4 JET Studies

4.1 OVERVIEW

2008/09 has been one of the most intensive periods of operation on JET, exploiting the highest heating power and best plasma measurement capability yet. This will continue up to a major shutdown starting towards the end of 2009 to install new plasma facing components (an ITER-like Wall – ILW), a major upgrade to the neutral beam system and a range of improvements to the plasma measurement systems (diagnostics). There has been an increased focus on issues directly relating to the finalisation of the design of ITER components, and the understanding needed for detailed design of ITER plasma scenarios and determining the operational capability. This will also contribute to the design of DEMO. Some of the results and improvement projects are described below, with the emphasis on work where UKAEA has had a strong contribution as an Association. UKAEA's role as operator of JET is described in Chapter 3.

On the tokamak physics there has been notable progress in the following areas (often with key input from theory and modelling work, Chapter 6):

- the properties of the plasma pedestal, which has a strong influence on the core temperature and density in ITER and hence the fusion power;
- methods to reduce the power flowing into the divertor and to reduce the temperature at the plasma edge, to reduce the very challenging demands on the plasma facing components in the ITER divertor and indeed on JET when the ILW is installed;
- the impact of Edge Localised Modes (ELMs) on the first wall and the divertor, and on techniques to reduce this, again related to the lifetime of the first wall and divertor components on ITER and DEMO;
- substantial performance increases in the so-called 'advanced tokamak' regimes, which are being developed to allow very long pulse or steady state in ITER/DEMO;
- significant improvements in the understanding of the stability at high pressure (resistive wall modes, sawteeth), especially the role of fast particles, and how to test stability non-destructively;
- improved empirical understanding of the confinement behaviour, in particular the dependence on plasma pressure (β), a source of uncertainty in recent years which affects the performance range on ITER and DEMO.

Much of the more detailed analysis will be done in the 2009/10 shutdown.

There have been several projects running, and highlights include:

- achievement of the technical objectives of the ITER-like ICRH antenna (ILA) – dynamic matching of the impedance, operation in ELMy plasmas and attainment of the target voltage on the antenna straps. The power coupled is limited by the plasma coupling impedance which is lower than expected, so the power target has not been reached. These results are key for the design of the ITER launcher (Chapter 8);
- demonstration of three ELM-resilient ICRH coupling schemes using an 'internal' conjugate T (ILA), external conjugate T and 3 dB coupler, all important to the JET programme (higher injected power) and the design and use of ITER systems;
- delivery and commissioning of all four new high-voltage power supplies for the neutral beam upgrade;

- delivery of a new 60 MVA power supply to improve the vertical position control in the presence of ELMs to allow higher current, higher performance operation of JET;
- development of an improved plasma position and shape controller, more resilient to ELMs and other disturbances;
- completion of an upgrade to the edge LIDAR Thomson scattering diagnostic, using ITER-relevant detector technology (Chapter 8), and good progress on other diagnostics projects.

4.2 INTRODUCTION

The 2008 JET work programme was agreed by EFDA in 2007 and was built under three ‘headlines’, in order of priority:

- high-level commissioning and full scientific exploitation of new systems and issues that could impact on the design of ITER components;
- qualification of integrated operating scenarios for ITER;
- physics issues essential to the efficient exploitation of ITER.

These covered the campaigns C20-C25 in 2008, and C26 in the first part of 2009.

The JET task forces for this period were:

S1	ITER baseline scenarios
S2	Advanced scenarios, towards long pulse and steady state
E*	Exhaust
M	Stability, MHD
T	Transport
D	Diagnostics – hardware systems and physics, interpretation
H*	Heating and current drive systems and physics
DT*	Preparation and issues for tritium and D-T operation
FT*	Fusion technology using JET capabilities and features

** UKAEA has leader/deputy*

The rest of this short summary of the UKAEA work on JET in 2008/09 is structured as follows:

- contributions to the baseline scenario (ELMy H-mode), notably the pedestal, ELMs, confinement and divertor compatibility;
- contributions to advanced tokamak scenario development, especially performance, stability, effects of toroidal-field ripple, formation of transport barriers and ‘integration’ issues (compatibility with the divertor and heating and current drive methods);
- preparations for the new phase with the ITER-like Wall, with its constraints on plasma regimes (very relevant experience for ITER);
- specific physics developments, e.g. fast ion effects, disruptions;
- improvements in the heating systems, plasma control and diagnostics;
- JET fusion technology work is covered in Chapter 7;
- finally a short description of future plans for JET.

The scientific work picked is not UKAEA work alone – the collaborative nature of the exploitation of JET means that seldom is any Association the only or indeed the dominant contributor. The work selected here generally has had presentations or papers led by UKAEA staff, or experiments where they were Scientific Coordinators.

4.3 BASELINE SCENARIOS FOR ITER – ELMy H-MODE

The highest fusion power from ITER is presently expected to come from plasmas in the well-established ELMy H-mode regime, i.e. plasmas with an edge transport barrier, a high pedestal, and turbulent transport in the core region. The precise performance that will be achieved depends on many factors, and a combination of empirical scalings and a detailed first-principles understanding will be required for the ITER experiments to be designed efficiently, in order to achieve the objectives as quickly as possible. The factors include:

- height and stability of the edge pedestal, and its formation criteria (L-mode to H-mode transition);
- the properties of ELMs, and the heat pulses and filaments associated with them;
- plasmas compatible with the divertor and first wall, e.g. suitable density and temperature at the plasma separatrix, which set the boundary conditions for the scrape-off layer and divertor which in turn have a major influence on the attainable power and plasma temperature at the divertor target plates. This last topic is also relevant for the advanced scenarios and so will be in a separate section.

4.3.1 EDGE PEDESTAL

H-mode and some other scenarios (such as the hybrid) are characterised by an edge transport barrier (low transport regime), which leads to a ‘pedestal’ in the temperature and density. This pedestal acts as a boundary condition for the interior profiles, and thus has a major influence on the core temperature and density and thus the fusion power. The high gradients also tend to trigger instabilities, the most important being the ELM which can deposit substantial energy on the first wall and divertor in short pulses, presenting a significant design constraint on ITER. New diagnostic capabilities on JET, in particular the High Resolution Thomson Scattering system (HRTS), have enabled much more precise measurements and experiments. There are four main lines of the research:

- scaling experiments on JET to determine the dependence on plasma parameters such as pressure, normalised ion Larmor radius normalised to the plasma size and collisionality (β, $\rho*$, $\nu*$). See Figure 4.1;
- identity experiments with other tokamaks, especially C-Mod and DIII-D in the US, and ASDEX Upgrade in Germany, in particular to identify the relative roles of plasma and non-plasma physics effects. Figures 4.2, 4.3;
- modelling of the physics giving rise to the pedestal, focused on fuelling and the density pedestal this year (Figure 4.4);
- stability calculations (presently primarily ideal-MHD, peeling-ballooning modes in particular). See Chapter 6.

These comparison experiments are very important in separating effects peculiar to a particular machine from common underlying physics which can be transported to ITER and DEMO. The figures below are to illustrate the types of measurement and, in the comparison with C-Mod (Figure 4.4), the detailed modelling that is needed to start to understand the role of neutrals (atoms and molecules) – these will be dependent on the gas fuelling of the specific tokamak.

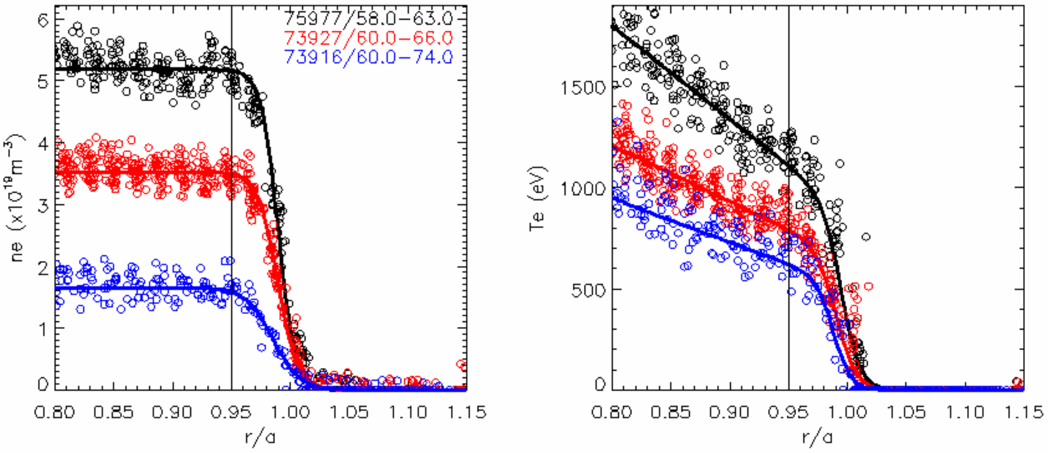

Figure 4.1: *Scaling experiments in JET – β, q, ν^* matched between different plasmas at different I_p, B_T (allowing a ρ^* scan). Both T_e and n_e pedestals appear unchanged in r/a. There was also no discernible dependence on ν^*,in a limited scan*

