4 Tokamak Science

4.1 INTRODUCTION

The Tokamak Science programme has continued its primary focus on key issues for ITER and DEMO-class facilities (DEMO and Component Test Facilities), integrating theory, modelling and experiment closely, and using results from MAST, JET and other tokamaks. As in recent years the programme is organised in five interacting areas:

- Integrated Plasma Scenarios
- Core Plasma
  - MHD Stability
  - Confinement and Transport
  - Fast Particle Physics
- Scrape-off layer and Divertor

In addition there are three "infrastructure" areas:

- MAST operations (see Chapter 5)
- Theory and Modelling
- JET campaigns and technical developments

This wide-ranging programme is the platform for the CCFE input to high performance operable plasma scenarios for ITER and DEMO-class devices (including Component Test Facilities), and is deeply integrated with the European programme. This integration that will continue and expand with the new Eurofusion Consortium at the heart of the magnetic confinement fusion research in the Horizon 2020 framework programme, and substantial effort was devoted in the latter part of 2013 in preparing CCFE’s participation in this.

The relevant MAST (M9) and JET (C31, C32) campaigns ran from late May to the end of September and from mid-July till mid October respectively. The MAST campaign was the final one before the major upgrade (see Chapter 5), and would thus provide the data for interpretation and research for the duration of the shutdown and the intervening conferences. The 2013 JET campaign was unfortunately terminated early due to a technical failure (now recovered), but not before some key new results had been obtained.

As in 2012-13, the programme was arranged under four overall long-term objectives:

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1 In the US the CTF falls under the general class of Fusion Nuclear Science Facilities, FNSF
1. Predictable integrated plasma scenarios for ITER, CTF, DEMO. In particular developing high performance scenarios on JET with the new ITER-like metal wall and on MAST, preparing scenarios for MAST-Upgrade, and developing integrated computational tools to model scenarios.

2. High performance core plasmas with tolerable instabilities. Specifically investigating turbulent heat, fuel and impurity transport to find ways to control it as a path to physics-based high performance regimes, avoid or mitigate large scale instabilities and develop understanding of fast-particle driven instabilities and their effects so as to minimise their impact in future devices.

3. An effective edge pedestal. This covers formation of a suitable edge pedestal based on physics understanding, including techniques to avoid or mitigate Edge Localised Modes (ELMs) and includes development of an understanding of the L- to H-mode transition.

4. Predictive capability to design credible exhaust systems for ITER, CTF, DEMO. In particular to improve understanding of the processes controlling the scrape-off layer (SOL) width and power load and how the power load at the plasma facing components can be reduced in ITER and DEMO-class devices. The studies of plasma facing materials including erosion and redeposition are described in Chapter 6.

Some of the tokamak science programme in 2013 was co-ordinated under EFDA projects, in particular the JET work is managed almost entirely under a system of task force leaders and deputies working with an EFDA close support unit at Culham. Three of the eight JET Task Force leaders and Deputies are from CCFE. The rest of the EU EFDA work is managed by two other EFDA departments, ITER physics and Power Plant Physics and Technology (PPP&T). There is some tokamak science contribution to PPP&T, described in chapter 7. The ITER Physics activity includes a number of specific scientific topics, and also the substantial Integrated Tokamak Modelling programme (the leader of the Integrated Scenario Modelling activity within this is from CCFE) and the scientific contribution to JT-60SA. CCFE is one of the larger participants in the EFDA ITER Physics activity.

Another high level structure helping to co-ordinate the work in preparation for ITER is the International Tokamak Physics Activity, ITPA, which is divided into seven Topical Groups which bring together the world’s leading fusion plasma scientists. CCFE is well represented in these groups, which develop co-ordinated experiments and modelling work around the world, and CCFE contributes to a significant number of these tasks.

4.2 INTEGRATED PLASMA SCENARIOS
The goal of developing tokamak plasmas with sustained fusion power output requires the integration of many factors, such as: effective heating, to raise the plasma temperature to the levels where fusion reactions occur; efficient fuelling, to supply the plasma with particles to fuse; good energy confinement, to maximise the ratio of fusion output to heating input; good plasma stability; to sustain the high rate of fusion reactions and protect the device structure; and effective exhaust strategies, to safely remove heat and impurity particles from the plasma. The simultaneous optimisation of these factors, called the development of integrated plasma scenarios, often requires compromise between conflicting requirements, and changes in the solution for any one element can have knock-on consequences for the whole system. This has been the case with the replacement of the carbon wall in JET with a metal wall using the materials that are incorporated in the design for ITER. For example, the change in wall materials has affected the impurity contamination of the fusion plasma, modifying the power loss due to radiation. Experimental strategies developed with the previous carbon wall have been adapted to compensate for such changes, providing valuable input to the development of plasma scenarios for ITER. Substantial progress has been made in 2013 on the understanding of confinement with the ITER-like wall and the reasons for the changes in the pedestal properties, but it will mostly be reported in 2014 so is not described here.

Likewise the future development of spherical tokamaks requires the effective use of heating and current drive systems to initiate and maintain the plasma. Experiments on MAST have shown the potential for radio frequency and neutral beam injection systems to be used efficiently for these tasks.

The extrapolation from present experiments and the design of integrated plasma scenarios for future devices relies heavily on the capabilities of predictive simulations. With this in mind the models currently available to describe integrated plasma scenarios are being tested against existing experimental data to identify areas where future development is needed and to increase confidence in the predictive capabilities (the integrated simulation tools also need to be tested continually for sensitivity to the impact of improved physics in the various modules).

### 4.2.1 Developing Tools to Control Plasma Radiation

It has been found that the deliberate injection of nitrogen gas in metal wall devices can increase the plasma temperature while simultaneously radiating power from the edge region, and so reducing the peak heat-loads on the wall. However, some nitrogen ions are expelled from the plasma periodically by plasma instabilities localised at the plasma edge (ELMs), and this generates a source of tungsten which eventually contaminates the plasma centre and radiates heat, cooling the region where fusion reactions are most intense. It has previously been seen in ASDEX Upgrade experiments that this effect can be counteracted by central plasma heating. Efficient central
heating can be provided on JET by transmitting high power radio frequency waves into the plasma at the resonant frequency of the central plasma ions, called Ion Cyclotron Resonant Frequency (ICRF) heating. During the early experiments with the JET metal wall the impact of the change in wall materials on the operation of this heating system was assessed. It was found that the waves could be transmitted into the plasma with similar efficiency to previous experiments with the carbon wall. Heat-loads on the tokamak wall were carefully monitored during operation of the ICRF system due to the risk of melting the beryllium tiles adjacent to the radio frequency (RF) antennas. The local heat fluxes were measured using infra-red thermography to be up to 4.5MW/m$^2$, within the metal wall power handling capabilities and consistent with modelling calculations. These encouraging results indicated that efficient high power ICRF heating is compatible with the JET metal wall, but an increase in the source of tungsten from the tokamak wall was also observed during ICRF operation, raising the question as to whether it could be used as an effective control tool for plasma tungsten contamination. Nevertheless, experiments where 3MW of ICRF heating was applied to a plasma with 15MW of neutral beam heating indicate that effective central heating was achieved and tungsten levels in the centre of the plasma were reduced, despite the additional source of tungsten from the wall (see Figure 4.1).

![Figure 4.1 Time traces of the electron temperature at the plasma centre for two JET cases: #83597, where only neutral beam heating was used; and #83603, where 3MW of ICRF heating was added from about 14.4s, resulting in significantly higher central temperature due to the additional central heating and reduced central tungsten contamination.](image)

This technique is now being used to mitigate the effects of tungsten contamination in integrated plasma scenarios that use nitrogen gas injection, leading to the possibility to sustain the good confinement and heat exhaust conditions for much longer periods. This
demonstration of plasma contamination control is also encouraging for future devices that produce high fusion power because deuterium-tritium fusion reactions produce high energy helium ions (alpha-particles) that are confined in the plasma and provide central heating in a similar way to ions heated by ICRF waves.

4.2.2 AVOIDING FAST ION REDISTRIBUTION BY INSTABILITIES

A key technique for heating tokamak plasmas to fusion temperatures is by providing highly energetic ions inside the plasma that heat the remainder of the ions and electrons by collisions. These, so called ‘fast ions’ can be injected using high energy neutral beams, accelerated using radio frequency waves (ICRF) or generated as the product of fusion reactions (primarily alpha particles). Whatever the origin, it is desirable to confine them in the plasma so as to gain the greatest effect in terms of plasma heating. However, it has long been known that plasma instabilities can redistribute fast ions in the plasma and even expel them from the plasma altogether. On MAST neutral beams provide high power heating and drive part of the plasma current, but the fast ion population generates a pressure gradient within the plasma that drives plasma instabilities, such as bursting ‘fishbone’ modes. The modes strongly redistribute the fast ions, potentially reducing the heating and current drive efficiency. Previous experiments have shown that the redistribution can be reduced by broadening the fast ion pressure profile within the plasma, for example by moving the neutral beam trajectory to an off-centre location, one of the main features of the MAST upgrade (Chapter 5). But the fast ion pressure gradient can also be reduced by increasing the plasma density, which increases the rate at which fast ions slow down in the plasma. So experiments have recently been performed to investigate the possibility that a domain of operation can be accessed at high density where fast ion redistribution can be avoided.

