications for maintenance and availability. Disruptions are a major concern – one large disruption could terminally damage the reactor so it is important to understand the loads they impart to the structure, and put in place appropriate protection and an effective avoidance/mitigation strategy. Managing the exhaust of heat and particles from the plasma is likely to be a significant issue, which may be mitigated if an advanced confinement regime can be identified. The advanced divertor structures that may be required to handle the exhaust have a significant impact on the design of a fusion reactor. Three strategies can be identified to take account of the physics challenges, with different implications for the timescale to fusion power: (1) a staged approach with the size of each step determined by our confidence in the predictive capability of our models; (2) a single, big step with contingency built into the design where possible to accommodate the uncertainty in physics predictions, and (3) a single big step with optimistic physics assumptions and no contingency, accepting the increased financial and reputational risk that comes with such an approach.

1. Introduction

There is a range of issues that influence the timescale to fusion power, including identifying appropriate materials for the challenging environment; solutions to difficult engineering and technology questions, and optimising the plasma for fusion power while respecting engineering limits. In practice these cannot be treated in isolation – if an attractive plasma solution can be found, that might ease the technological challenges; if a new technology or material is identified that might ease the plasma constraints. An integrated approach to the development of fusion power is therefore key. In this paper we focus on the tokamak and some of the plasma physics challenges that need to be addressed as part of this integration which, if not resolved, would delay fusion power to the grid.

In the tokamak approach to fusion power, the deuterium (D) and tritium (T) fuel mix is confined in plasma state by a toroidal geometry of magnetic field, which has two components (see Fig 1). The toroidal component is typically the dominant one. It is created by a number of current-carrying toroidal field coils surrounding the DT plasma to provide the main guide-field that defines the toroidal geometry about the axis of symmetry. A purely toroidal magnetic field is insufficient to confine the plasma because of the particle drifts associated with the inhomogeneous and curved magnetic field. The effect of these drifts can be nullified by imposing a poloidal component to the magnetic field as shown in Fig 1. This poloidal component is created by passing a toroidal current through the plasma. Finally, the shape of the plasma cross section can be influenced by a set of current-carrying poloidal field coils which are centred on the axis of symmetry.

The aim of this paper is not to provide a thorough, in-depth review of the rich variety of tokamak plasma physics – that would require a complete volume – but rather to allow an

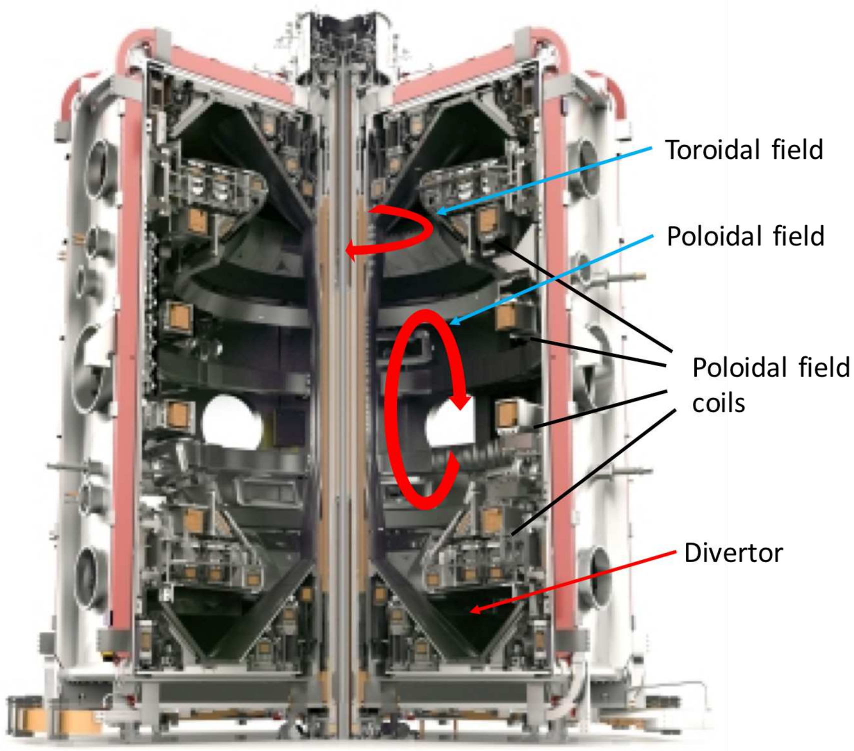

Figure 1: Cross section of the MAST Upgrade tokamak showing the directions of the poloidal and toroidal components of the magnetic field, together with the poloidal field coils and the divertor

assessment of the impact of plasma physics issues on the timescale to fusion. Thus we have placed an emphasis on describing some key physics phenomena, their consequences for fusion power and our ability to quantify predictions to guide reactor design and construction. For a more complete description of the physics, and a more complete set of references, we refer the interested reader to the ITER Physics Basis document [1].