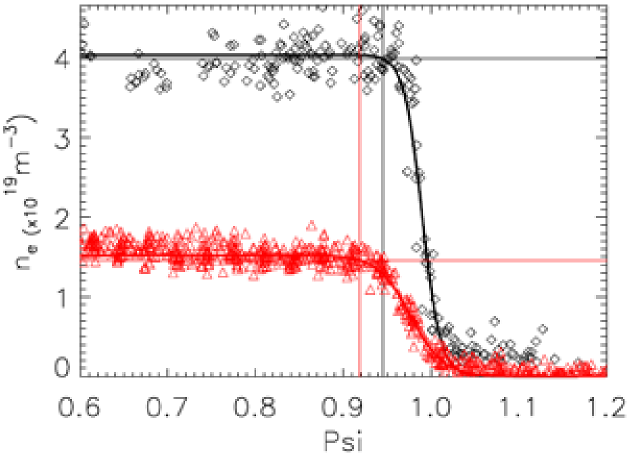

Figure 4.2: *Pedestal identity experiments between JET (lower curve, red) and DIII-D. Psi~$(r/a)^2$*

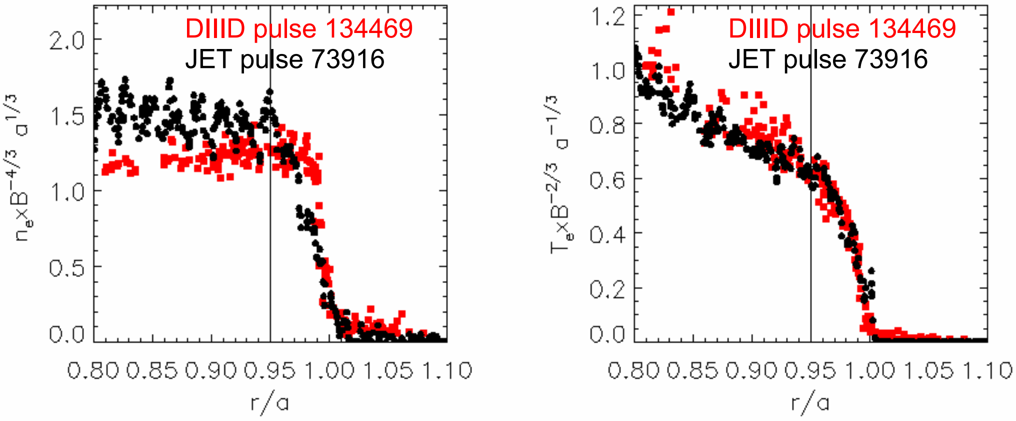

Figure 4.3: *Pedestals scaled according to plasma physics rules between JET (black circles) and DIII-D, showing that temperature pedestal seems to be identical in r/a, but the density pedestal is narrower in DIII-D than in JET, suggesting a role of neutral penetration (cf. C-Mod below)*

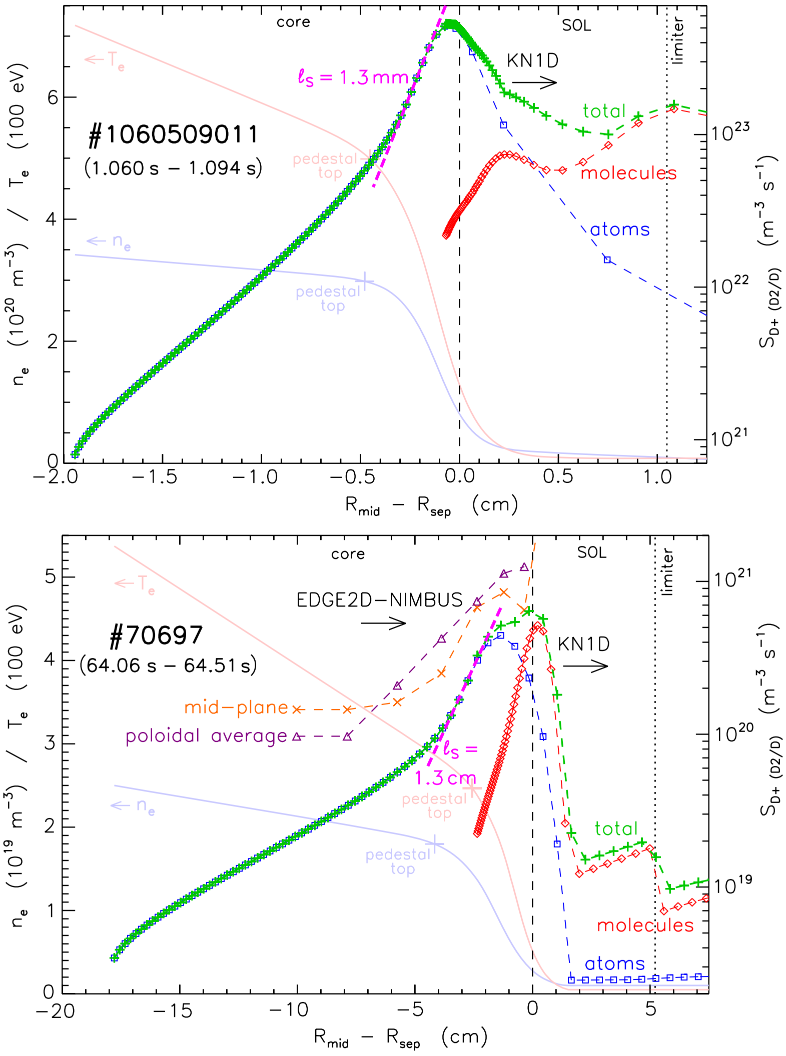

***Figure 4.4:** Edge ionisation source rates (right-hand scales) calculated with the 1-D KN1D code for (a) C-Mod and (b) JET shots with non-dimensionally identical pedestal tops. Components directly from molecular dissociative ionisations and from atoms, plus their sum, are shown. Source decay-lengths in the pedestal are indicated by heavy dashed lines. Respective plasma profiles (left-hand scales), on the same spatial coordinates centred on the separatrices, are shown as faint lines. Also superimposed for JET: source rates within the pedestal, at the mid-plane and poloidally-averaged around flux surfaces, computed with the 2-D EDGE2DNIMBUS code suite*

The pedestal is closely related to the criteria for access to and exit from H-mode which are important for ITER as they relate to the quantity of installed heating power. Activities were described in the 2007/08 Annual Report and expanded for an invited talk at the 2008 European Physical Society conference on Plasma Physics.

4.3.2 ELMs – UNDERSTANDING, EFFECTS AND MITIGATION

The two aspects to address are the trigger and drive which initiate the instability and then the mechanism whereby the heat and particles reach the plasma facing components. On the first, there is now wide acceptance that the largest ELMs (Type I) are a consequence of the pressure and current–driven peeling-ballooning mode being driven unstable (although there are some questions outstanding). The emphasis here is on the second aspect. Detailed measurements (Figure 4.5) have been made of the edge plasma just around the time of an ELM, again using the new high resolution Thomson scattering (TS) system, with careful data collection selecting TS profiles taken just before or after

ELMs, given there is no real-time synchronisation so far possible in the experiments. The data shows that the scrape-off layer (SOL), just outside the confined plasma, conducts heat very rapidly, but particles more slowly – as expected. The new aspect is the separation of the ELM energy loss (at least the electron component) into ‘conductive’ and ’convective’ parts (Figure 4.6), i.e. those losses associated with the temperature and density drop respectively. The trend is that at lower particle collisionality (i.e. closer to ITER conditions), the conductive part rises. This helps: (a) the understanding of the kind of mechanisms that eject energy; and (b) the distribution of the energy that enters the SOL (e.g. many cooler particles or fewer hotter ones) which will affect the choice of divertor scenario.

A closely related topic is the role of plasma filaments in exhausting heat and energy during ELMs. Models are being developed on JET and MAST to estimate the energy left in the filament when it reaches the first wall. This is done by estimating the rate at which energy is conducted along the filament into the main SOL and then the divertor, and comparing with the velocity of the filament towards the wall (i.e. how much energy is conducted away before it hits the wall).

Finally, one of three ELM control/mitigation techniques being developed on JET is shown in Figure 4.7. The interpretation is likely to focus on the changes in edge current density from inductive effects as the plasma is moved with respect to the metal wall and the external fields, and this will help test the stability models.