Figure 4.2 Time traces of MAST magnetic field perturbations generated by plasma instabilities for 8 plasma pulses. The left hand side have 1.5MW of neutral beam heating compared with 3MW on the right. The density increases from the bottom to the top for the time window between the blue lines. Bursting ‘fishbone’ instabilities can be seen to diminish as the density increases during this period, and are avoided altogether in the high density case at 1.5MW.
Figure 4.2 shows the effect of increasing the plasma density and decreasing the neutral beam heating power on the ‘fishbone’ instabilities. At the same density the instabilities are weaker at lower power, corresponding to lower fast ion pressure gradient. As the density increases the instabilities become weaker still, eventually disappearing altogether in the low power case. This suggests that there is a threshold density above which ‘fishbone’ instabilities can be avoided, and that this threshold depends on the level of neutral beam power applied. It is expected that other factors that affect the fast ion pressure, such as plasma temperature, should also be taken into account, but in practice the result indicates the existence of a domain at high plasma density where this class of instability can effectively be avoided in MAST. This is a notable advance – in the past it had only been shown that the instabilities could be reduced (but not eliminated) by off-axis neutral beam injection.

If there is no redistribution of the neutral beam fast ions by instabilities, it is expected that the rate of production of fusion neutrons should be roughly proportional to the applied neutral beam power if the power is varied with all other plasma and beam parameters kept constant. Actually, for MAST the effect of doubling the power from 1.5MW to 3.0MW (as in Figure 4.2) is to increase the neutron rate by slightly more than a factor of 2 due to slight increases in plasma temperature and fusion reactions produced by fast ions colliding with each other. An empirical ratio of about 2.3 has been established from these experiments when it is thought that no redistribution is occurring. Figure 4.3 shows the effect of the fast ion redistribution on the ratio of neutron production from 1.5MW and 3.0MW plasmas at the same density in Figure 4.2. It can be seen that the fast ion redistribution becomes progressively stronger as the density is reduced from the threshold value.
4.2.3 STARTING PLASMAS WITH RADIO FREQUENCY SYSTEMS

Most tokamaks use magnetic field coils (the solenoid) in the centre of the device to initiate the plasma and provide the current inside it through induction. This technique is highly efficient, but cannot be sustained indefinitely due to the requirement that the current in the coils must be continuously and monotonically changing. An additional issue for spherical tokamaks, such as MAST, is that the space for this central magnetic coil is limited, and it is desirable to avoid them altogether in future spherical devices. Because of this experiments have been performed on MAST to achieve plasma start-up without using the central coil. The plasma initiation and current ramp-up were provided instead by an RF system operating at 28GHz which excites a wave in the plasma called an electron Bernstein wave (EBW). Previous experiments on MAST have demonstrated that this technique is effective for initiating the plasma and plasma currents up to 33kA were achieved without the use of the central coil. Recent experiments have extended this work, using longer RF pulses (up to about 0.5s compared with about 90ms previously) to increase the plasma current to 73kA, a new record for MAST and an impressive current drive efficiency of approximately 1Amp per Watt. Figure 4.4 shows the comparison of the previous achievement with the new result.

Figure 4.3 Ratio of neutron produced in MAST by 3.0MW and 1.5MW of neutral beam power at the same plasma density. At high density no fast ion redistribution is observed, but the impact of the redistribution becomes gradually stronger as the plasma density is reduced.
4.2.4 SIMULATING INTEGRATED PLASMA SCENARIOS

A key aim for the future is to be able to confidently predict plasma conditions in future devices to provide confidence for engineering design decisions and to prepare plasma scenarios for the eventual operation. One issue that requires integrated modelling of the plasma core and the region outside the main plasma and reaching as far as the tokamak wall is the efficiency of plasma fuelling. Many present devices rely on gas injection into the tokamak chamber to fuel the plasma, and on MAST gas inlets on the centre post are used to fuel the plasma and this will continue after the upgrade, since it has been found to be an efficient technique. However, previous MAST experiments with such a fuelling location highlighted low core-fuelling efficiency accompanied by a peak of the electron density inside the separatrix in correspondence of the location of the gas inlet. They also exhibited a strong poloidal asymmetry over the outer magnetic surfaces. Since the JET integrated modelling code, JINTRAC, has recently been modified to allow simulations to be made for MAST, this provided an opportunity to test whether a simulation could capture this effect and allow understanding the observed phenomena.

Figure 4.5 shows the radial profile of the electron temperature and density for a plasma that was being gas fuelled from a location on the centre post. A density peak can be clearly seen on the inside edge of the plasma density profile. Initial simulations using the EDGE2D edge code and EIRENE neutral particle model with low wall-pumping efficiency did not reproduce the localised plasma density peak near the gas inlet. However by increasing the pumping in the MAST vessel and accounting for radiation in the SOL it was possible to find conditions in which the accumulation and the asymmetry can occur, namely when the carbon tiles on the main chamber wall act as a
pump (e.g. when two particles strike the chamber wall in the model, only one particle is released back into the chamber). When this assumption is used in the model the density peak near the centre post is reproduced, as seen in Figure 4.5. This preliminary modelling helps ruling out more complex effects such as breaking of magnetic surfaces and island formation in proximity of the valve due to increased resistivity. This case illustrates how the sensitivity of plasma phenomena to plasma or material conditions can be investigated with integrated modelling tools.

Figure 4.5 (left) Electron density (top) and temperature (bottom) for a MAST plasma fuelled from the centre post, showing a density peak near the fuelling location. (right) A simulation showing the 2-D edge plasma density for a case with gas fuelling from the centre post, also showing a density peak near the fuelling location when strong pumping by the plasma facing components is included

4.3 CORE: MHD STABILITY ELMS AND PEDESTAL

The main effort in 2013 has continued to be in the control of ELMs, the physics behind the application of ELM-mitigating resonant magnetic perturbations (RMPs) and the understanding of the stability of the edge pedestal (whose free energy drives ELMs). Plasma flows are key to the physics of RMPs and also to the stability of high-$\beta$ instabilities, notably resistive wall modes (RWMs) that are likely to be the ultimate limit of pressure in a tokamak (this will be a key topic on the new JT-60SA tokamak under construction in Japan under the Broader Approach).

4.3.1 RESONANT MAGNETIC PERTURBATIONS

A Rotating resonant magnetic perturbations
The control of edge localised modes (ELMs) is a key requirement for ITER to control the heat flux to the divertor, which acts to limit the divertor lifetime. The control of ELMs can take the form of ELM suppression, where the ELMs are completely removed, or ELM mitigation where the ELM frequency is increased and the amplitude reduced. The increase in the ELM frequency decreases the energy loss per ELM and thereby decreases the peak heat load to the divertor. ELM control can be achieved by using a non-axisymmetric resonant magnetic perturbation (RMP). The application of a RMP to the plasma gives rise to the formation of X-point lobes due to the 3D nature of the applied field. These lobes extend down to the divertor, and in the regions where the lobes intersect the divertor there is a deposition of power. The lobes therefore give rise to a splitting of the strike point. As the splitting is intrinsically non-uniform toroidally, this may cause the surface of the divertor to become uneven, which may lessen the ability of the divertor to handle the power directed onto it.

![Image](image.png)

Figure 4.6. The effect of rotating n=3 RMPs on a) the plasma, showing the ELM mitigation (smaller more frequent spikes on the D(\alpha) emission as a function of time through the discharge) and b) the strike point splitting as measured by infrared thermography as a function of time along the x axis and spatial position along the y axis during a quarter rotation (the diagonal stripes an artefact of the technique used to make the figure).

One means of mitigating the erosion from the asymmetric splitting is to apply an RMP that rotates toroidally around the machine as a function of time. The effect of applying an n=3 rotating field is shown in Figure 4.6 a) which shows the effect of the RMP on the ELM frequency. It can be seen by comparing the middle panel (RMP off) with the bottom two panels (RMP applied) that the application of the RMP doubles the ELM frequency, irrespective of the direction (clockwise or counter clockwise) of rotation.