We begin in the following section by considering the impact of energetic particles created by the plasma heating mechanisms. Section 3 addresses the quality of the confinement, and how it is degraded by plasma turbulence; this has a direct impact on how much heating is required to achieve fusion conditions as well as the size of the reactor core. We describe how the plasma can bifurcate from a low confinement turbulent state, called L-mode, to a high confinement state with suppressed turbulence near the plasma edge, called H-mode. Steep pressure gradients form in the region where turbulence is suppressed, and this can provide the free energy to trigger a sequence of filamentary plasma eruptions called Edge Localised Modes, or ELMs. We discuss the consequences of these in Section 4. Sometimes the system is unable to control the tokamak plasma, resulting in a rapid termination of the discharge in a disruption which can impart high thermal and electromagnetic loads on the tokamak structure. We address this in Section 5 before moving to discuss the handling of the tokamak exhaust in Section 6. Conclusions and the implications for the timescale to fusion energy are provided in Section 7.

2. Energetic particles in the tokamak plasma

To achieve fusion-relevant conditions requires high power heating systems, operating at the multi-MW level. One approach is to launch electromagnetic waves into the plasma at frequencies that resonate with characteristic frequencies in the plasma, such as the cyclotron frequency associated with the gyration of the charged electrons or ions about the magnetic field lines. This accelerates those particles to high energy, which then transfer that energy to heat the whole plasma. Another approach is to accelerate ions outside the plasma to high energy (10's kev to MeV, depending on the size of the tokamak) and then neutralise them to form a high energy beam of neutral atoms, typically deuterium, that can then be injected into the plasma; this is called neutral beam injection. The energetic atoms are quickly ionised so that they are confined by the tokamak magnetic field and then give up their energy to the plasma. Both of these heating mechanisms provide a distribution of particles which have a much higher energy than the thermal energy of the plasma, which is typically a few keV. Finally, in a fusion reactor with a deuterium and tritium fuel mix, there will be fusion reactions that generate helium ions (or alpha-particles) at 3.5MeV and neutrons at 14.1MeV. The neutrons escape the magnetic field to become absorbed in the blanket, where heat for electricity generation and tritium fuel are created. The charged alpha particles need to be confined within the plasma so that they give up their energy to maintain the fusion conditions, offsetting the external heating power that would otherwise be required (zero external heating in the case of an ignited plasma).

Magnetised plasmas exhibit a number of waves. If these tap into the free energy of the plasma inhomogeneities, they can grow in amplitude leading to instabilities that have a deleterious impact on the confining magnetic field. Magnetohyrodynamic (MHD) waves have a phase velocity that can be characterised by the Alfvén speed. This is typically much larger than the thermal speed of the plasma ions, so there is little scope for resonant interaction and instability growth. However, the energetic particles arising from heating systems, or the fusion alpha particles in a DT plasma, can result in a significant number of particles that can resonate with these MHD waves, driving instabilities. A consequence is that the resulting electromagnetic fluctuations can enhance the transport of the energetic particles from the plasma. If they are lost before they transfer their energy to the background thermal plasma, then this degrades the heating capability and makes achieving the fusion conditions more challenging.

In addition to the deleterious impact on heating, these instabilities can threaten the structural integrity of the tokamak. For example, the electromagnetic fluctuations can eject the energetic particles in localised regions, leading to potentially large local thermal loads on the plasma-facing components of the tokamak vessel. These heat loads need to be quantified and, if necessary, appropriate protection built into the components that are affected – or restrict plasma operating scenarios to those where this has an acceptable impact. A recent example from the JET tokamak is shown in Fig 2 [2]. Mirnov coils detect frequent bursts of high frequency magnetic field fluctuations that are characteristic of a particular instability called the fishbone mode. This coincides with the appearance of a hot spot on the JET vessel wall. Simulations show that the predicted electromagnetic fluctuations can eject fast particles and work is in progress to quantify the impact for the wall.

ITER will be a key facility for research into fast particle instabilities driven by the alpha particles, and how those instabilities influence the alpha particle confinement. ITER has an objective to achieve Q=10 – that is ten times more fusion power than the external heating power injected into the plasma. As the alpha particles carry one fifth of the fusion energy, this means that the power available in the alpha particles to heat the plasma is twice the external heating power injected. This will be the first time that a tokamak plasma has been predominantly heated by its own fusion processes, and it will be key for developing our understanding and testing our confidence in models to provide input to the design of fusion power plants. There is only one other tokamak in the world that can explore the physics of fusion alpha particles, and that is JET – JET is the only tokamak in the world that can operate with the DT fuel mix, achieving Q=0.6 during its last DT