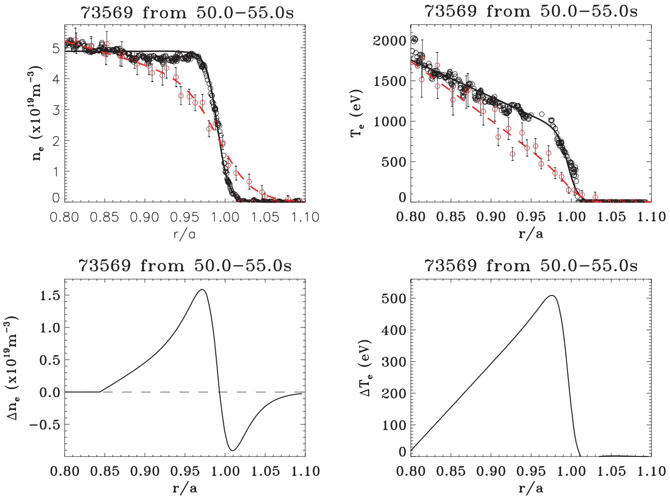

Figure 4.5: *ELM profile dynamics from ELM-selected HRTS data in a low-triangularity, low-collisionality Type I ELMy H-mode (δ=0.2, ν_e*=0.15, f_{ELM}=10 Hz). Black: pre-ELM profile between 3ms and 0.1ms before the ELM event. Red: post-ELM selected profiles between 0.2 and 1 ms after the ELM event*

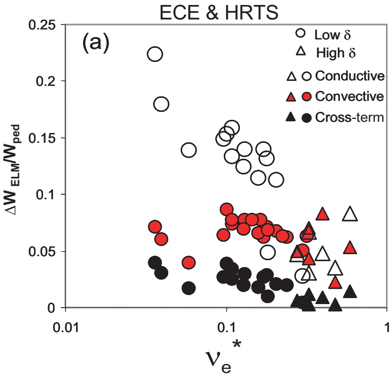

Figure 4.6: *ELM energy losses normalised to the pedestal energy content W_{ped} versus pedestal collisionality ν_e^*. $\Delta W_{conductive}$, $\Delta W_{convective}$ and $\Delta W_{Cross-term}$ using ΔT_e from the ECE diagnostic, and T_e, n_e and Δn_e from the HRTS system*

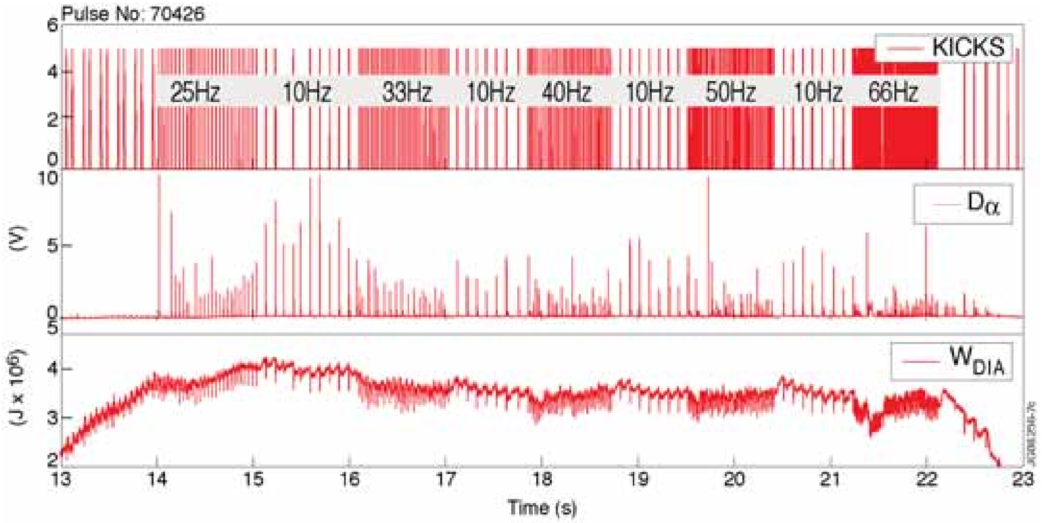

Figure 4.7: *Small 'kicks' applied to the plasma vertical position appear to trigger ELMs, allowing the ELM frequency to be increased and thus the energy of individual ELMs reduced*

4.4 ADVANCED SCENARIOS – TOWARDS STEADY STATE

Plasmas that are very long pulse or steady-state need to have a significant fraction of the plasma current driven non-inductively. This means that they need a large current drive contribution from external sources (e.g. neutral beam injection and lower hybrid current drive) and a substantial bootstrap component. This pushes the regime towards high pressure (to maximise bootstrap drive and, in ITER and DEMO, fusion yield) and low plasma current (to minimise the total current drive requirement). In practice this is challenging in terms of both plasma confinement and stability, and two approaches have been adopted: (i) scenarios with a localised region of good thermal insulation in the plasma interior, called an internal transport barrier (ITB); and (ii) scenarios with good confinement across the whole plasma radius but without the steep pressure gradient due to a localised transport barrier in the plasma core. The latter are often called 'hybrid' scenarios, as they allow very long pulses by a hybrid of inductive and non-inductive current drive. Both of these approaches employ an edge transport barrier, as in the baseline ELMy H-mode, but in these advanced regimes the current profile is generally broad or hollow resulting in low or negative magnetic shear. In general ITB plasmas have the potential to achieve very good confinement whereas the broad pressure profiles produced without ITBs are favourable for good stability. Recent JET experiments have been working at the boundary between these two scenarios with the aim of developing plasmas with good confinement but without the extremely steep internal pressure gradients that can lead to instability. UKAEA has a wide involvement in the advanced scenarios work, and the main elements from 2008/09 are indicated below.

4.4.1 PERFORMANCE IMPROVEMENTS IN STEADY-STATE RELEVANT DOMAIN – HIGHER BETA$_N$ AND ENHANCED CONFINEMENT

In 2008/09 significant increases have been achieved in β_N (normalised pressure) values at high and low q_{min} and two favourable domains of operation have been identified: one with q_0 close to 1 and the other with q_{min} close to 2. q_{min} determines the lowest m,n of instabilities that can appear, and is found to be an important parameter in optimising advanced scenarios (lower m,n modes are generally more damaging). This has been done partly by optimising the power waveform under β_N real-time control (Figure 4.8a) and, in the case of the domain with q_0

close to 1, by reducing the toroidal field strength. In parallel, progress has been made at higher magnetic field to integrate high power heating and current drive into plasmas in the domain with q_{min} close to 2. Good energy confinement has been achieved in this domain, exceeding the normal ELMy H-mode scaling used for baseline scenarios by 25% with only a weak increase in the internal pressure gradient due to an ITB.

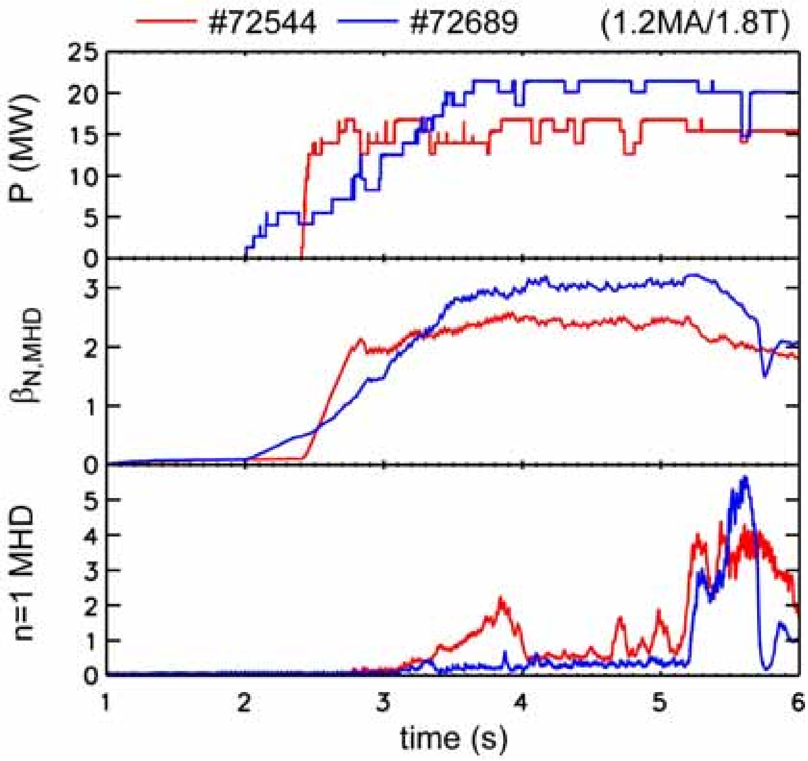

Figure 4.8a: *Optimisation of the power waveform (#72689) to avoid low m,n instabilities, at least temporarily*

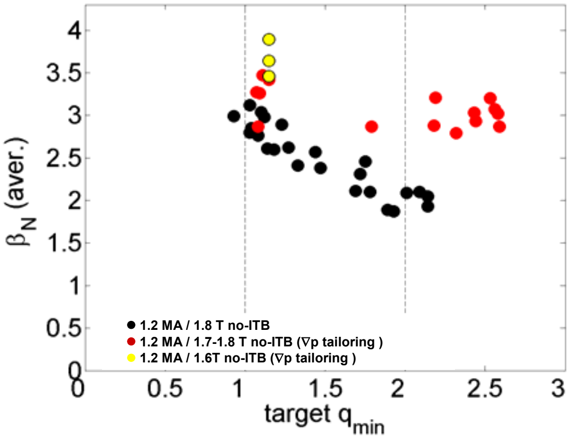

Figure 4.8b: *Performance of advanced tokamak plasmas with the range of magnetic configurations relevant to ITER steady-state operation. These experiments were specifically designed to test performance (in terms of pressure normalised to the plasma current and magnetic field) as a function of q_{min}. The red and yellow points were generated in 2008/09*

4.4.2 CONFINEMENT IN THE HYBRID DOMAIN

Whereas the ELMy H-mode regime is envisaged to allow operation in the domain $\beta_N \leq 2$ for the ITER baseline scenario, it is possible to produce stable plasmas with much higher values of β_N by shaping the plasma current waveform. Current confinement scalings have mainly been based on plasma regimes with moderate values of β_N and there is some uncertainty as to whether these describe well the effects as β is varied. Initial trends show that confinement degrades as β_N increases, as expected from the ELMy H-mode scaling. This degradation is however weaker in plasmas with less triangular shapes (see Figure 4.9). Plasma triangularity is not a parameter used in the present H-mode scaling expressions, and it is thought its main contribution is by changing the kinetic pressure at the edge of the plasma (the pedestal height).