The effect of rotation on the strike points is shown in Figure 4.6 b), which shows the heat flux to the divertor surface, as measured by IR thermography. The splitting pattern can be seen to be localised at \( \Delta R_{\text{LCFS}} = 0.0, 0.015 \) and 0.03 m at the start of the rotating phase and the RMP field is rotated, the strike point splitting pattern changes, becoming more closely spaced at the end of the rotation compared to the start of the rotation.
B  Lobe Imaging

The spatial structure of X-point lobe structures due to RMPs was studied in detail in the previous M8 (2012) experimental campaign using a filtered camera viewing the divertor and recording light from He\(^{+}\) ions in the scrape-off layer (SOL). In the M9 campaign, the recently commissioned coherence imaging diagnostic has been used to make the first measurements of C\(^{2+}\) impurity flow velocities within the lobes. Figure 4.7 shows measurements of line-averaged impurity flows, which support preliminary EMC3-EIRENE code simulations that transport within the SOL has a 3-D structure in the presence of RMPs. EMC3 is a 3-D fluid code for the edge plasma.

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**Figure 4.7:** Left - coherence imaging measurements of line-averaged C\(^{2+}\) flows within the lobes. Right - preliminary EMC3-EIRENE simulation of the Deuterium parallel velocity in a MAST shot with RMPs.

B  The effect of RMPs on LH-transition

The baseline scenario on ITER depends on getting into H-mode with reasonable injected power, yet there is a strong desire to mitigate the first ELM after the transition, and it has been found that ELM-coils can make the transition into H-mode more difficult. A series of experiments were carried out on MAST applying RMP in \(n=2,3,4,6\) configurations to a 400kA lower single-null discharge to determine the impact on the L-H transition. The RMPs are applied before the L-H transition time in the discharge with no applied RMP.

The discharge shown in figure 4.8 has 1.4kA of current in each of the RMP coils and 1.5MW of beam power. Applying 0.5MW of beam power is sufficient to cause an L-H transition when no RMPs are applied, hence these discharges are significantly above the no RMP L-H transition power threshold. The shots are in density feedback, maintaining the density at the transition time closely the same as for the no applied RMP discharge.
The application of \( n=3 \) and \( n=4 \) RMP in figure RMP3 cause a short delay of the L-H transition time (\( \sim 1 \tau_e \)). The application of \( n=6 \) also results in a delayed transition, although the delay is significantly longer. The magnitude of the applied RMP field in this case is sufficient to cause an approximate doubling of the ELM frequency in the \( n=3,4 \) and 6 cases. At the same RMP coil current level, the application of \( n=2 \) RMPs causes a complete suppression of the transition.

### C Modelling of RMP rotation braking

One critical theoretical aspect of controlling ELMs with RMPs is to understand how the applied external field penetrates into the plasma, and how the toroidal rotation of the plasma is self-consistently modified during this process, since rotation is a key factor in the penetration process. In order to provide a quantitative modelling capability for this process, a full toroidal, quasi-linear model (MARS-Q) has been used. This model couples the linear, single fluid plasma response to a toroidal momentum balance equation, which includes torques induced by both fluid electromagnetic force and by (kinetic) neoclassical toroidal viscous (NTV) force. The numerical results for a test toroidal equilibrium quantify the effects of various physical parameters.
parameters on the field penetration and on the plasma rotation braking. The neoclassical toroidal viscous torque plays a dominant role in certain region of the plasma, for the RMP penetration problem considered in this work. Shown in Fig. 4.9 is one example of the simulation results on the time evolution of both radial profile and amplitude of the rotation change during the RMP application period.

Figure 4.9 MARS-Q simulated time evolution of the radial profile and amplitude of the change of the toroidal rotation frequency during the n=1 RMP penetration for a toroidal plasma. The arrow indicates time, and at the end the plasma rotation is essentially stopped or even slightly reversed between q=3 and q=4 surfaces (the rightmost dashed lines)

4.3.2 Resistive Wall Mode Modelling
The stability of the resistive wall mode (RWM) critically depends on the plasma flow. On the other hand, 3D magnetic perturbations due to the onset of the mode, which is essentially locked to the resistive wall, tend to brake the plasma flow. The non-linear interplay between the RWM and the toroidal plasma flow is numerically investigated in a full toroidal geometry, by simultaneously solving the initial value problems for the n=1 RWM and the n=0 toroidal force balance equation. In this study, the neoclassical toroidal viscous torque is identified as the major momentum sink that brakes the toroidal plasma flow during the non-linear evolution of the RWM. This holds for an RWM that is initially either unstable or stable. For an initially stable mode, the braking of the flow, and hence the eventual growth of the mode, depends critically on the initial perturbation amplitude as shown in the figure 4.10
Figure 4.10 Time evolution of the amplitude of the perturbed radial field at the $q=2$ surface during the non-linear interaction between the initially stable RWM and the plasma flow, with various initial amplitudes for the mode. The non-linear coupling between the mode and the plasma is switched on at the vertical dashed line.

4.3.3 ELMs and Pedestal Stability

A Observations of type I ELM triggering mechanism in MAST

Since the pedestals of most tokamak plasmas with Type I ELMs are found to be limited by the peeling-ballooning mode (PBM) stability boundary, these modes are likely to play an important role in the ELM crash which is observed to be linked to explosively growing filaments. In MAST filament structures rotating in the co-current (ion diamagnetic) direction and erupting from the edge plasma are observed using the Beam Emission Spectroscopy (BES) during an ELM crash. However, since the edge plasma can exist near the PBM stability boundary for a large part of the ELM cycle, a mechanism for the triggering of PBMs is needed. About 100 µs before the ELM crash the BES detects modes that grow near the top of the pedestal (Figure 4.11). The mode has frequency of about 20 kHz, poloidal wavelength of 6-10 cm (corresponds to a toroidal mode number $n = 40$). Radially they extend about 2 cm. These ELM precursors rotate in the opposite direction to the ELM filaments. Just before an ELM crash, this growing mode reduces the flow shear in the pedestal region allowing the PBM to grow into filaments exiting the plasma as an ELM.
Figure 4.1. An ELM precursor detected by BES at the top of the pedestal. Each frame shows a snapshot of the density perturbation in a 16x8 cm region near the plasma boundary. The time advances from top to bottom and from left to right. The dashed line shows the location of the last closed flux surface.

B The Core Effects on Pedestal Height

It is generally believed that the H-mode pedestal pressure affects strongly the core pressure through stiff profiles set by the turbulence. However, the opposite effect (core pressure affecting the pedestal) is less studied. In MAST double-null plasmas it is possible to prevent the LH-transition by keeping the plasma shifted up or down by a couple of cm from the balanced X-point configuration even when the plasma is heated well above the LH-threshold limit of a balanced plasma. When the plasma is moved to a balanced configuration, an immediate LH-transition is triggered. This feature allows conducting an experiment where $\beta_p$ (global ratio of pressure and plasma current) at the moment of the LH-transition of the plasma is varied by adjusting the start of the heating with respect to the LH-triggering.

During the period between the LH-transition and the first ELM, plasmas have equal conditions except that the core pressure of the plasmas that have had early heating in L-mode phase is higher than those with late heating. Not only does this lead to a difference global $\beta_p$ at the LH-transition but also in the pedestal top electron pressure before the first ELM is triggered. Figure 4.12 shows that the pedestal top electron pressure just before the first ELM is increased by 100% by an increase of 20% of global $\beta_p$ at the moment of the LH-transition.

Associated stability analysis shows that the stability limit for PBMs in the pedestal indeed increases as the core pressure is increased.
4.4 CORE: TRANSPORT AND CONFINEMENT

Predictive capability for future devices is developed by studying global (confinement) and local (transport) plasma dynamics. Transport of particles (including impurities) and energy are critical processes, and both have been investigated in terms of the dynamics of the main plasma components (electrons and deuterium), of heating (e.g. fast ions, see other sections) and of fuelling (e.g. pellets). In 2013 the emphasis has been on pellet fuelling physics, impurity transport (including helium, the ash of fusion reactions), detailed properties of turbulence as the plasma flow changes, some new data suggesting micro-tearing modes may have been detected on MAST and finally some initial studies on intrinsic rotation. As elsewhere, the material reported here is almost entirely from MAST and theory.

4.4.1 Particle transport

A Pellet fuelling in the presence of ELM mitigating RMPs

In ITER, the main fuelling and density control tool is likely to be the injection of frozen deuterium pellets. An important step in the development of adequate density control is to test the compatibility of pellet fuelling with ELM mitigation (see above). In MAST high field side pellet fuelling and ELM mitigation by RMP n=6 coils has been applied simultaneously to single null plasmas. The data are compared with the basic assumptions in ITER pellet fuelling model. Single pellet fuelling event has two distinct phases: pellet deposition and post pellet density evolution.