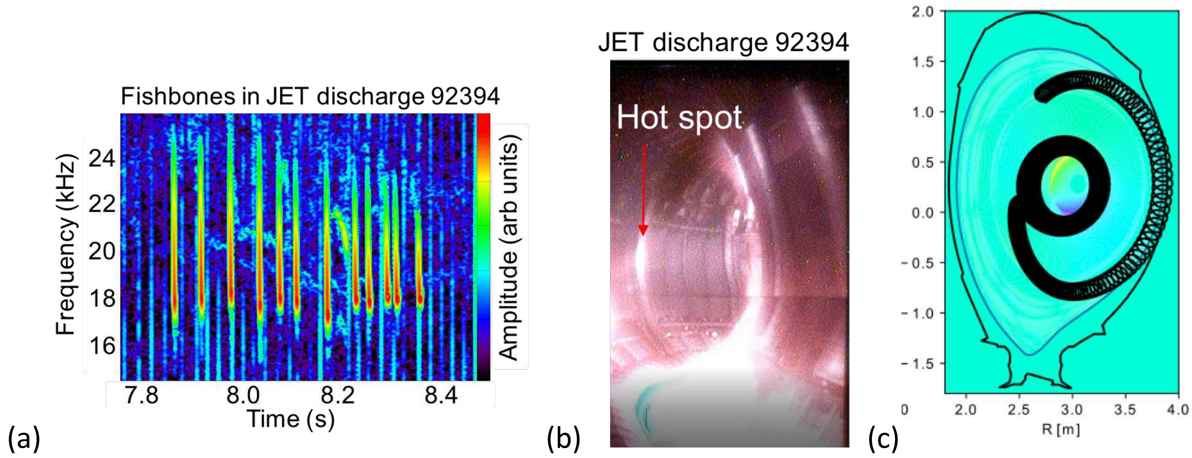

Figure 2: (a) Mirnov coil signal characteristic of a fishbone mode in JET; (b) a hot spot observed on the vessel wall during the fishbone activity, and (c) a simulation of the impact of the predicted electromagnetic fluctuations on a fast particle orbit, showing it can strike the vessel wall [2].

campaign [3]. Another DT campaign on JET is planned in the near future, and these experiments will provide valuable input for optimising ITER plasma performance, as well as testing our models and physics understanding.

3. Confinement

Confinement refers to the physics of how the magnetic field configuration confines the heat and particles of the plasma. In an idealised situation, charged particles execute closed orbits around the magnetic field lines and, provided these closed orbits have a width much less than the system size, one has perfect confinement. Particles have collisions, and the resulting scattering can cause a diffusion of heat and particles across the field lines. In a tokamak, there is a class of particles that is trapped on the outboard, low magnetic field region, bouncing back and forth along the magnetic field lines and drifting away from the flux surface containing those field lines by a distance ~cm (for ions, ~mm for electrons) called the banana width. This is larger than the Larmor radius by a factor ~5, and provides the step length for the diffusive transport process. The time step is set by an “effective” collision frequency to scatter trapped particles into passing orbits. The resulting neoclassical transport is usually negligible, except in some parameter regimes (e.g. spherical and/or small tokamaks, and steep pressure gradient regions) where the heat conducted through the ions has a significant neoclassical component.

Unfortunately, the observed confinement is substantially degraded compared to the neoclassical prediction; this is now known to be a consequence of plasma turbulence. The drive for this turbulence has a range of different mechanisms. Some predominantly result in fluctuations in the electrostatic potential, with negligible fluctuations in the magnetic field, while others result in fluctuations in the full electromagnetic field. The latter is typically associated with regions of very steep pressure gradient, or regions of high β, where β is the ratio of the thermal to magnetic energy densities in the plasma – as found in spherical tokamaks, for example. Turbulence in neutral fluids is already complicated – the interaction between the oscillating charged particles of the plasma and the fluctuating electromagnetic fields adds a further complication, creating bifurcation and self-organisation phenomena that exhibit features of a complex system. It is a multi-scale physics challenge with several feedback mechanisms.

The strong heating required to approach fusion conditions has the effect of "stirring" the turbulence, and enhancing the loss of heat and particles. The result is that the heating is typically not so effective at raising the plasma temperature, and the confinement is rather lower than might be expected. We call this state the low confinement or L-mode. In principle, one can overcome the confinement degradation through a number of ways, such as: (1) inject more power to overcome the excessive losses; (2) make the tokamak bigger so that fusion conditions can be achieved in the core even if the turbulence constrains the temperature gradient to be low, or (3) improve our understanding of turbulence and identify approaches to suppress it.

Clearly confinement is a key input to tokamak design: (1) it determines the heating power that must be incorporated into the reactor design to achieve the fusion conditions; (2) it determines the size of the reactor core, and (3) it determines how much exhaust power has to be handled (whatever heating power is required has to be exhausted somehow – including from alpha particles). We know empirically that the energy confinement time (the time taken for half the thermal energy of the plasma to diffuse out) increases almost linearly with plasma current in a given regime. In addition, the maximum plasma current that can be accommodated is proportional to the magnetic field or violent instabilities develop (see Section 5). Thus confinement is a significant driver for determining the plasma current and magnetic field that must be accommodated in the design.

The most compact fusion reactors will benefit from high confinement. This will support a steep pressure gradient in the plasma which, in turn, will provide a fusion relevant core pressure in a smaller plasma. A key question, then, is can we suppress the plasma turbulence? From experiments and theory, turbulence is believed to be suppressed by sufficient sheared flows. Actually, theory predicts that the situation is more subtle: sheared flows across magnetic field lines suppress turbulence, while sheared flows along magnetic