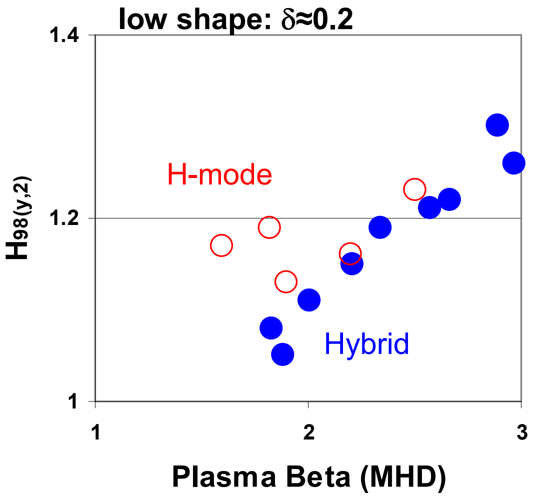

Figure 4.9: *Confinement with respect to ELMy H-mode scaling in plasma scenarios at low triangularity (δ~0.2). 'H-mode' plasmas start with a diffused current profile whereas 'Hybrid' plasmas are modified to have low magnetic shear in the core. The Hybrid cases show improved confinement at higher β_N*

4.4.3 INTERNAL TRANSPORT BARRIER: PHYSICS ISSUES OF FORMATION

The formation process for internal transport barriers appears to involve a variety of factors including plasma flow shear, rational q surfaces, and magnetic shear. Experiments using different combinations of neutral beam injection (NBI, providing heating with strong torque), Ion Cyclotron Resonance Heating (ICRH, providing heating with usually negligible torque) and toroidal field ripple (providing rotation braking) have helped elucidate this. For some plasmas at least, the thesis is that the initial ITB trigger is assisted by low magnetic shear and or proximity to low order rational magnetic surfaces (q=2, q=3) (Figure 4.10), but that for the development of a steep local pressure gradient at the barrier, substantial rotational shear is needed (or develops with the barrier). This rotational shear can partly be provided by the reduced momentum transport due to the ITB itself, and the observation of the importance of flow shear is generally consistent with stabilisation of Ion Temperature Gradient (ITG) modes (Figure 4.11). These figures indicate the level and range of plasma diagnostics needed to advance the understanding in this complex topic.

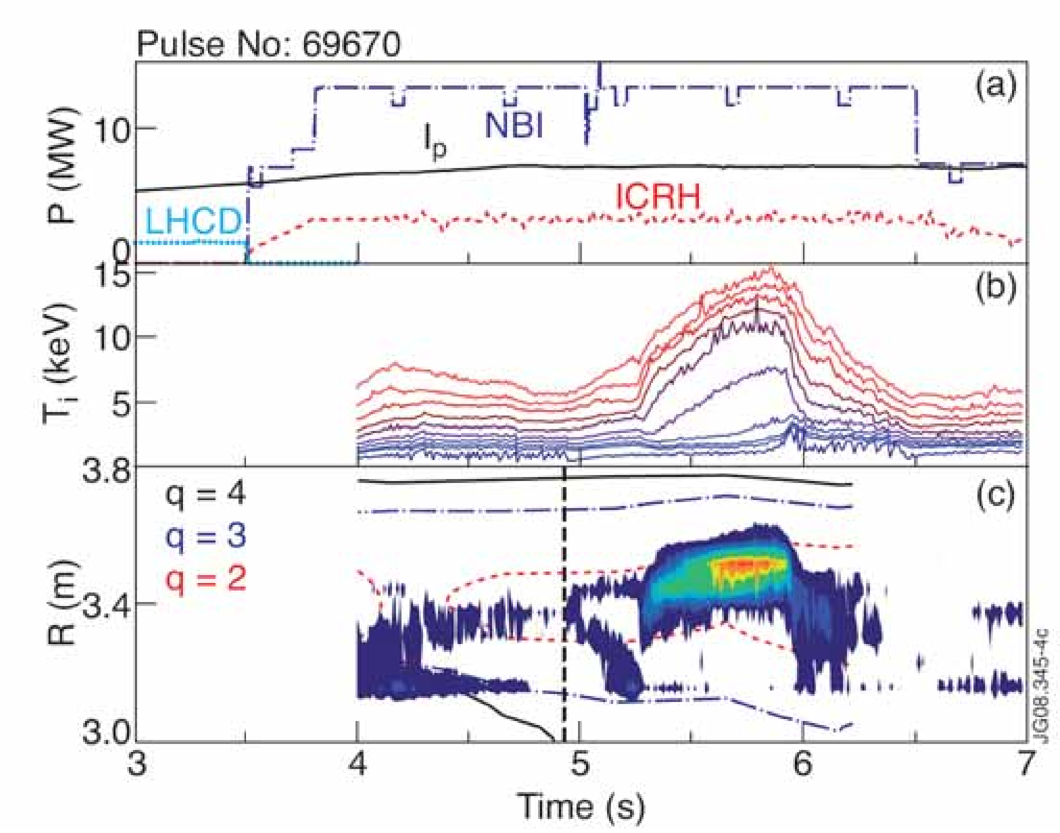

Figure 4.10: *a) NBI, ICRH and LHCD powers and the plasma current, I_p, in arbitrary units, for discharge, 69670 b) The ion temperatures measured by charge exchange recombination spectroscopy at various radial locations (from the centre to the plasma edge). c) The integer value contours of the q-profile (from motional Stark effect measurements), overlaid with the contours of the ρ^*_{Ti}, with in blue the minimum level $\rho^*_T=\rho^*_T{}^{crit}$ and in red the maximum level ($\rho^*T=0.050$). ρ^*_{Ti} is the inverse ion temperature gradient scale length normalised to the ion Larmor radius at the sound speed and $\rho^*_T{}^{crit}$ is the level above which an ITB is considered to exist at JET. The dashed vertical line at about 4.9s gives the time of the ITB trigger*

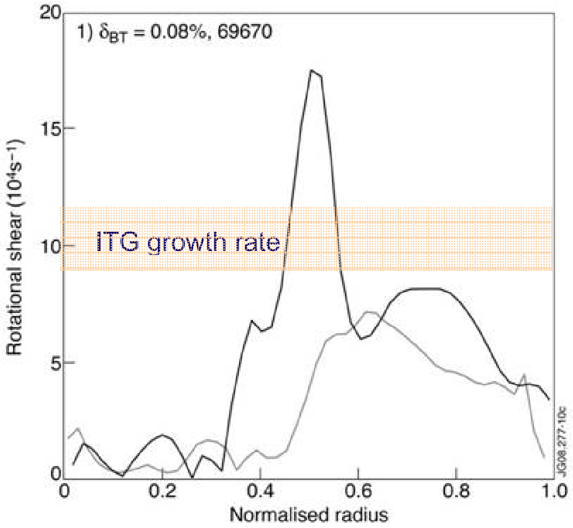

Figure 4.11: *Profile of rotational shear across the plasma (shear is the rate of change of rotation speed with radius). Dark line: during the growth phase of the ITB, faint line, before the ITB forms. The shaded band indicates approximately the range of the ITG linear growth rates across the region of interest, and a common model is that the ITG modes are stabilised when the shearing rate exceeds the linear growth rate. The profiles show that the data are generally consistent with this model*

4.4.4 STABILITY AT HIGH PRESSURE AND HOW TO TEST IT

Advanced scenarios are often designed to avoid significant internal instabilities, e.g. by operating with high q(0) or q_{min}, but this does not allow the external ideal-MHD kink mode to be avoided as the pressure is increased. If the plasma pressure is above the value that would make it unstable in the absence of a perfectly conducting vessel wall (the no-wall limit), then with a resistive wall (as in reality) it can be unstable to a slow-growing Resistive Wall Mode (RWM). Given the benefits of operating at high pressure (or β) there has been intense activity (see for example Chapter 6) on various factors that can stabilise these modes, notably rotation and kinetic effects from fast particles. Whatever stabilising effects there may be, it will be very important to know for any given plasma state how close it is to instability. This has generated interest in a technique called Resonant Field Amplification (RFA), which measures the plasma response to externally applied helical fields as it approaches instability. The simplest model has this amplification growing rapidly as β exceeds the no-wall limit, but the real situation is more complicated – the amplification is affected by the details of the current profile, plasma rotation and fast ion effects. Figure 4.12 shows a case when the simple model works reasonably well, but it is found that RFA appears well below the no-wall β-limit. A more sophisticated use of the data is being developed by comparing it with simulations using the MARS-F RWM code. This has shown that $d\ln|RFA|/d\beta_N$ is a maximum at the no-wall limit, holding the promise of a more reliable diagnostic of when the plasma exceeds the no-wall limit and is thus susceptible to RWMs (i.e. we would know that we have moved into a region where we have to rely on mode stabilisation mechanisms). See Chapter 6.