Concerning the first phase, pellet deposition, the MAST data have been collected with SND configuration to provide information about
the pellet cloud, which in turn is linked to plasmoids whose drift controls the fuel deposition. A fibroscope combined with a fast camera, both generously loaned by NIFS, Japan for a dedicated collaborative experiment in 2013, produces images of the pellet cloud in D\(\alpha\) and continuum spectra with 100\(\mu\)s resolution, in addition to fast camera images in visible light. The ratio of intensities of the Balmer D\(\alpha\) line and the continuum can be used to produce 2D images of the density and temperature of the pellet cloud. Such measurements in MAST show that the pellet cloud has a size \(~5\)cm with no strong elongation along the magnetic field lines. The cloud density and electron temperature can be deduced. These data can in future constrain the models for plasmoid formation which sets the initial conditions for plasmoid drift – a critical mechanism for ITER.

Concerning the post pellet density evolution the main assumption of ITER pellet fuelling model is that pellet fuelling and ELM mitigation by RMP fields are compatible. In MAST we have significantly extended the database of plasmas with simultaneous use of high field side pellets and ELM mitigation. An example of such a plasma is shown in figure 4.13. It is seen that RMP ELM mitigation is broadly preserved after pellets but the detailed picture is quite complicated. Analysis of a larger database shows that the size of first post pellet ELM is somewhat (\(~1.6\)x) larger than pre pellet ELM as measured by fast interferometer signal. Nevertheless the size of the post pellet ELM responds to ELM mitigation though some influence of pellet size could also play a role.

![Figure 4.13: Details of fuelling pellet with RMP-mitigated ELMs. Change in line integrated density \((\Delta n_L)\), D\(\alpha\) emission and RMP current.](image)

From figure 4.13 it is seen that the raw pellet retention time \(\tau^{*}\)pel is controlled by ELMs and by inter-ELM gas refuelling (not shown). As
gas fuelling will be inefficient on ITER the relevant ingredient of pellet retention time is the particle loss by ELM.

To investigate particle loss by a post pellet ELM we used pellet triggered Thomson scattering diagnostics. An example is shown in figure 4.14. Profiles of electron density loss during post-pellet ELMs show that the ELM-induced redistribution of the pellet fuel extends quite far into the plasma. This area encompasses the region of inverted density gradient indicating that convection rather than diffusion is responsible for core particle transport during the post-pellet ELM. To elucidate this mechanism further beam emission spectroscopy data on the peripheral turbulence are being analysed. This work shows how important it will be to ensure small (or no) ELMs for good fuelling efficiency.

Figure 4.14: Left column: Change of density profiles due to the pre- (blue) and post-pellet ELMs. Right column: Line integral density $n_eL$ and timing of profile measurements. Plasmas with weak (top row) and strong (lower row) ELM mitigation are compared. The drops in the density waveform are due to ELMs with the sudden rise due to the pellet.

**B Helium transport**

Helium ash (from the D-T fusion reaction) will need to be transported effectively from the core of the plasma in ITER and fusion power plants. During the last operational campaign on MAST, perturbative helium gas puff experiments have been performed to expand the experimental measurements of light impurity transport and to improve the quality of the measurement of the evolution of the impurity concentration.

Radial profiles of the diffusive, $D_{He}$, and convective, $V_{He}$, transport coefficients are shown in figure 4.15 and over-plotted are the neoclassical rates calculated by NCLASS. The shaded region near the plasma edge represents a region of higher uncertainty in the transport coefficients. For the plasma current scan in L-mode, a decrease in diffusion, $D_{He}$, and convection, $V_{He}$, is found at high
current. In H-mode, the \( \text{He}^{2+} \) density gradient becomes weaker due to the increase in \( \text{D}_{\text{He}} \) and decrease in the core \( \text{V}_{\text{He}} \). The \( \text{D}_{\text{He}} \) follows a similar trend to the effective total heat diffusivity (although \( \sim 0.25 \) less) during the L-mode plasma current scan. The same cannot be said for the confinement scan at high current as the heat diffusivity reduces in H-mode whereas \( \text{D}_{\text{He}} \) moderately increases. The L-mode plasmas are dominated by anomalous transport from mid-radius to the plasma edge; the H-mode plasma on the other hand is dominated by neoclassical transport up to the pedestal region.

Anomalous transport observed in L-mode on one flux surface \( (r/a=0.7) \) has been analysed using linear gyrokinetic simulations with the GS2 and GKW codes. A number of results from the linear gyrokinetic simulations suggest that trapped electron mode (TEM) turbulence is the driving the impurity transport. Firstly, the ion temperature gradient (ITG) modes are stabilised by equilibrium flow shear; further towards the plasma edge the equilibrium flow shear decreases and the ITG modes become significant. Secondly, the helium peaking factor calculated by GKW agrees in both magnitude and direction in the TEM region. The main contribution to the helium particle flux comes from the ion temperature gradient and thermo-diffusion coefficient. Lastly, the most significant difference between the micro-instability growth rate spectrum in L- and H- mode is the stabilisation of the TEMs in H-mode due to the lower gradient of electron density.

The results indicate that the most favourable scenario to transport the helium ash population originating in the plasma core of future fusion STs would be H-mode. Core convection is directed outwards in H-mode and therefore acts with the diffusivity to produce an outward particle flux. In L-mode, the trapping of helium in the core due to the inward convection is greatest at low plasma current, therefore a higher current is recommended to reduce the amount of helium trapped in the core. In conclusion, the density gradient plays a crucial role in the transport of light impurities in spherical tokamaks; a steep gradient causes an inward pinch of helium directly from neoclassical transport and indirectly from the particle flux associated with the TEMs.
4.4.2 Plasma Turbulence

A Influence of flow shear on ion-scale turbulence and heat flux

One of the central paradigms for improving tokamak performance by reducing energy transport is that flow shear acts to improve micro-stability and reduce turbulence levels, at least for modest shear. Intimately combined theory and experiments are now being used to explore how this really works. This work is collaboration primarily between CCFE, Oxford University, the Korea Advanced Institute of Science and Technology and the Wigner Research Centre for Physics, Budapest, Hungary.

Our previous work to characterise the ion-scale turbulence in MAST using data from the imaging BES diagnostic has been extended to study the influence of sheared equilibrium flow on the 2D structure of this turbulence and the resulting anomalous ion heat flux. Radially sheared toroidal flow, as driven by the torque from the heating beams, results in a perpendicular \((E \times B)\) component with shearing rate \(\gamma_E = B_\phi / B_\theta \frac{dU_r}{dr}\). This mean shear distorts the turbulent 'eddies', tilting the drift-wave vector \(k\) (which is initially aligned perpendicular to \(B\) within a flux surface) by an angle \(\theta = \tan^{-1}(k_r/k_\theta)\), resulting in a finite radial component \(k_r\). Theoretically, this angle is expected to scale as: \(\tan(\theta) \propto \gamma_E \tau_c\), where \(\tau_c\) is the correlation time of the turbulent eddies.
This can be understood intuitively – the longer that eddies exist the more they are sheared.

Figure 4.16 Examples of 2D measured (top) correlation functions from a MAST L-mode shot (r/a ~ 0.8) at times with differing levels of flow shear ($\gamma_E$ ~ 0.7 (left), 1.0 (middle) and -0.1 (right) x10$^5$ s$^{-1}$). Model fits are shown in the bottom row, resulting in tilts $\Theta$ ~ -24°, -37° and -10° respectively.

This tilt can be measured from 2D (radial/poloidal) correlation functions of the density turbulence measured by the BES system. To do this the 8×4 array is split into two 5×4 inner/outer sub-arrays, to reduce the radial coverage to ~ 10 cm, i.e. ~ 20% of the minor radius. Examples of such 2D correlation functions measured at the periphery of an L-mode plasma (r/a ~ 0.8) are shown in Fig. 4.16. Here, the tilt of the measured correlation functions is seen to increase with the prevailing shearing rate. The tilt can be quantified by fitting a model function which represents a poloidally oscillatory ($k_y$), decaying ‘wave-packet’ with radial and poloidal correlation lengths $l_{x,y}$ in a radially sheared coordinate system $\Delta y' = \Delta y + \tan \theta \Delta x$. Results of fitting such a function are shown in the bottom row of Fig. 4.16, which yield the $k_y$ parameters $k_{y0}$ and $l_{x,y}$ which characterise the turbulence.

Such fits have been performed for data from 1 ms intervals throughout ~50 MAST discharges and these parameters added to our database of turbulence and equilibrium parameters. This data can then be used to study the scaling of the turbulence characteristics.