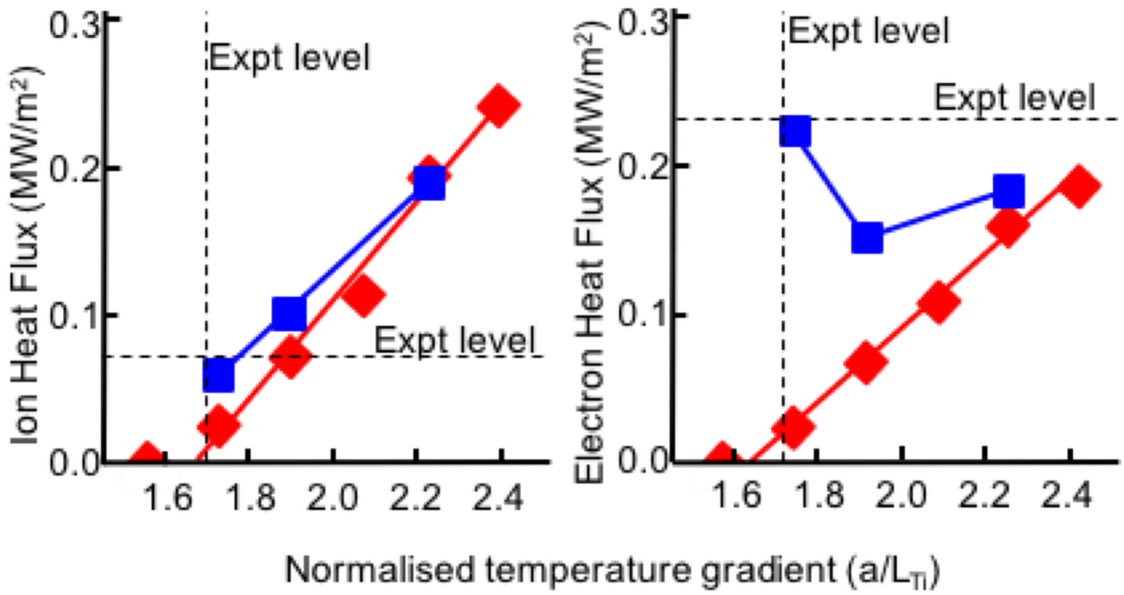

Figure 3: Predictions for the ion (left) and electron (right) heat flux as the normalised ion temperature gradient is increased. The red diamonds show the results when fluctuations at the electron scale are not treated; the blue squares show the heat flux predictions obtained when both ion and electron scale fluctuations are treated self-consistently, including interactions. Figure reproduced from [3].

field lines drive turbulence. Furthermore, turbulence itself can drive flows via Reynolds' stress, and this is thought to play a key role in the turbulence saturation mechanism in some situations. Turbulence can also lead to momentum transport, so that torques providing momentum injection at one point in the plasma can result in flows elsewhere. The situation is further complicated by the large range of temporal and spatial scales involved. For example, turbulence involves interacting scales all the way from the sub-millimetre electron Larmor radius to the centimetre "meso-scale" between the ion Larmor radius and system size. This turbulence drives transport that influences phenomena at the system scale, such as the thermal gradients providing the turbulence drive, and the flows discussed above; these then feed back on the turbulence. An example of the importance of a self-consistent treatment of the different scales in simulations is provided in Fig 3 [4]. This shows the importance of the electron scales on the heat loss through the electron channel, but also that it has an impact on the heat loss through the ion channel close to marginal stability. Such multi-scale turbulence simulations are at the edge of what is achievable on today's high end computers, and will benefit greatly from the advent of exascale computing.

4. The pedestal

As discussed in Section 3, if turbulence can be suppressed then steep thermal gradients can form, opening up the prospect of a compact fusion device. As the heating power is increased through a threshold, a tokamak plasma typically undergoes a rapid bifurcation from the turbulent low confinement L-mode state to a high confinement H-mode state with reduced plasma turbulence. Actually, when the plasma is probed in more detail, it is found that the turbulence is only suppressed in the last few centimetres at the plasma edge. This creates an insulating region that keeps the whole core plasma hot, raising the central pressure significantly. We call such an insulating region a transport barrier. In the case of the H-mode, because the whole core pressure is raised to sit on a "pedestal" created by

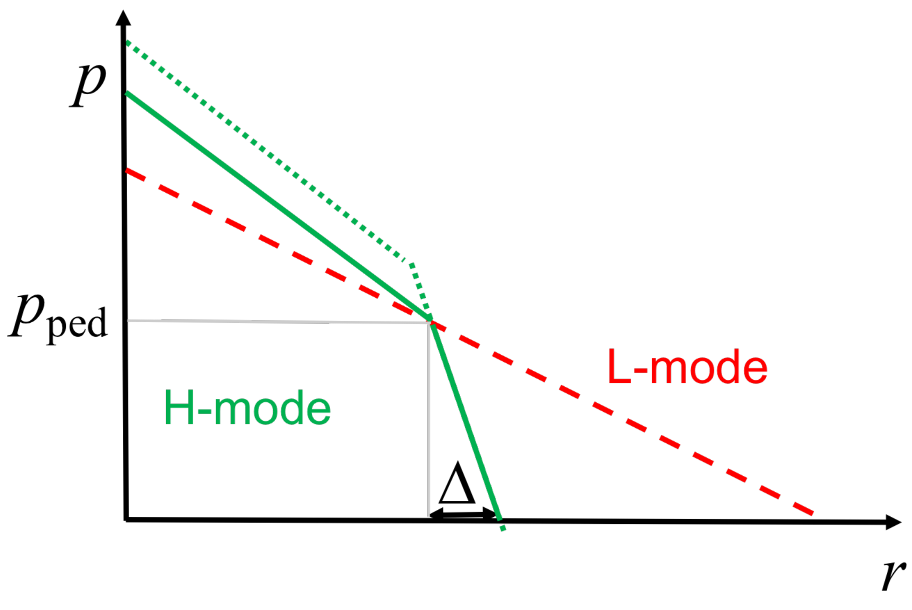

Figure 4: Cartoon of the pressure distribution from the hot core of the plasma (r=0) out to the cold edge for a turbulent L-mode plasma (red dashed line) and for a smaller H-mode plasma with a narrow (full line) and wider (dotted line) pedestal. The pedestal width, Δ, and height p_{ped}, are shown for the narrow pedestal case. [The diagram shows cartoons of two different plasmas, to illustrate that the same central pressure can be achieved in a smaller H-mode plasma than L-mode - the L-H transition does not result in a shrinking of the plasma.]