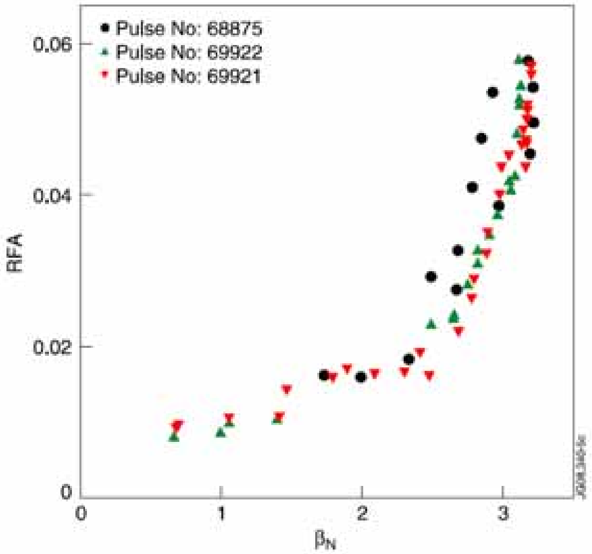

Figure 4.12: *Resonant field amplification (RFA) in JET plasmas 68875, 69921 and 69922 as a function of β_N. The sudden rise in the RFA is in broad agreement with the predicted no-wall limit, $\beta^{\infty}_N = 2.7$*

4.4.5 OTHER ACTIVITIES IN SUPPORT OF ADVANCED SCENARIO DEVELOPMENT

Advanced tokamak scenario evolution has been modelled to see how well the simulation codes predict the plasma spatial profiles, in particular the plasma current profile. Several other studies are under way for which it is premature to report in detail:

- effect of magnetic field ripple on ITBs and ITB comparison between JT-60U (Japan) and JET;
- integrated advanced scenario development (combining models of the different parts of the plasma with models of the various actuators – heating, current drive and fuelling);
- impurity transport in advanced scenarios, noting that impurities and helium ash could be confined at least as well as the fuelling species, leading to fuel dilution.

4.5 SCENARIOS FOR THE ITER-LIKE WALL ON JET

Installation of an all-metal (Be + W), ITER-like wall (ILW) in JET (see Chapter 3) will enable us to take a major step forwards in developing plasma scenarios for ITER. However it introduces stringent constraints upon heat loads which can be tolerated by the new plasma-facing and divertor-target surfaces, at the same time that the heating power and pulse length are being upgraded. Plasma scenarios will have to be designed to suit, and control and protection systems improved. Work has started in 2008/09.

The ELM mitigation schemes indicated above will be a part of the approach, but it is also important to tackle the power and energy deposited between ELMs. The two main mechanisms are high recycling in the divertor to reduce the charged particle energy at the divertor target, and high radiation by impurities in the plasma boundary, SOL and divertor to reduce the power approaching the target.

At present, radiation in JET tends to be dominated by intrinsic carbon from the carbon wall, so a replacement is needed. The preferred route is to seed plasmas controllably with extrinsic gaseous impurities. Systematic studies have been undertaken of both ELMy H-mode and advanced tokamak scenarios seeded with either neon or nitrogen (Figure 4.13 shows the configurations). These species have differing radiation, ionisation and recycling characteristics and therefore should allow a wide range of ILW strategies. Both species have substantially reduced the inter-ELM target load (Figure 4.14a and b), at the expense of some loss of energy confinement and plasma purity. Evidence from recent experiments is that nitrogen tends to stay more within the divertor and assists the density rise required for the approach to ‘detachment’, i.e. separation of the plasma from the target by a recombining region. Figure 4.14 c and d compares Ne and N_2, including ELMs. These results will be used to develop optimised pulse designs for the ILW campaigns, for example by using active feedback control of seed impurities.

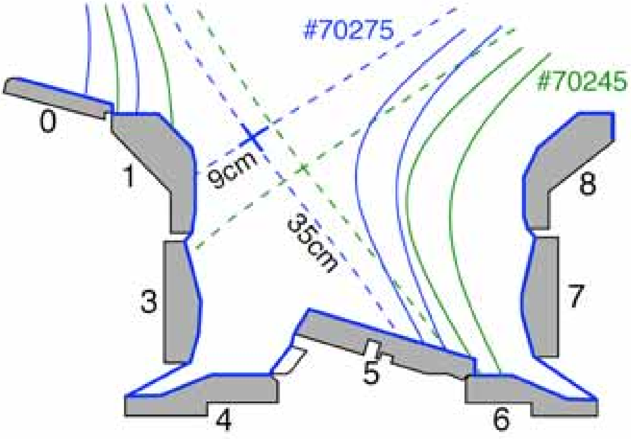

Figure 4.13: *Divertor structure used for the high triangularity advanced tokamak (blue, 70275)*

and low triangularity baseline H-mode scenarios used for the power deposition studies (Figure 4.14 below)

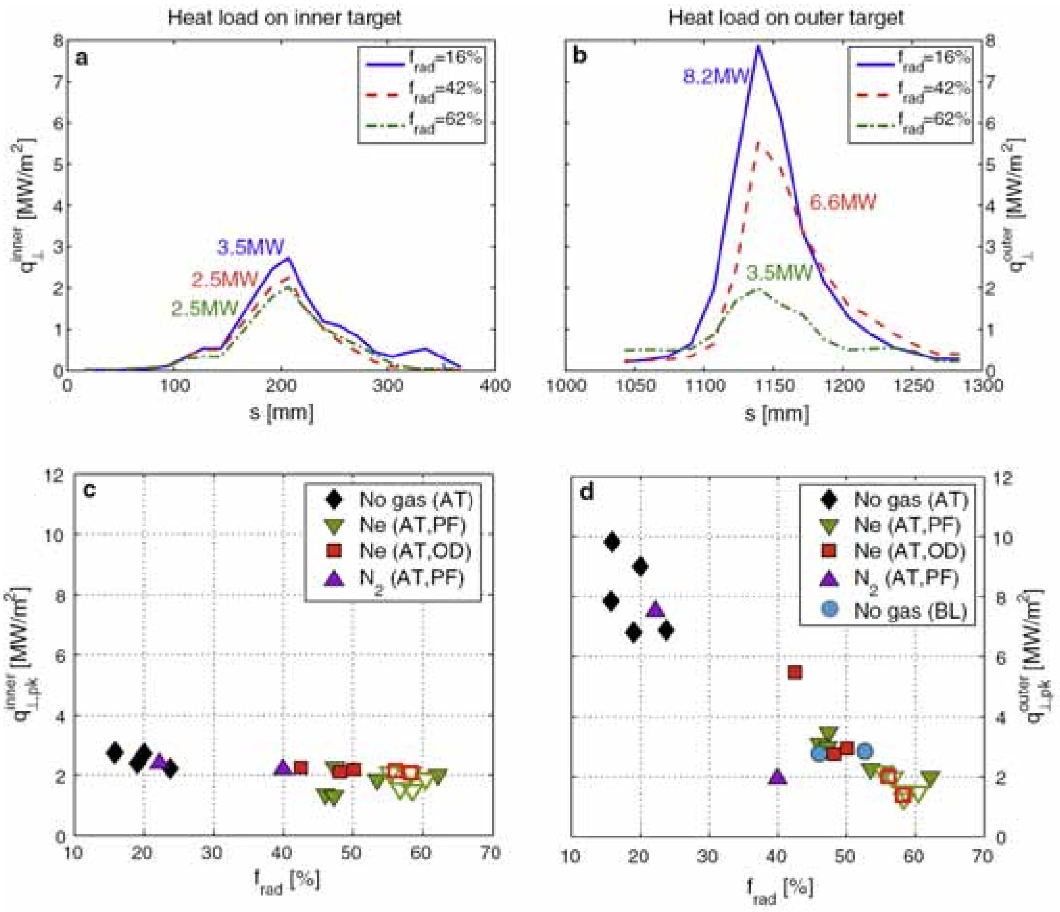

Figure 4.14: *Inner (a) and outer (b) heat load poloidal profiles for three radiative fractions: 16%, 42% and 62%, achieved with neon seeding. Inner (c) and outer (d) peak heat load as a function of f_{rad} achieved with neon injected from the private flux region (PF) or from the outer divertor (OD), or with nitrogen injected from the private flux region. The unseeded discharges are indicated by 'no gas'. The open symbols denote L-mode plasmas, and BL denotes baseline ELMy H-mode*

4.6 SPECIFIC PHYSICS DEVELOPMENTS

4.6.1 FAST ION PHYSICS

The strong ICRH and some relatively new fast ion diagnostics (here gamma ray cameras for confined alphas, and a phase-space imaging scintillator for lost ions) makes JET a good environment to study fast ion populations and loss processes, as understanding of them is very important for predicting the behaviours of alpha particles both in fusion reactors and ITER. Figure 4.15 shows the use of the scintillator probe to measure variations in lost alpha particles and Figure 4.16 shows evidence of redistribution of fast ions after a sawtooth crash, as determined from gamma ray emission profiles.