In Fig. 4.17, the scaling of the tilt angle $\Theta$ with the parameter $\gamma_E T_e$ is shown for an L-mode discharge in which a resonant magnetic perturbation was applied to brake the plasma rotation and hence vary the flow shear. The eddy tilt is clearly seen to increase with the shearing parameter as predicted theoretically. The performance predictions for ITER generally assume gyro-Bohm-like transport coefficients, so here the measured heat flux is normalised to the gyro-Bohm value to see the relevance of any trends. An estimate of the turbulent ion heat flux, normalised to the gyro-Bohm level, can be obtained using the relation: $Q_{LBES} / Q_{GB} \sim k_y R (R/\rho_i)^2 (T_e/T_i) (\delta n/n)^2$, where $Q_{GB} = n T_i (\rho_i/R)$ and $\nu_{ri,i}$ is the ion thermal velocity.
4.22

Figure 4.17: The scaling of the measured tilt $\theta$ with the shearing parameter $\gamma E \tau_c$ for an L-mode discharge ($0.6 \leq r/a \leq 0.6$). The gyro-Bohm normalised ion heat flux $Q_{\text{BES}} / Q_{\text{GB}}$, which can be determined from the BES data alone, is shown by the colour scale.

A clear reduction in the heat flux $Q_{\text{BES}} / Q_{\text{GB}}$ with increasing shearing $\gamma E \tau_c$ and hence tilt $\theta$ of the turbulence is evident. This reduction in heat flux occurs largely due to a decrease in fluctuation level $\delta n/n$, although there is also a decrease in $k_y$ with increasing shear, which also contributes to the reduced heat flux due to a slower eddy turnover time. This reduction in turbulent amplitude and heat flux is consistent with linear theory of ITG-PVG driven turbulence (PVG: parallel-velocity gradient), which predicts a strong reduction of growth rate with increasing radial wave number $k_x = k_y \tan \theta$, i.e. increasing tilt, due to the onset of Landau damping.

B Cross-Polarization Doppler Backscattering and microtearing modes at the top of the MAST H-mode Pedestal

Microtearing modes (MTMs) have been predicted to be unstable at the top of the H-mode pedestal in MAST, playing a role in determining pedestal transport between Edge Localized Modes (ELMs), and therefore of pedestal structure and ELM stability. MTMs are electromagnetic instabilities that are expected to cause significant levels of magnetic field fluctuations. MTMs are very hard to detect or unambiguously identify against a background of other turbulence, but they do have a magnetic component, and this is predicted to cause polarisation changes to incident microwaves.

A novel diagnostic technique has been developed, combining Doppler Backscattering (DBS) and cross-polarization scattering (CP-DBS). This has enabled measurements sensitive to local, internal magnetic
field fluctuations at the radial location and wavenumber range where MTMs have been predicted to be unstable in MAST. Measurements were obtained in both standard DBS configuration for density fluctuations and in CP-DBS configuration for magnetic field fluctuations.

Figure 4.18 compares the total scattered power (time averaged in colour over-plotted on un-averaged in black) and Dα emission between one of the shots with standard DBS configuration and a later repeat with CP-DBS. Both measurements were at r/a≈0.94-0.96 at the end of inter-ELM periods. Both cases were at k⊥≈5-7 cm⁻¹ for scattering. This is in the range where MTM was predicted unstable in MAST. During the inter-ELM period, the received power for CP-DBS was about two orders of magnitude lower than in the standard configuration. There is also a difference in temporal behaviour. In 30150 the signal rises quickly then saturates during most inter-ELM periods, while in 30422 the signal slowly rises during the inter-ELM period until the crash. While evolution of the density profile also plays a role it cannot fully explain the difference in the observations, which is further evidence that two different fields are being measured. The results show a significant level of magnetic field fluctuations at the top of the pedestal and are consistent with predictions for unstable micro-tearing modes. Further analysis of the measurements are planned to be compared to simulations of the new experimental conditions.

Figure 4.18: Top: Total scattered power. Bottom: Dα emission.

4.4.3 Intrinsic Rotation

Rotation is known to play a critical role in stabilizing instabilities like resistive wall modes and influencing turbulence through ExB shear (see above). Neutral beams are not expected to be able to drive strong core toroidal rotation in fusion reactors as they do in current devices. However, tokamak plasmas rotate intrinsically even without neutral beams. Understanding this intrinsic rotation source is important for predicting rotation in future devices.
A sixteen channel Doppler Backscattering (DBS) system was installed on MAST during the 2013 campaign. This enabled measurements sensitive to the radial electric field, from which information about toroidal rotation can be inferred, even in Ohmic plasmas.

Figure 4.19 shows an example, where the measured Doppler shift indicates a change in sign of the core toroidal rotation, from counter-current rotation at high line-averaged density to co-current at low density. This is the first observation of intrinsic rotation reversals in a spherical tokamak, which had previously only been observed in large aspect ratio experiments. A database with good DBS data covering was obtained which also includes both asymmetric lower single null and balanced up-down symmetric plasmas, as well as both L-mode and H-mode plasmas. The intrinsic rotation reversals were observed to scale linearly with density as seen on some other tokamaks.

**Figure 4.19** DBS measured scattered electric field, plasma current, and density, showing an intrinsic rotation reversal in MAST between 300 ms and 400 ms.

### 4.5 CORE: FAST PARTICLE PHYSICS

The key long-term aim of the CCFE fast particle physics programme is to guide the achievement of high-performance burning plasmas with tolerable instabilities and transport of the fast ions, in particular
the fusion $\alpha$-particles needed to sustain burning thermonuclear plasmas. In pursuit of this goal, we aim to: interpret observations of fast particle behaviour in JET and MAST, and understand the mechanisms of energetic particle-driven instability and energetic particle transport; develop and employ links between energetic particle diagnostics and modelling, for example through the use of “synthetic diagnostic” data; validate existing numerical tools via the study of the most important experimental results, including $\alpha$-particle and three-dimensional effects; and predict energetic particle-driven instabilities in MAST Upgrade, JET deuterium-tritium plasmas and ITER, where possible taking into account nonlinear processes and identifying the most crucial effects for further study. The highlights of the fast particle programme in 2013 are summarised below.

4.5.1 FIDA/neutron camera measurements of fast ion transport in MAST

MAST is equipped with a four-channel scanning neutron camera and a fast-ion deuterium-alpha (FIDA) spectrometer, which provide complementary diagnostics of the evolving fast ion population. In the 2013 experimental campaign the use of these two diagnostics was carefully coordinated to yield a more comprehensive picture of fast ion transport resulting from various instabilities. These include fast particle-driven internal kink modes (fishbones), which are sufficiently reproducible that profiles of the neutron emission with high spatial and temporal resolution can now be obtained by using

![Figure 4.20: Colour contours: evolving radial profile of neutron camera count rates during quasi-periodic fishbone activity in MAST shot 29207 and subsequent repeat shots. White curve: root mean square Mirnov (dB/dt) coil signal from shot 29207; each peak of this curve corresponds to a fishbone.](image)
different lines of sight of the neutron camera in repeat shots (Figure 4.20). Modelling using the global transport analysis code TRANSP, reproduces the coarsest features of the fast particle data during periods of fishbone activity as long as \textit{ad hoc} anomalous diffusion of the fast ions is included, but the spectrally- and spatially-resolved FIDA measurements suggest that the distribution exhibits structure on a finer scale than can be accounted for by this model. Clear evidence has emerged that strongly-driven chirping toroidicity-induced Alfvén eigenmodes (TAEs) can also cause fast ion redistribution in MAST, particularly when modes with more than one toroidal mode number are excited simultaneously.