the transport barrier, this region of steep pressure gradient is also called the pedestal region. It is characterised by three parameters, illustrated in Figure 4: the pedestal width, Δ, the pedestal gradient and the pedestal height p_{ped}. These three are, of course, related. The key parameter for confinement is the pedestal height, which is the pressure at the top of the steep gradient region – the pressure at the plasma centre is almost proportional to this pedestal height in most situations. Thus the confinement and, in particular, the proximity to fusion conditions at the plasma centre, are highly dependent on the pedestal height. A model for this requires two parameters – the pedestal width and the pedestal gradient.

The pedestal is believed to form because strong flow shears are created at the plasma edge. These are certainly observed. While there are a number of theoretical models for how the flows form, and then how the turbulence is subsequently suppressed, there is no single universally accepted model and no quantitative predictive theory. If one accepts that the L-mode turbulence is suppressed by flows, then that allows the pedestal to form. As it approaches high pressure gradient, a number of electromagnetic instabilities are triggered which drive turbulence that can stop the gradient rising further. There are two broad classes of electromagnetic instabilities – those that ripple the magnetic flux surfaces are called twisting modes, while those that tear magnetic flux surfaces apart resulting in reconnection, are called tearing modes. A particularly strong twisting mode is the kinetic ballooning mode, or KBM, and this is a plausible candidate for constraining the pressure gradient [5]. Micro-tearing modes have also been observed near the pedestal top, so these may influence the pedestal width evolution [6]. A particularly successful model, which can predict the pressure pedestal height to 20% accuracy over a wide parameter range is the EPED model [5]. This assumes that the gradient is constrained by the KBM, and the pedestal widens until there is sufficient free energy to drive a plasma eruption called an edge localised mode, or ELM. This ELM collapses the pressure gradient for the cycle to then repeat. There is evidence from the evolution of the pedestal between ELMs on JET that supports a role for the KBM, but other micro-instabilities likely also influence the pedestal transport [7]. EPED only predicts the plasma pressure pedestal; recent extensions to this model seek to separate out the different roles for the density and temperature, for example [8, 9].

ELMs are a major concern for ITER and any next step tokamak with a large thermal energy stored in the plasma. Figure 5 shows three cross sections of the MAST plasma. The first is an L-mode plasma, characterised by fine scale (~cm) plasma filaments that light up the surrounding neutral gas and give the plasma boundary a blurred appearance. The second is an H-mode plasma, characterised by a sharp boundary that signifies a reduction in the edge plasma turbulence. Finally, we show a plasma with an ELM. Each eruption lasts in the region of 50-100μs on MAST, and there can be 100's of them per second [10]. One can clearly see that the eruption does not throw off a uniform shell of plasma, but rather a number of filaments, typically ten or more. These filaments erupt far from the plasma surface, depositing high thermal loads on plasma facing components, and transporting large amounts of heat and particles into the exhaust region, called the divertor (see Fig 1 or 6). Our theoretical understanding is that a combination of pressure and current density gradients provide the free energy for the instability drive – called a peeling-ballooning mode [11, 12]. Analytic theory [13] and simulations [14] provide an interpretation of the dynamics of the filaments, but this is extremely challenging and a quantitative predictive model for the energy expelled in an ELM event remains elusive. One intriguing result observed on JET is that the ELMs can sometimes be paced at a fixed frequency (or harmonics of it) [7] – while there are ideas for a possible mechanism, it is difficult to understand this in terms of the standard ELM model.

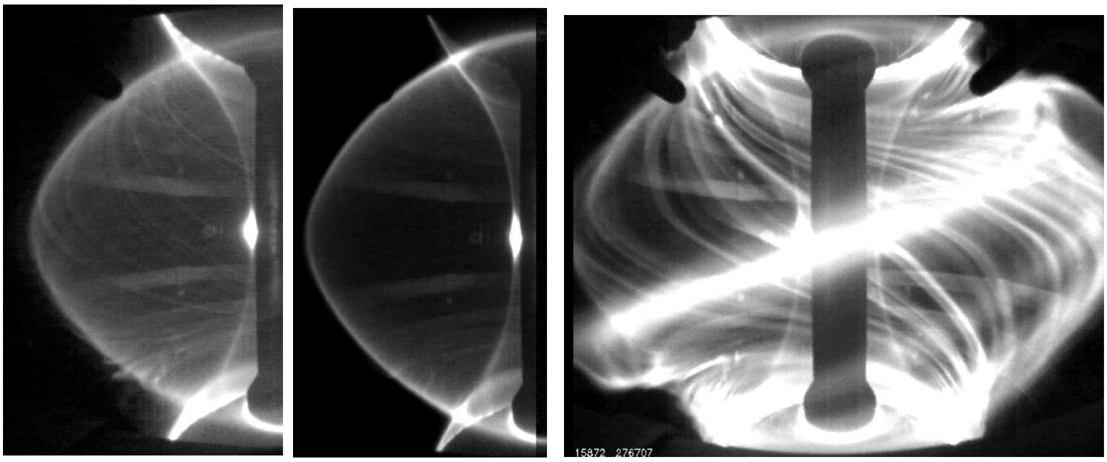

Figure 5: MAST plasma cross section in L-mode (left), H-mode between ELMs (middle) and H-mode during an ELM (right). Only the left half of the plasma is shown for the left and middle figures as they are approximately symmetric.