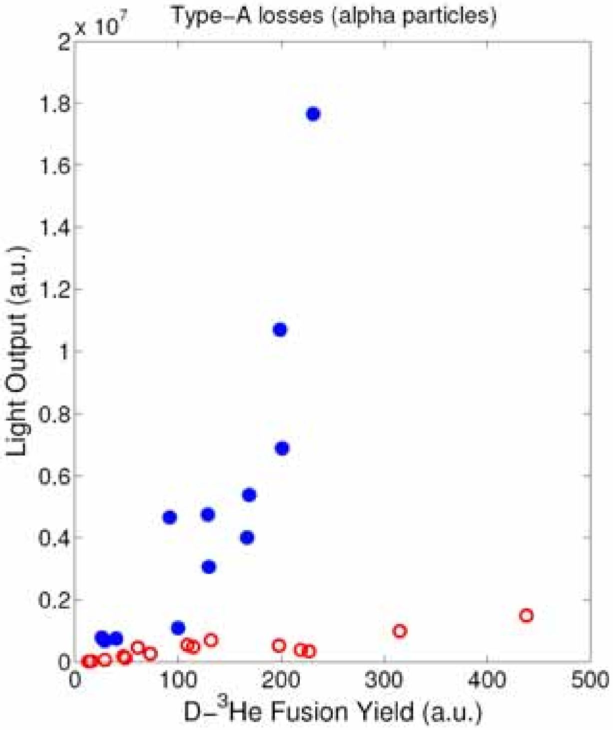

Figure 4.15: *Loss rate of 3.6 MeV alpha particles from D-^{3}He reactions for plasmas with Alfvénic MHD instabilities (solid blue) and no measurable activity (open red). The light output from the scintillator actually increases approximately linearly with the amplitude of the MHD instability*

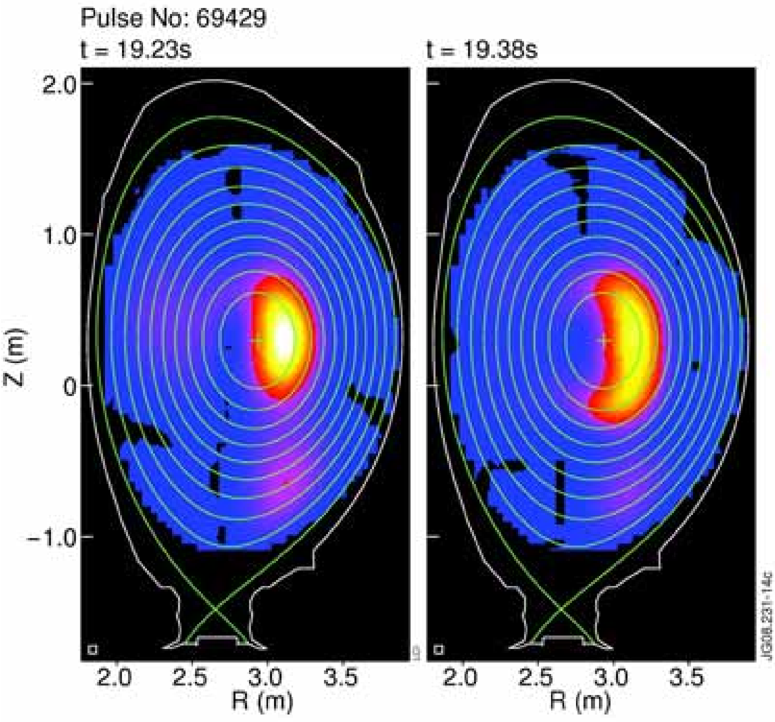

Figure 4.16: *Tomographic reconstructions of γ-ray emission profiles obtained with the 2D γ-camera [3] in a ^{3}He-minority ICRH discharge #69429 (I_P=3MA, B_T =3.1T): left – before a sawtooth crash; right – after the crash*

4.6.2 DISRUPTIONS

Disruptions – i.e. rapid terminations of the plasma – have been a concern for very many years due to the high transient thermal loads and electromagnetic forces that ensue, especially on large tokamaks, and these are important for the design of the in-vessel components on ITER and the operation regimes/rules on JET and ITER (and other tokamaks). Over the last year some of the halo current detectors (Figure 4.17) installed in 2004 and improved in 2007/08 have started to give very useful data on the poloidal distribution of halo currents (Figure 4.18), not previously available to any real extent at JET. The data confirm the assumptions on current density and poloidal extent of the halo region, at least for upward disruptions, and the information has been transferred to ITER. Heat flux measurements with the wide-angle infra-red camera (after the complex geometrical corrections needed) have been performed for these discharges (and others) which show that the halo currents flow in about the same region that is heated by plasma contact, as would be expected to first order.

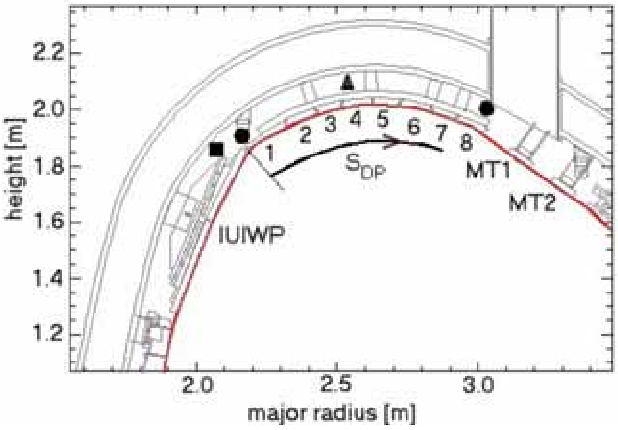

Figure 4.17: *Location of the halo current sensors in the upper dump plate, used seriously for the first time in 2008/09*

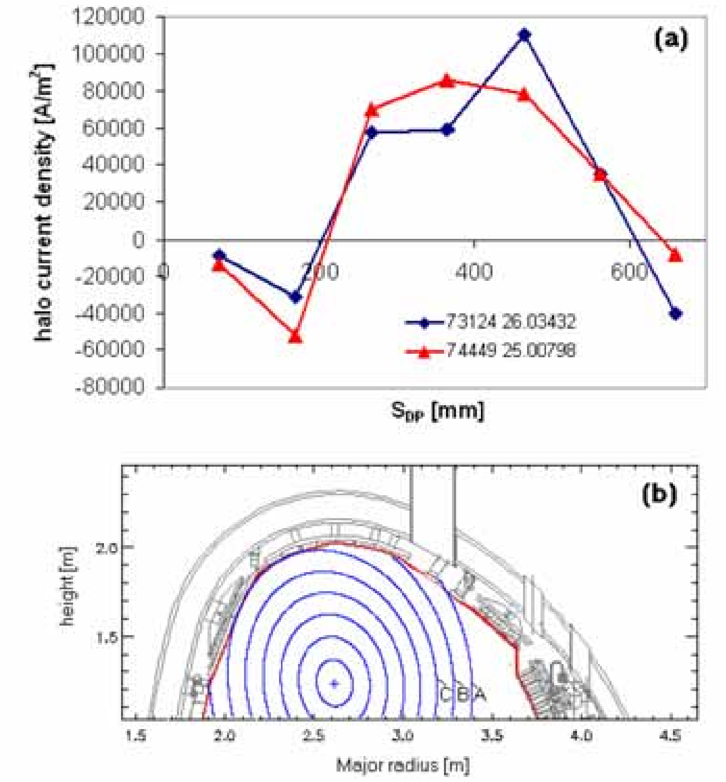

Figure 4.18: *(a) Halo current density profile on the upper dump plate for both 74449 and 73124 and (b) a cartoon of the plasma at the time of the profile based on a computational reconstruction. 74449 is a 2 MA Ohmic plasma, 73124 (2.25 MA) has high beam power (~15 MW)*

4.7 SYSTEM IMPROVEMENTS – JET ENHANCEMENT PROGRAMME

4.7.1 HEATING AND CURRENT DRIVE PROJECTS

A ICRF – ITER-like antenna and External Conjugate T

The single largest activity in the science programme this year has been the commissioning and optimising of the new ICRF ITER-like Antenna (ILA) on plasma, and more recently of an external (outside the vessel) conjugate T (ECT) between two (C and D) of the four existing ICRF antennas (see Figure 4.19). This conjugate-T configuration also present on the ILA system (but in-vessel with tuneable capacitor to adjust the conjugate-T properties) allows the generators to be isolated from the large transient changes in coupling impedance generated by ELMs. Figure 4.20 shows that by using the ILA, the ECT and the 3 dB systems (installed in 2005 between antenna A and B), an impressive 8.3 MW of ICRF power has been coupled into an H-mode plasma with type I ELMs. The ILA is also aimed at providing very high power density through a dense array of straps, as needed for ITER. A dense array of straps leads to strong inductive coupling between individual straps, which makes matching the four pairs of straps to the plasma very complex. The whole process of design, commissioning off plasma and commissioning on plasma has been enormously valuable and has been a major technical success, without which the ITER antenna design would be much less secure. So far the target power density in H-mode has not been achieved, due largely to a lower than expected coupling impedance to the plasma.