\subsection*{4.5.2 High frequency modes in MAST hydrogen/deuterium plasmas}

Alfvén cyclotron instabilities driven by fast ion energy gradients have been observed previously as compressional Alfvén eigenmodes (CAEs) in MAST. It is believed that such modes would be the dominant fast particle-driven instabilities in a spherical tokamak burning plasma and, when driven by beam ions, they may affect the efficiency of beam current drive by causing non-collisional pitch-angle scattering. However the properties of these modes are modified when two or more dominant ion species with unequal cyclotron frequencies are present, as in deuterium-tritium (DT) burning plasmas. In order to investigate the behaviour of CAEs in such cases, dedicated experiments were performed in MAST with low field side hydrogen gas puffing used to create otherwise similar plasmas with a range of hydrogen/deuterium (H/D) density ratios, from zero to more than 0.6. The H/D concentration was determined in the plasma edge from Balmer-\(\alpha\) line intensities of the two species, and in the core plasma using neutron detectors. It was observed that at increasing H/D concentration, Alfvén cyclotron instabilities are suppressed. The suppression effect is especially strong (the modes disappear completely) between the deuterium and hydrogen cyclotron frequencies, where ion-ion hybrid resonances can provide strong damping (Figure 4.21). A similar suppression effect of fast particle-driven instabilities in the tritium and deuterium cyclotron range is expected in DT plasmas.
4.5.3 Fusion proton measurements in MAST

In order to obtain as much information as possible on fast ion redistribution and loss, it is desirable to study these processes using several independent diagnostics, covering different regions of phase space. The low magnetic field of MAST, combined with the use of neutral beam injection, provides a unique opportunity for diagnosing the beam-thermal fusion reactivity profile using charged fusion products. These particles (mainly 3 MeV protons and 1 MeV tritons) typically leave the plasma after executing less than one Larmor orbit (see left frame of Figure 4.22), thereby allowing fusion reactivity profiles to be determined. This information can be used to study the temporal behaviour of the energetic beam ions. A prototype fusion proton detector provided by Florida International University has been used in MAST to study fast ion transport due to several types of MHD activity, including sawtooth crashes (right frame of Figure 4.23), ELMs and fishbones. The detector had four channels, corresponding to a range of birth positions along orbits intersecting the plasma midplane. Sufficient counts were recorded in these channels to provide a time resolution of 1ms, which is comparable to that provided by the MAST neutron camera, but with a much more compact detector (charged fusion products can be detected more efficiently than neutrons). The results provide a proof-of-principle that this type of diagnostic can be used to infer evolving fast ion profiles in spherical tokamaks.
4.5.4 TAE stability in ITER baseline scenario

TAEs driven by energetic ions in ITER could affect the confinement of those ions, and predictive modelling of TAE stability in this device is consequently a high priority of the fast particle programme. The use of relatively high beam energies in ITER (1MeV is planned) means that beam ions as well as fusion $\alpha$-particles will be sufficiently energetic to drive TAEs. In an analytical study of the ITER baseline scenario, two distinct regions of the plasma have been identified: a core region (where most fusion $\alpha$-particles will be confined) in which thermal ion Landau damping of TAEs has been found to exceed the total fast particle drive; and an external plasma region, with relatively few $\alpha$-particles but many gaps in the shear Alfvén continuum and hence the possibility of radially-extended TAEs. A local analysis indicates that the aggregate $\alpha$-particle and beam ion drive in this region may exceed the damping, although confirmation of this would require further (numerical) analysis. This work suggests that TAE-induced redistribution of fast particles will have only a minor effect on the fusion burning process, although it is possible that there could be significant consequences of TAE-induced fast particle losses for the first wall. Beams will be applied throughout an ITER pulse, but ion Landau damping (likely to be the dominant damping process) will be negligible at low ion temperature, i.e. during the initial heating phase. Thus in order to evaluate the likely effects of TAEs in ITER it is important to consider the path to the baseline scenario as well as the scenario itself.

4.6 SCRAPE-OFF LAYER AND DIVERTOR
The exhaust of power and particles is one of the major challenges to be met before a DEMO-class tokamak can be finalised. Operation of high performance plasmas on ITER while remaining within the thermal limits of the divertor plasma facing components will provide one of the more demanding tasks for ITER, and the attention given to this on JET via the ITER-like Wall is a consequence. In reflection of these factors the CCFE exhaust physics programme is being intensified, with a focus on deeper and usable understanding of the mechanisms and physics involved, and of course the major investment in the advanced divertor configurations in MAST Upgrade.

The topics addressed here include imaging of the static and turbulent structures and flows in the scrape-off layer (SOL) and divertor; scaling of the SOL thickness; modelling of filaments and the SOL stability, modelling of detachment in the new super-X configurations to be studied with the upgrade, and some new advanced diagnostic techniques.

4.6.1 Tungsten Melt Experiments on JET

A sequence of experiments were conducted in 2013 with a specially design divertor module with deliberately raised tungsten lamellae on a not-normally used part of the divertor target. The idea was to explore the heating and melting by ELMs at these raised edges, to inform the decision whether to install a tungsten divertor on ITER from the outset, and as input to the detailed design. Figure 4.23 shows the special lamellae and the results of the melt experiment. The measured temperature rise and the movement of the molten tungsten are now being compared with models.
4.6.2 FAST DIVERTOR AND SOL IMAGING

Previous research on MAST has showed that intermittent transport of particles due to filaments play an important role in determining the distribution of power and particle fluxes to plasma-facing surfaces in the main chamber and divertor. An unfiltered fast camera loaned from the EPSRC was used to look at filament propagation in the main chamber and divertor with unprecedented spatial coverage and temporal resolution. Fluctuations in the divertor below the X-point were imaged at 120kHz to investigate the effect of the strong magnetic shear in this region on the structure and propagation of filaments. Fig 4.24 shows examples of the data collected from an L-mode and H-mode period from the same discharge.
Figure 4.24: Raw image of filaments in the divertor in L-mode (left) and background subtracted images in L-mode (centre) and H-mode (right) from shot 29564. The centre column and P2 coil armour are highlighted in red.

The camera data suggest that filaments in the SOL reach the divertor in L-mode and between ELMs in H-mode, although there are far fewer filaments visible during the H-mode phase. There is also evidence for filaments in the private flux region propagating from the x-point, along the inner leg toward the inner divertor target. Direct observation of filaments in the private flux region has not been reported previously, and could provide an explanation for divertor heat flux at the inner target being similar both in the open (SOL) region and the private flux region. An experiment conducted to record filtered imaging data of ELMs in the divertor will be used by colleagues at ITER to test models of ELM propagation to plasma-facing surfaces.

High-speed imaging at the mid-plane at 100kHz has provided data on the propagation of filaments and ELMs in the main chamber SOL, see 4.25. This data will be used together with imaging of the divertor to develop models for how filaments influence power deposition at the divertor strike point.

Figure 4.25: High-speed camera image of a single-null plasma prior to (left) and during (right) an ELM event.

4.6.3 Scrape off layer flows
Flows in the divertor and SOL are important for the transport of energy, particle and impurities (and for retention of particles and impurity in the divertor chamber). Measurements help both to develop an empirical understanding and to test and develop models where handling flows has traditionally proved demanding.

A new ‘coherence imaging’ diagnostic, developed with Durham University and the Australian National University, has been used to image impurity ion flow in the MAST divertor and scrape-off-layer. Using Doppler shifts of visible emission lines, this diagnostic can provide images of line-of-sight averaged impurity flows over a 40° field of view, with flow resolution typically between 1 – 4 km/s. It was run routinely throughout the 2013 campaign on MAST and, in particular, in dedicated flow imaging experiments. Fig 4.26 shows examples of flow and intensity images for He II in the main chamber SOL, in an experiment investigating the effects of the connection length between the midplane and the divertor targets on SOL flows. The stagnation point on the low field side, which is indicated by stars on the images, is observed to move poloidally downwards as the magnetic axis of the plasma is shifted vertically upwards. This behaviour is in broad agreement with OSM-EIRENE modelling, suggesting the stagnation point moves in response to the change in the connection length of the field lines in the SOL and recycling sources at the divertor strike points.

![Figure 4.26: Flow and intensity images for He II in the main SOL (the centre column is visible on the extreme right of the images). The stagnation point on the LFS SOL (indicated by stars) moves poloidally as the magnetic axis is shifted upwards (images from left to right).](image)

Fast imaging of divertor flows with frame rates up to 1 kHz has also been achieved. Fig 4.27 shows wide angle images of C III emission and line-average flow in the MAST divertor during an H-Mode LSND plasma immediately before, coincident with and following an ELM. Flow reversal in the high field side divertor leg near the target is clearly seen coinciding with the ELM crash: the impurity flows towards the divertor target in inter-ELM periods but away from the target at the time of the ELM crash. A range of other measurements undertaken with this diagnostic include divertor flows in the presence of RMPs.
For divertor measurements, the wide field of view and high spatial resolution of this diagnostic has allowed tomographic inversion of the line-average data to obtain poloidal profiles of the impurity parallel flow (assuming predominantly parallel flow and toroidal symmetry). Examples of recovered divertor flow and emissivity profiles for C III are shown in Fig 4.28, for L mode and H mode phases of the same LSND plasma. The direction of the impurity flow: towards both inner and outer divertor targets, and the increased flow speeds in H mode over L mode, have been observed in multiple impurity species and in most discharges. 2D images such as these are well suited for comparisons with both interpretive and predictive modelling.
4.6.4 INTER-ELM SCRAPE OFF LAYER FALL OFF LENGTHS

The length over which the power leaving the plasma falls away is a key issue for ITER as it sets the power load that will be deposited onto the material surface inside. The fall off length is seen to vary as a function of the plasma current, density and input power on several devices. The heat flux width can be measured via infrared (IR) thermography for a range of different plasma parameters. The widths, along with the plasma parameters at the time that they were measured can then be regressed to give a multiple plasma parameter expression, known as a scaling, for how the width varies across the operating space of the tokamak.