Extrapolations to ITER suggest that the largest of these ELMs cannot be tolerated as they would cause excessive damage to components. While regimes with small ELMs do exist on today's tokamaks, whether these extrapolate to next step tokamaks is uncertain. Therefore, a number of control mechanisms have been developed and tested, including oscillating the plasma up and down [15], or firing small pellets into the plasma edge [16]. In both cases the ELM frequency can be paced, and higher frequency ELMs have lower thermal energy thus mitigating the damage they cause. Another approach is to use coils arranged around the plasma to perturb the confining magnetic field in the pedestal region. There exist certain geometries of magnetic perturbations which influence the pedestal sufficiently to completely suppress the ELMs [17,18]. The physics is complicated by the complex plasma responses that occur, as well as the influence of plasma flows. Nevertheless, a picture is emerging [18], and resonant magnetic perturbation (RMP) coils will be installed on ITER as part of its ELM control tool set. It is also possible that ITER will be able to access parameter regimes where ELMs are absent altogether, with no need for external control. One example is the so-called QH-mode, which appears to require conditions such that a saturated electromagnetic structure sits in the vicinity of the pedestal and relaxes the pressure gradient below that necessary to trigger the peeling-ballooning mode that leads to ELMs. Plasma flows seem to play an important role [19], so a key question is whether ITER can produce the required flow with its high energy (and therefore low momentum per unit power) neutral beams; applied magnetic fields may provide a route to generate sufficient rotation [20].

5. Disruptions

An ELM is relatively localised in the pedestal. It causes a collapse of the pedestal plasma pressure gradient, but the main core plasma survives and feeds the pedestal, allowing it to grow again before the next ELM crash. There are, however, more global events from which the plasma cannot recover. These are called disruptions and can be caused by a number of events, including large scale plasma instabilities and fragments of material falling into the plasma from the vessel wall. There are three operational limits where instabilities, and therefore the likelihood of a disruption ("disruptivity"), tends to increase: a limit to the plasma current; a limit to the density, and a limit to the pressure. The plasma current is limited to a value that is broadly proportional to the magnetic field for a fixed tokamak geometry, but spherical tokamaks, for example, can carry much more current per unit field than a conventional tokamak. The physics of this limit is related to a plasma instability called the kink mode, and is relatively well understood. The disruptivity also increases as the density approaches the so-called Greenwald limit, which itself is proportional to the plasma current. There is a lot of empirical data for this limit, but we do not yet have a good quantitative model for the physics and consequences. Finally, the pressure is limited to a value that increases with magnetic field and, at least in some cases, the plasma current. There are a number of instabilities that can limit the pressure – ideal MHD ballooning modes are well understood, but instabilities that require a more complicated plasma model require further research to gain a complete understanding. The neoclassical tearing mode, for example, results from a filamentation of the current density that creates large scale magnetic island structures in the plasma. When these islands are small they degrade confinement, but when they are large they can cause disruptions. ITER has a control scheme that seeks to use focussed microwaves to cancel the current filamentation that drives the island to large amplitude, thereby shrinking them down below a threshold size when we believe the plasma will self-heal. This has been demonstrated on ASDEX Upgrade, for example [21]. Another type of instability is the resistive wall mode. This occurs when the pressure is increased beyond the theoretical stability limit for a plasma in an infinite vacuum – placing a superconducting wall near the plasma can stabilise the plasma allowing higher pressure for a given current. In practice, the wall will have finite resistivity, and then a slowly growing instability called the resistive wall mode can eventually lead to a disruption in a complex mechanism that involves electromagnetic torques between the plasma and vessel wall that influences the flow evolution.

The closer one positions the plasma to the operating limits, the more likely a disruption will be. Disruption avoidance strategies are being explored, but avoiding them altogether is a challenge wherever one chooses the operating point. The consequences are serious and constitute the main threat to the structural integrity of the tokamak – terminal failure of the device is a very real possibility if appropriate safeguards are not taken. First, the loss of control of the plasma during a disruption results in a high interaction with the wall, with extreme thermal loads that can cause melting. Second, a large fraction of the plasma current, several MA in the largest tokamaks, will flow through the vessel components as so-called halo currents. Given the Tesla-level fields that exist in the tokamak, this imparts huge forces on the vessel components and structure which the tokamak has to be designed to withstand. The problem is exacerbated by the fact that these currents do not flow uniformly, and their distribution may be related to the details of the non-linear plasma structures associated with the MHD instability that led to the disruption. This makes it difficult to quantify the magnitude of the forces, but also the directions in which they will act. The third challenge is that very high electric fields can form in the plasma during a disruption which will accelerate electrons to very high energy – so-called runaway electrons that can rapidly multiply via an avalanching effect. These can cause very serious damage to the tokamak structure if not mitigated.