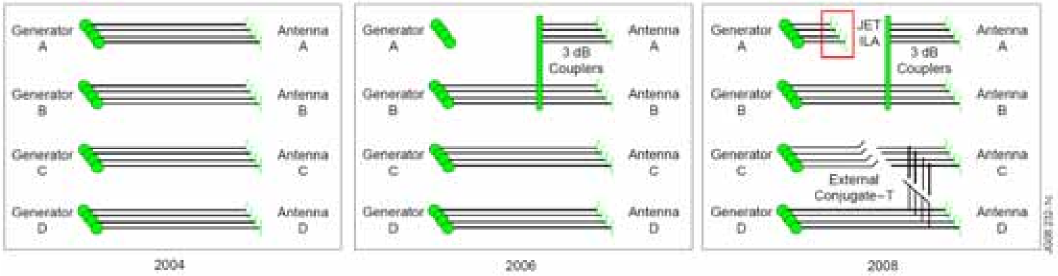

Figure 4.19: *Recent evolution of the JET ICRF configuration*

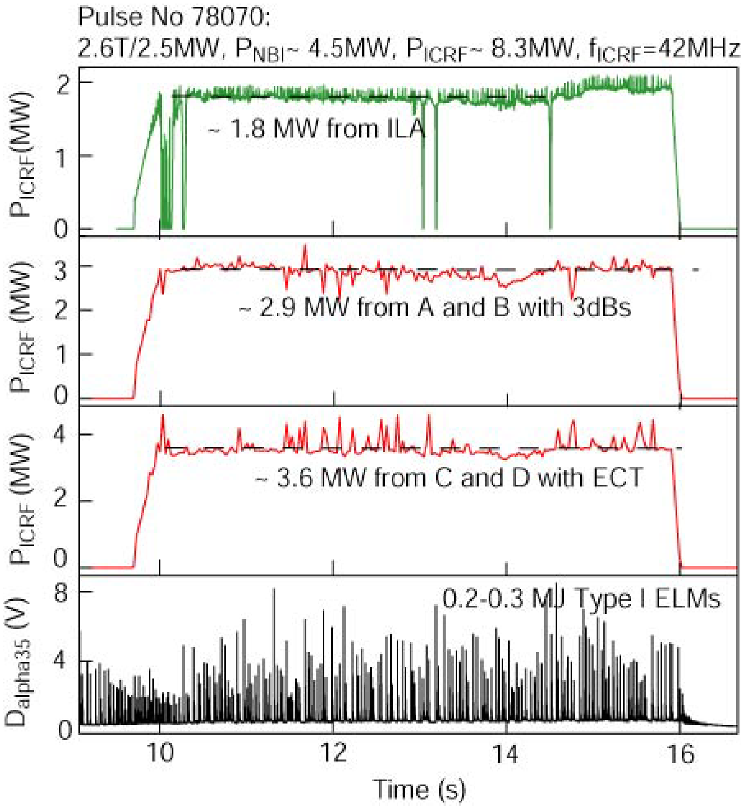

Figure 4.20: *ICRF coupled power from the ITER-like antenna, the A2 antennae A and B equipped with 3 dB coupler and the A2 antennae C and D equipped with ECT. Up to 8.3 MW of trip-free ICRF power has been coupled to a type I ELMy H-mode plasma)*

Arc detection is a critical issue for JET and ITER. Indeed, because of the introduction of the conjugate T configuration and of the operation during ELMs, it has been necessary to develop new arc detection systems. On the ILA, in order to detect arcs around the T-junction, an arc detection system based on two sets of measurements detecting change in the matrix impedance of the antenna and referred to as Scattering Matrix Arc Detection (SMAD) was successfully implemented. Another system based on measurement of transients during arcs and referred as Sub-Harmonic Arc Detection (SHAD) system was extensively tested. Finally for the antennas C and D (ECT), the newly implemented Advanced Wave Amplitude Control System (AWACS) designed to be sensitive to arcs undetectable by traditional methods, has proved to be an efficient protection tool for the conjugate-T circuit.

The nature of the internal conjugate T on the ILA makes local measurements difficult, so a new approach has been developed, SMAD, where the network of straps and capacitors in the vacuum is analysed from external measurements (more than are necessary if the network was working normally), and changes in the characteristics of one element due to an arc is detected via a change in the matrix linking two sets of external measurements, and the system tripped (see also Chapter 3).

B Neutral beam enhancement (NBE)

The present beam system is limited in pulse length due to the thermal capacity of some of the high heat flux elements, and it was recognised that there is also scope for a power increase by changing the source configuration and accelerator grids in the 16 Positive Ion Neutral Beam Injectors (PINIs). Together with four new high 125 kV / 130 A voltage power supplies (two PINIs per supply) this forms a large project led by the EURATOM/UKAEA Fusion Association. UKAEA also has responsibility, as JET Operator, for provision of supporting infrastructure, installation of equipment on the JET machine, and for integrated commissioning of the enhancement. The aim is to increase the maximum power to at least 34 MW (with deuterium) for 20 s pulses, compared to the 23.8 MW record and 10 s pulse length at present. In particular, the elements which protect the walls of the narrow torus entry ports of the beamlines against direct beam interception and re-ionisation (from beam-gas collisions) must be upgraded to allow operation for up to 20 s beam pulse length from each NB box even at the highest gas fuelling rates required for high plasma current operation. The project has been running for several years, and will be completed in the 2009/10 shutdown. The project will replace the scrapers in the torus port (duct), some other high heat flux components, the first-stage neutralisers, some of the accelerator grids in the PINI and the PINI backplates, and reconfigure the magnetic field on the PINIs. See Figure 4.21.

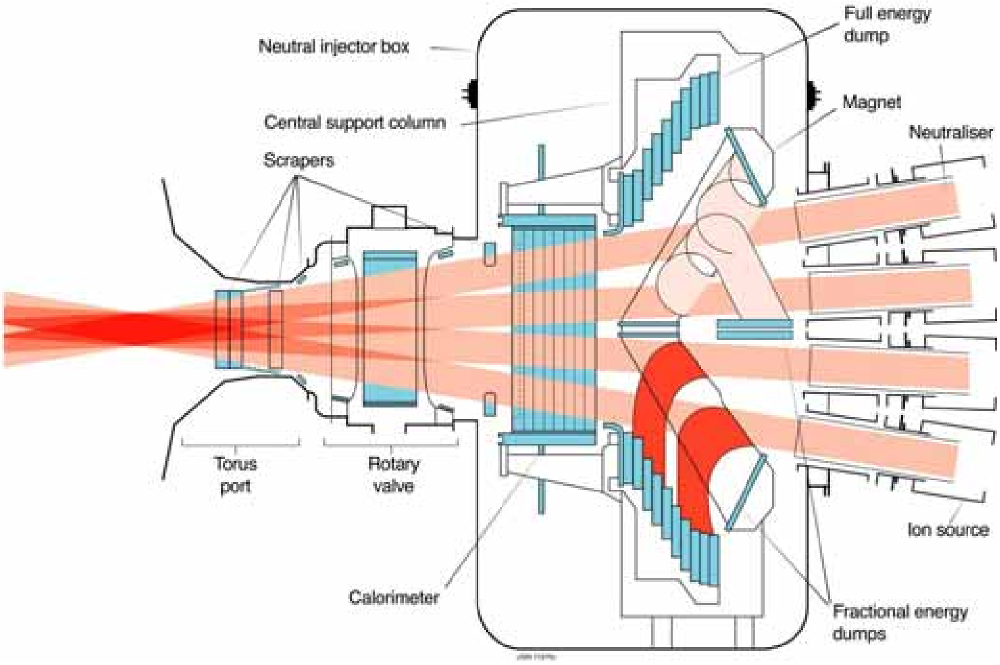

Figure 4.21: *Schematic of one of the two JET 8-injector neutral beam boxes*

On-site commissioning of the four new high voltage power supplies is now complete. All the major replacement beamline components are either delivered of in the final stages of manufacture and same applies for the PINI components. A further four PINI units with accelerators modified and tested in NBE configuration have been exchanged during the past year. These exchanges released further PINI accelerators for conversion to NBE configuration, and bring the total number of exchanged PINIs to eight. Under the present planning, the total of five spare PINIs currently available for conversion will be built and tested before the start of the EP2 Shutdown. This would leave just three PINIs to be converted and tested during the Shutdown itself when there is a higher risk of disruption to operation of the neutral beam test bed.

Figure 4.22: *Components of the hypervapotron duct scrapers during manufacture and assembly*

4.7.2 PLASMA DIAGNOSTICS

The necessary scientific progress to advance fusion research calls for continually improving plasma measurements, to match developments in theoretical models and the need for more precise experiments (e.g. the pedestal identity experiments described above) and improved plasma control. These developments are led by EU and international fusion laboratories. Two of the many diagnostics in the EP2 JET enhancement programme are led or co-led by UKAEA. The first is an upgrade to the edge LIDAR Thomson scattering system. The other is a suite of upgrades to the spectroscopic diagnostics (visible / near IR to X-ray) so that we are better equipped to study the impurity sources and the impurities in the plasma after the ITER-like Wall has been installed, in particular tungsten and beryllium. This is a long project and will only be completed after the 2009/10 EP2 shutdown, so is best reported in later annual reports. More recently a project was launched to design and implement a multi-band microwave reflectometer to provide the first high space and time resolution density profiles on JET.

A Edge LIDAR upgrade

This project improves measurements of the edge density profiles on JET and is very relevant for the developments for ITER (Chapter 8). See Figures 4.23 and 4.24.