Measurements of the heat flux width have been performed in MAST plasmas with currents between 0.45 and 0.92 MA, with a 50% variation in density and five fold range in input power. The measurements allow the heat flux width to be determined via fitting. The fitted widths have then been regressed in order to relate the fall off length to the plasma parameters:

\[ \lambda_q [\text{mm}] = 4.57(\pm0.54)I_p^{-0.44(\pm0.15)}P_{\text{SOL}}^{0.12(\pm0.08)} \]

Where \( \lambda_q \) is the fall off length, \( I_p \) is the plasma current and PSOL is the input power.

The scaling shows that the plasma current is the dominant factor in setting the width of the strike point at the divertor. The density does not have a strong influence on the fall off length and has been excluded from the final fit. Figure shows the fall off length plotted as a function of the plasma current. The data shows that the fall off length decreases with increased plasma current, which is consistent with results from other tokamaks, such as ASDEX Upgrade, DIII-D, JET and NSTX.
4.6.5 MEASUREMENTS OF THE PLASMA POTENTIAL

The plasma potential plays an important role in determining the SOL turbulence characteristics, both directly through fluctuations and indirectly through the sheared electric field and thus the sheared flow that exists near to the separatrix. Measurement of the plasma potential is challenging due to the tendency of a conductor to charge up to the floating potential when inserted into a plasma. The ball pen probe (BPP), originally developed by Adamek et.al (Institute for Plasma Physics, Czech Republic) is able to measure a potential which is a close approximation to the true plasma potential. In collaboration with J. Adamek during 2013 a ball pen probe has been developed for MAST as a modification of the Gundestrup probe. Figure 4.30a shows the plasma potential profile obtained from a deep reciprocation of the BPP compared to the floating potential measurement with a floating Langmuir probe (LP) in a 400kA connected double null Ohmic L-mode plasma (#28819). As can be seen the BPP signal is not simply scaled version of the floating potential. The electron temperature ($T_e$) can be extracted by combining the BPP and LP potential measurements. Figure 4.30b compares the $T_e$ measured by the BPP with the that from the Thomson scattering diagnostic. The comparison is favourable, which suggests that the BPP is making a measurement close to the plasma potential. Using this as a best estimate for the plasma potential, the radial electric field has been estimated and is shown in Fig 4.30c.
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Figure 4.30 a) Plasma (BPP) and floating potential profile measured with the BPP and an LP in shot 28819 showing differences in shape and polarity between the two signals. b) Comparison of electron temperature measured by a BPP and by the TS system showing good agreement between the two diagnostics. c) Radial electric field estimated by fitting a 15th degree polynomial to the BPP profile and taking the derivative with respect to the $R - R_{LCFS}$

4.6.6 3D PLASMA FILAMENTS DYNAMICS IN THE SCRAPE OFF LAYER WITH BOUT++

The scrape off layer (SOL) of magnetic confinement devices is an inherently turbulent region. Both in L- and H-mode, anomalous transport of particles and energy perpendicular to the magnetic field is dominated by the propagation of coherent field aligned filamentary structures. Understanding their dynamics is essential for optimising the exhaust design of future fusion devices such as DEMO, as they determine, among other relevant SOL features, SOL widths and particle and heat fluxes to the first wall.

Using the BOUT++ framework, a non-linear 3D reduced fluid code for filament and turbulence studies has been developed. A number of benchmark studies were performed in preparation for full turbulence simulations. Analytic equilibrium solutions for the SOL equilibrium in the absence of curvature were obtained and compared with the code’s results. The numerical evolution of the parallel dynamics of isolated filaments moving on these steady state solutions were shown to be correctly reproduced by the Rankine-Hugoniot shock wave relations. The transient physics was further benchmarked with the existing 1D and 3D codes, SOLF1D and TOKAM-3X, as complete theoretical solutions are not available (see Fig 4.31). Apart from detailed aspects which differed due to differences in the numerical schemes, the expansion of the filament was found similar in all the
codes, both concerning the amplitude of the wave and its velocity. The BOUT++ simulation included electron inertia, and allowed for a differential response of the parallel ion and electron velocities, which can lead to large parallel currents (closed by polarisation currents). Such currents were transiently observed if the filament was sufficiently shaped in the parallel direction. A systematic numerical campaign was performed, in which the propagation of filaments with varying initial amplitude, parallel extension, and distance to the sheath was simulated. Also, different numerical schemes were studied in order to reduce the numerical noise and avoid unphysical instabilities. It was found that both hyperviscosity and staggered grid allowed reliable solutions in the parallel direction.

![Figure 4.31](image)

**Figure 4.31** Comparison between the BOUT++, SOLF1D and TOKAM-3X simulations of the plasma density and ion velocity during parallel evolution of a filament (there is a Mach 1 boundary condition at the sheath at the top of the plots). The fields are plotted against the parallel coordinate (vertical axis) and time (horizontal axis).

Using the BOUT++ framework a flux tube geometry has been created which closely represents the magnetic geometry of the MAST SOL. Seeded filament simulations have been performed in this geometry under varying conditions of electron temperature and density. A transition has been found to occur between a regime where the advective motion of the filament is dominated by classic 2D interchange behaviour, to a regime dominated by the inherently 3D Boltzmann response. In the interchange regime strong radial fluxes are observed, which are suppressed in the Boltzmann regime by a
local spinning motion of the filament. Figure 4.32 a and b show the cross-sectional profiles at the midplane of a filament in the interchange and Boltzmann regime respectively. Shown in Figure 4.3c is the volume averaged radial flux arising from simulations over a temperature density parameter space, clearly showing the transition from the interchange to Boltzmann regime. Such a transition may explain the phenomenon of increased advective losses as edge density is increased. It may also explain the stationary phase of motion observed in inter-ELM filaments on MAST.

**Figure 4.32** a) Electrostatic potential (colour) and density (contours) of a filament cross-section in the interchange regime. The filament develops the potential lobe structure associated with the 2D advective motion commonly seen in slab simulations. b) Filament cross-section in the Boltzmann regime. The potential now contains a dipole component, as well as a monopole at the filament centre. This causes the filament to spin and suppresses its radial motion. The effect arises due to inhomogenous drive for the motion along the length of the filament, which causes the formation of parallel density gradients which act as a source of potential. c) Volume averaged radial flux (normalized by $n_0 T_e$) showing a transition from the interchange regime at low temperatures to the Boltzmann regime at higher temperatures.

4.6.7 Ballooning Treatment of the Intrinsic Scrape Off Layer Instabilities for Standard and Advanced Divertors

Next generation tokamaks will have to operate under the stringent constraint posed by the interaction between plasma and solid surfaces. Current experimental extrapolations for the Scrape Off Layer (SOL) width in ITER predict a 1mm thickness at the outer midplane (in inter-ELM phase at low collisionality). Such sharp gradients might induce instabilities in the SOL, which could induce perpendicular turbulent transport and a consequent flattening of the SOL profiles.

Intrinsic SOL instabilities have been studied using a ballooning formalism which incorporates the appropriate sheath boundary conditions at the target. The linear growth rate of the modes, their structure and the associated diffusion coefficient were obtained. The latter were estimated using a simple $\gamma/k^2$ approach for the fastest growing mode. The model used includes curvature and sheath drives, finite Larmor radius effects, resistivity and partial line tying at the target. The magnetic geometry was obtained using current carrying...
wires, representing the plasma current and the divertor coils, and naturally generates X-point geometry and magnetic shear effects.

The calculation allows a comparison of the turbulence and anomalous transport between standard and advanced divertor configurations (Snowflake and Super-X). It was found, for example, that in the low collisionality regime the fastest growing mode in the Snowflake divertor has a larger growth rate with respect to similar Single Null configurations. In addition, the spatial variation of the perpendicular wave number is such that the estimated diffusion coefficient is roughly twice as large for the Snowflake than for the Single Null geometry in the upstream region, while it is smaller (1/4th) in the divertor leg, i.e. beyond the X-point (figure 4.33). This effect is due to the inclusion of realistic X-point geometry, the strong magnetic shear of which determines the reduction of the turbulent structures and of the transport associated with them. As the SOL width is directly related to the anomalous perpendicular transport, this result could indicate how different magnetic geometries affect the wetted area at the divertor.