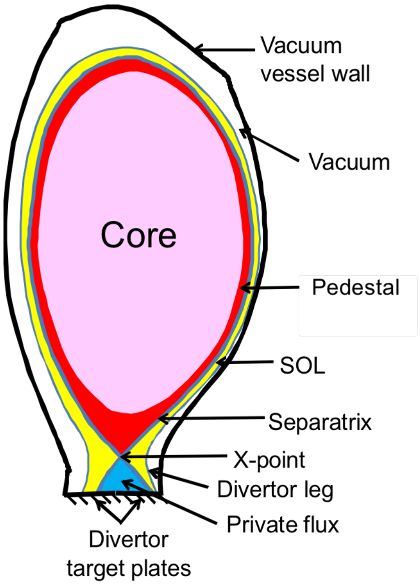

Figure 6: Cross-section of a typical tokamak plasma showing the hot fusion core, the pedestal and the scrape-off layer (SOL).

The most robust mitigation strategies revolve around sensing when a disruption is imminent and then rapidly increasing the neutral particle inventory in the vessel. The resulting radiation rapidly cools the plasma, which increases the resistivity, reduces the plasma current and mitigates the runaway electrons. Injection of large amounts of neutral gas has been demonstrated to be effective on today's tokamaks. However, this is insufficient for ITER, and a new technique based on

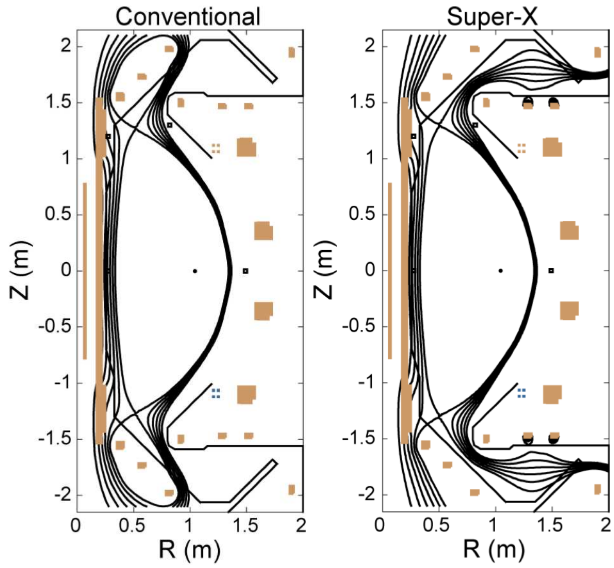

Figure 7: The flux surfaces of the scrape-off layer in the MAST-U tokamak for a conventional divertor configuration (left) and the Super-X (right). The squares show the cross section of copper coils; the geometry is rotationally symmetric about the vertical axis.

firing shattered pellets of deuterium ice into the plasma is soon to be trialled on JET [22].

6. Plasma Exhaust

All of the heating power that is injected into the plasma to bring it to fusion conditions, plus any power in the alpha particles in a burning plasma has to be exhausted, and managing this is a challenge. The power diffuses from the core via turbulence, as discussed in Section 3. After it crosses the pedestal, it enters a layer separated from the core plasma by a magnetic surface called the separatrix to enter a region called the scrape off layer (see Fig 6). This separatrix is formed using the poloidal field coils mentioned in Section 1. Inside the separatrix (ie the core and pedestal) the magnetic field lines lie in closed, nested toroidal surfaces that provide the good confinement required. Outside the separatrix, the field lines connect to material target plates at the top or bottom of the vessel (or both in some cases, such as MAST). Thus heat and particles that have diffused from the core cross the separatrix and are then transported along the scrape-off layer into the divertor where they are removed. The result is that all the power strikes the divertor target plates in two rings (or four if there are divertors at the top and bottom of the vessel). The area over which that power is deposited is thus a product of the circumference of the ring and the thickness of the scrape-off layer (often enhanced somewhat by the expansion of the magnetic flux between the tokamak mid-plane and the divertor region, and by inclining the target plates). The thickness of the scrape-off layer is another quantity that there is some uncertainty over. It depends on the ratio of the transport across flux surfaces to that along them, which is a small number, of course. The transport across flux surfaces is governed by turbulent processes, and probably not diffusive – a quantitative, first principles predictive model remains elusive, but there has been significant progress recently that provides estimates for future tokamaks [23]. Unfortunately, those predictions show that the exhaust is extremely challenging and we do not have a robust technical solution.

Radiating power from the edge plasma region is likely to be important in future reactors; while this reduces the power going to the divertor, it has to be achieved while maintaining the fusion conditions in the plasma core. Also by introducing gas into the divertor region it is possible to create a radiative buffer that protects the divertor target from the hot scrape-off layer plasma. This is called detachment. Understanding the conditions for detachment is an area of active research – we need to make sure that the scrape-off layer plasma cannot burn through it, whilst preserving good fusion performance by keeping the detachment front away from the core plasma. Studies give us confidence that detachment is a viable solution for ITER, but it is unlikely to be sufficient on its own for fusion power plants beyond that.