![Figure 4.33](image)

*Figure 4.33 Top: growth rate of the fastest growing mode as a function of the toroidal mode number, N, for Snowflake and Standard divertors. Bottom: measure of the anomalous diffusion as a function of N*

4.6.8 **Fluid modelling of the SOL**

SOLPS has been used to study the effect of the Super-X divertor on the transition to detachment and the distribution of volumetric power losses in the divertor. Two divertor geometries are considered – MAST Upgrade with a conventional divertor (CD) and with a Super-X divertor (SXD). The transition to detachment is shown in the Figure 4.34 (1) and (2) as the roll-over of the particle flux and the pressure drop for two different input powers to the SOL (a) $P_{\text{inp}} = 1.7\text{MW}$ and (b) $P_{\text{inp}} = 0.85\text{MW}$. Detachment is achieved much more readily in the super-X configuration. In Figure 4.34 (3), the reduction of the target power load $Q_t$ between CD and SXD is shown as a function of the separatrix density. At the lowest collisionality, the reduction is comparable with the effect of magnetic flux expansion. At higher
collisionalities, the importance of volumetric power losses increases and the $Q_t$ reduction reaches a factor of $\sim 100$ when detached.

![Figure 4.34](image)

(1) The target particle flux in CD and SXD as a function of the separatrix density. (2) The pressure drop between the midplane and the target. (3) The reduction of the peak target power load in SXD with respect to CD and compared with magnetic flux expansion. Two cases (a) and (b) are characterized by different input power to the SOL, (a) $P_{\text{inp}} = 1.7\text{MW}$ and (b) $P_{\text{inp}} = 0.85\text{MW}$.

The distribution of the volumetric power losses has been studied for different collisionality regimes. The radiation zone separates from the target and moves upstream in the divertor leg during detachment (see 6). The power loss region broadens radially with increasing density and expands along the field line in SXD with respect to CD. This also means that the radiation volume and the total radiated power increases in SXD.

![Figure 4.35](image)

Figure 4.35 The distribution of the volumetric power loss in MAST super-x configuration in detachment.

4.7 THEORY AND MODELLING - IMPROVING SIMULATION AND DATA MANAGEMENT CAPABILITIES
Much of the theoretical work within the CCFE programme is reported in the above sections. Here we report specific code development work. While much of this is specific to Tokamak Science, some has wider applicability, e.g. to improving CCFE’s materials modelling and neutronics capabilities described in other Chapters.

Last year we reported on opportunities being explored to enhance CCFE’s high performance and future computing capabilities. This work has continued and expanded in 2013, to a position where we will launch in early 2014 a number of major code and data management projects. The main aim of this initiative is to tap into other areas of “big science”, to exploit for fusion best practice in high performance computing and management of scientific data. This means we have interacted with many external organisations, most notably universities, the UK Science and Technology Research Council (STFC) and specialist industry. As well as improving scientific output, better scientific data management is needed to ensure CCFE complies with open access requirements.

There have been too many developments to describe all these in detail; instead we give a few examples.

4.7.1 Simulation Codes

The fast particle tracking code LOCUST uses graphical processor units (GPUs) to accelerate the computation of very large numbers of fast ion orbits, thereby making it possible to generate smooth distribution functions suitable for a number of purposes including calculating instabilities. During 2013, new GPU hardware was installed, at very modest cost, with large improvements in performance. The use of GPUs has made LOCUST the most advanced code of its type in the world and recently CCFE has been awarded a contract to use it as the basis for developing ITER’s fast ion simulation capability. When tracking particle orbits for very long times it is essential that their dynamics is treated as accurately as possible, and in 2013 CCFE started a collaboration PPPL (Princeton, US) to port their variational symplectic integrator platform to LOCUST-GPU. IPP Greifswald (Germany) is also involved as IPP is interested in using LOCUST-GPU as part of its stellarator DEMO design work.

To ensure accurate simulations, LOCUST and other codes (e.g. neutronics) need to use highly sophisticated geometries to allow for complicated 3D details of ITER, MAST and other tokamaks. During the year CCFE started a project to improve the transfer of CAD engineering information to meshes uses in physics simulations for application in LOCUST and other codes (e.g. neutronics). This includes testing commercially available software. Figure 4.36 shows the level of detail being modelled in typical ITER simulations. Existing codes typically approximate the vessel in two dimensions, whereas in
contrast, LOCUST is now capable of modelling power loading to plasma facing components in the presence of 3D field structure down to cm level accuracy in full 3D geometry.

![LOCUST mesh of the ITER plasma facing components used in power loading calculations, partly cut-away and rendered in Paraview.](image)

Fig. 4.36: LOCUST mesh of the ITER plasma facing components used in power loading calculations, partly cut-away and rendered in Paraview.

Many tokamak simulation codes are very computationally intensive, especially if they track plasma behaviour for long periods of time. Faster algorithms for time-advancement are therefore of great interest. In 2013, CCFE recruited an expert in the “Parareal” technique (she was previously a Monaco Fellow at ITER). While much of her work is focused on helping to improve gyrokinetic plasma turbulence simulations (especially the GS2 code – used for some of the research reported in section 4.4), she was asked by EFDA to see whether Parareal could give significant improvements in the run-time of SOLPS, a code widely used to model the Scrape-Off Layer at the edge of tokamak plasmas. This work was very successful resulting with the time-advancement aspects of SOLPS being improved by around an order of magnitude. The project so far has involved installation of IPS-Parareal on EFDA’s ITM-gateway, the building of various components (fine and coarse options, Parareal correction, convergence checks etc.), the running of various cases and detailed analysis of the results. This EFDA task is continuing in 2014 under EUROFusion.

### 4.7.2 DATA ACCESS AND MANAGEMENT

CCFE’s powerful and flexible data access software IDAM is mainly used on MAST. However, it has been installed at JAEA Naka, Japan for evaluation as part of the F4E Broader Approach procurement arrangements for the Remote Experimentation Centre (REC) for JT-60SA at Rokkasho. IDAM is a candidate platform for providing access
from and to the REC for data analysis, visualisation, modelling and machine operations. Currently the installed IDAM server at Naka provides access to MAST data and JT60 legacy data. IDAM will be compared with other options, e.g. MDSPlus Testing will involve some additional developments of the IDAM framework, particularly to overcome the inherent data throughput limitations of high bandwidth, high latency networks. These developments (parallel TCP/IP and UDT/IP data transport modules) are also very ITER relevant.

CCFE has started an initiative to improve its central CCFE storage system for retention and access of MAST modelling data, partly driven by Research Council requirements for long-time availability of data that underlies scientific publications. As a first case, a set of scripts to record and track the provenance of the MAST efit++ equilibrium reconstruction is undergoing testing. The scripts generate a pulse dependent input file set and run the code accordingly within the prescribed workflow (e.g. defined physics constraints, high time resolution, serial/parallel, etc…). Provenance metadata are collected during the run and will be stored in a specific IDAM provenance metadata catalogue. The code and scripts are being installed in the MAST scheduler for testing purposes and for reprocessing M9 campaign data.

4.8 JET TECHNICAL DEVELOPMENTS

CCFE is leading two enhancement projects: one ("CDT") to explore ways in which camera views of the inside of the torus can be made available in the planned DT campaign, and the other ("DMV2") to install a second disruption mitigation valve onto JET for greater resilience especially during the DT campaign.

a) CDT: This project is a feasibility study to see which cameras can be made useable in DT operation (which “whites out” or irreversibly damages cameras which are close or with direct lines of sight). The best solution is to move the cameras away and shield them with concrete from direct neutron flux. It is not practical to do this for all the 25 cameras in use in 2013, so three likely systems have been identified: a wide angle IR view of the main vessel (KL7) to give the Be wall temperature; a wide angle visible view of the main vessel (KL14) and an IR view of the outer tungsten divertor target (KL9B). All 3 systems are expected to be moved to behind the main shield wall (e.g. to the roof lab).

Substantial work has been done on the optical design to explore the general feasibility (which is challenging). A conceptual design report is in preparation as the basis of the necessary follow-up project (CDT2) which would implement whatever is decided to be feasible.

b) DMV2: the project is to install a second disruption mitigation valve for massive gas injection, with a shorter gas delivery tube for faster
response time. This has various purposes: testing the effect of a different location (DMV2 is on a horizontal not vertical port); providing resilience: at least one DMV is a requirement for high current operation, especially during DT and related to the latter, providing a tritium compatible valve. Finally it will allow more sophisticated disruption mitigation experiments e.g. with different gases injected in sequence to try to quench the plasma and avoid runaway electron production reliably.

The installation on the torus was completed in March 2014 (after the reporting year), and the systems is planned to be commissioned in the restart in April/May 2014 for use in the 2014 campaigns. All the equipment, especially the high pressure gas injection valve and the modified all-metal gate valve and the all-metal sealed gas handling system is designed to allow operation with tritium plasmas.