As part of addressing the challenge to identify a robust exhaust solution for fusion power plants, one option is to explore and test new magnetic geometries, including engineering feasibility in a reactor. One avenue is the so-called snowflake divertor that is being developed on the TCV tokamak and elsewhere, for example [24]. This establishes a magnetic field structure to divert the power along multiple legs, creating multiple strike points. Another approach is the Super-X divertor [25] to be developed on MAST-U [26]. Figure 7 shows two magnetic geometries available in MAST-U for a conventional divertor and the Super-X. One can see that in the Super-X, the outer leg is pulled out to larger major radius, which increases the heat deposition footprint because of the larger circumference of the strike point ring. However, there are other advantages: (1) the flux expansion is greater for the Super-X and can be controlled by the divertor coils; (2) there is scope to introduce gas into the closed box of the divertor, keeping it remote from the core, and (3) the length along a field line from the mid-plane to the target is very much longer for the Super-X than for the standard divertor configuration (the field lines map around the vertical axis many more times).

7. Conclusion remarks

We have reviewed a number of tokamak plasma physics issues that influence the design and construction of fusion power plants. The review is not intended to be exhaustive, but rather focuses on some of the more generic issues that are expected to underpin all designs. Other issues one might introduce include impurity transport and accumulation, and non-inductive current drive for example. An aim of this paper is to assess the extent to which plasma physics influences the timescale to a tokamak fusion power plant. This is a rather subjective issue, and depends to a large extent on the risk – a financial risk for the funder and reputational risk for the organisation that takes the decision to construct. Let us consider each area discussed in previous sections in turn.

Models for fast particle instabilities and their impact on fast particle transport are not complete, but we have an advanced predictive capability that enables us to make quantitative predictions. There are uncertainties, but these will likely need a burning plasma to validate our models and further improve confidence. The anticipated experiments in DT fuel planned for JET will provide good data to further reduce uncertainties and to help accelerate the development of operating scenarios on ITER. ITER itself is expected to clarify any remaining issues so that one can design a power plant beyond that to take account of fast particle effects with some confidence. This issue need not slow down the development of a fusion power plant design.

Confinement has a significant influence on the design. It drives the size of the device to a large extent, but also the current it should carry and, therefore, the magnetic field required. It determines the heating power that is necessary to achieve fusion conditions – external heating plus alpha power. All this heating power has to be handled in the exhaust system, and we have already discussed how challenging this is in Section 6. If we can achieve high confinement then we can access fusion conditions with less total heating power, thus easing the exhaust constraint if we reduce the fusion power accordingly. We have not discussed current drive very much, but it is a challenge for steady state operation of a tokamak if that is thought to be a design requirement. Advanced confinement regimes would allow operation at lower current; this not only reduces the total current required, but also increases the fraction of “free” bootstrap current [27] that arises from the plasma pressure gradient. Clearly the confinement time is a major driver in designing a fusion reactor and is an essential input.

A first principles approach requires knowledge of the turbulence in the core and a model for the pedestal height (see Section 4). The community is making good progress in developing a predictive capability, and there are encouraging signs that this may be possible within a few years. One could proceed in advance of such a predictive capability provided either: (1) one adopts a staged approach to fusion, each stage a modest extrapolation from the last, or (2) one builds sufficient contingency in the design, or (3) one accepts the risks that the device may not meet its fusion power requirements, and, ideally, has a backup strategy to nevertheless demonstrate progress and therefore return on the investment. Approach (1) is lowest risk but likely results in the longest timescale – one also has to ensure it doesn't stifle innovation through incremental advances; approach (2) could be faster, but could also lead to enhanced capital and operational costs, and approach (3) is also potentially faster, but carries financial and reputational risk.

If one is relying on the good confinement of the H-mode, then an ELM control or avoidance strategy is important. RMP coils, for example, would need to be embedded in the design from the outset, with due attention given to the impact of fusion neutrons. To proceed without a proven concept for managing ELMs in mind is likely to lead to excessive erosion of material surfaces, requiring high levels of maintenance and low availability with a consequent detrimental impact on potential investors in fusion power. Disruptions are an even larger issue – one major disruption could do terminal damage to the fusion reactor. It is therefore crucial to have a disruption avoidance and mitigation strategy in place. Advanced confinement regimes, if they exist, can again help the disruption issue, allowing operation at lower current and magnetic field and thereby reducing electromagnetic forces.

The divertor has a significant impact on the tokamak design and therefore it is difficult to envisage proceeding to design and construct a fusion reactor without a specific exhaust concept in mind. There are a number of recent innovative ideas to manage the plasma exhaust which offer much promise. However, it remains unclear which, if any, is optimal and what the impact on the core plasma performance is. Data to come out of existing tokamaks in the coming years, together with accurate models to extrapolate reliably, will provide a solid basis on which to proceed.

In summary, there are a number of plasma physics issues that have a significant influence on the design of a tokamak fusion reactor, and often these interact. While we have focused on the plasma physics questions here, solutions will involve advances in both science and technology – some of the plasma physics issues can be eased by advances in technology and vice versa. An optimal path forward must therefore integrate the two, but clearly the plasma physics questions have a big influence on the timescale to fusion power whatever the approach. Quantifying that influence depends on the level of financial risk that investors are prepared to make, and the reputational risk that organisations can carry.

Acknowledgements. The author would like to thank J.W. Connor, J. Harrison, A. Kirk, B. Lipschultz and A.W. Morris for helpful comments. This work has been carried out with funding from the RCUK Energy Programme [grant number EP/P012450/1].

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