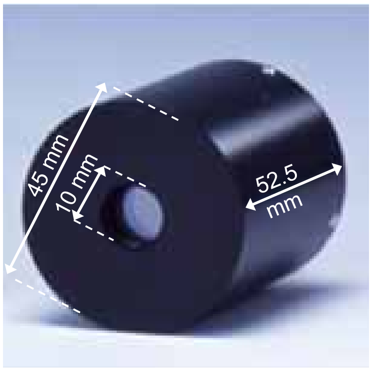

Figure 4.23 a): New GaAsP detector for the JET edge LIDAR system

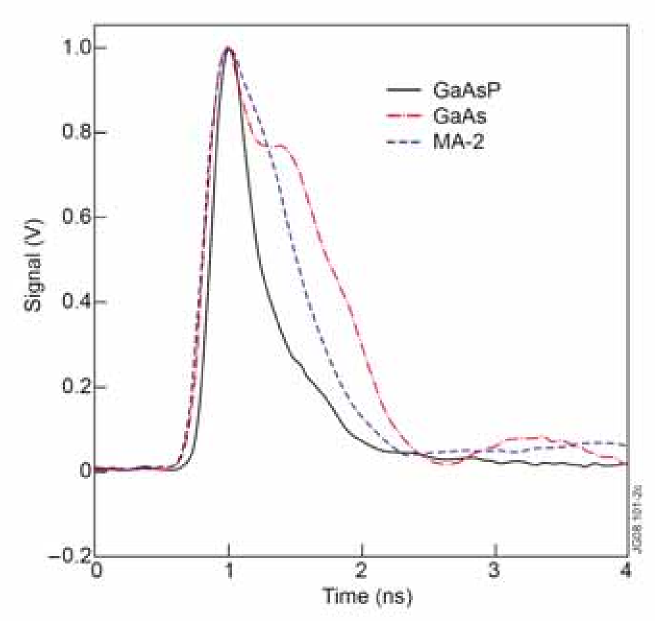

***Figure 4.23 b):** Time response of new GaAsP detectors to a very short light pulse, compared to the previous detectors. FWHM=290ps for the GaAsP*

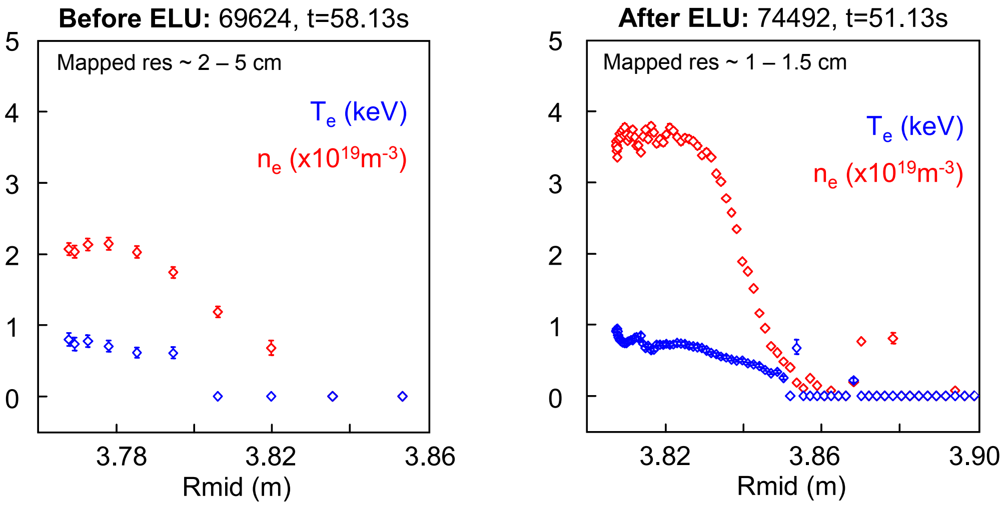

***Figure 4.24:** Processed data from the edge LIDAR system, showing electron density and temperature profiles measured before and after the upgrade*

B New profile reflectometer

Reflectometry is an established method of fast density profile measurement. It is based on reflection from cut-off layers in the plasma, where a specific frequency is reflected at a specific density, depending on the wave polarisation (O-mode or X-mode), and also on magnetic field. The project (KG10) is led by UKAEA and conducted in collaboration with IST (Portugal) and CEA (France). IST and CEA are providing the microwave sources and detectors in the Q-band (33-50 GHz), V-band (50-75 GHz), W-band (75-110 GHz) and D-band (110-170 GHz). UKAEA is providing the microwave optics and the fast digitisers. Figure 4.25 shows how the various bands are used to cover a range of density profiles at different magnetic fields. Figure 4.26 shows one of the quasi-optical boxes manufactured in the UK.

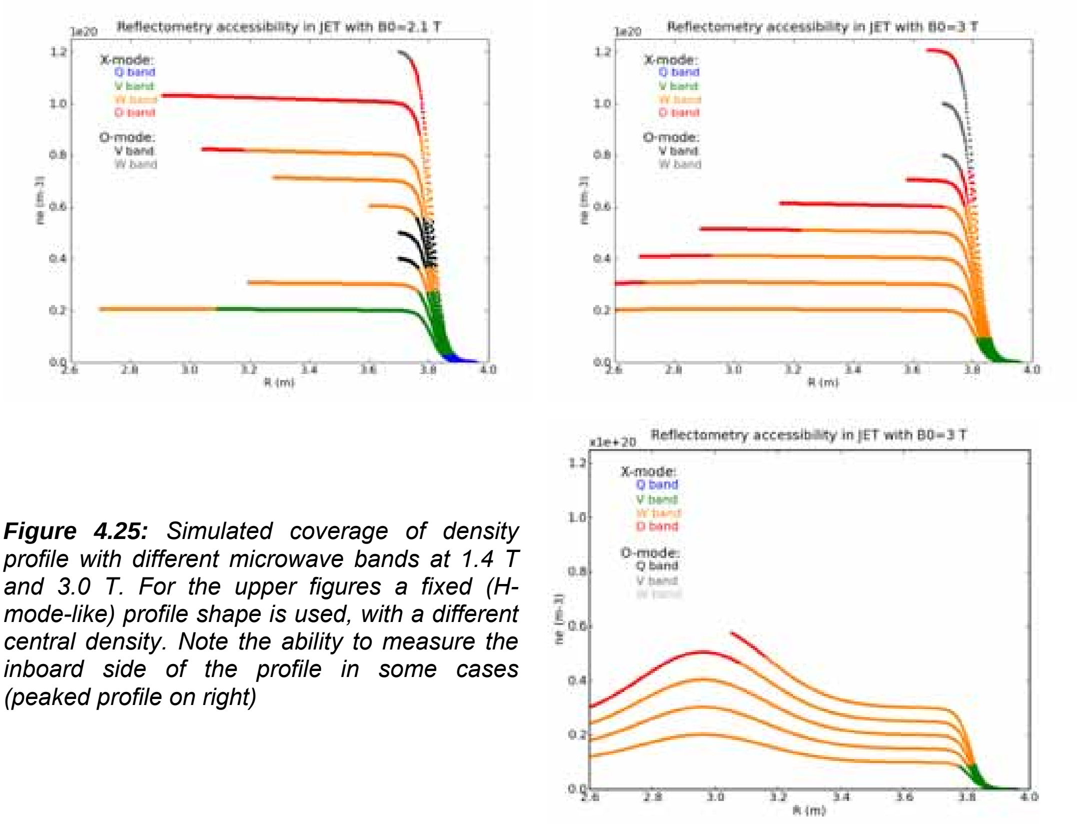

Figure 4.25: *Simulated coverage of density profile with different microwave bands at 1.4 T and 3.0 T. For the upper figures a fixed (H-mode-like) profile shape is used, with a different central density. Note the ability to measure the inboard side of the profile in some cases (peaked profile on right)*

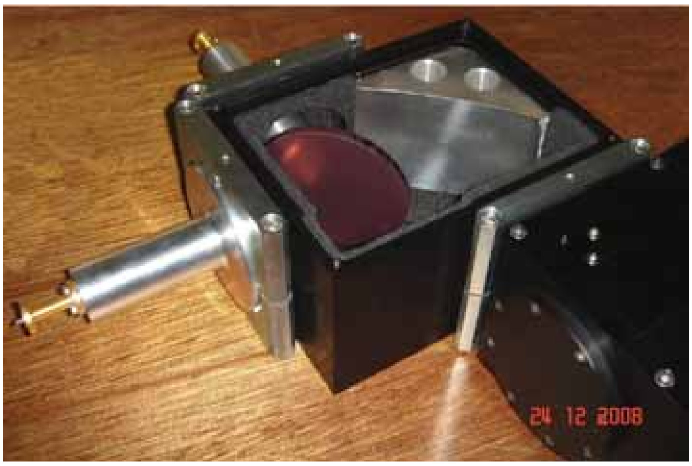

Figure 4.26: *Two of the quasi-optical boxes for the new wideband reflectometer. The open box is fitted with two antennae and two waveguide transitions. A 60 GHz filter has also been inserted. these boxes serve as combiners, splitters and filters to allow the six microwave systems (four bands, all in X-mode and two of them also in O-mode), to use four waveguides to/from the torus*

4.7.3 PLASMA CONTROL – ENHANCED RADIAL FIELD AMPLIFIER, AND SHAPE AND POSITION CONTROL SYSTEM UPGRADE

Elongated plasmas are generally vertically unstable, so an active feedback system is needed. On JET it is found that the present systems is limited when there are large ELMs – the disturbance to the position and the control signals is too large to be recovered in some cases and a vertical displacement event can ensue, terminating the discharge. This tends to be more likely in the higher current higher power plasmas needed for the future campaigns. To address this a multi-part project was launched, to improve the 'observer' (the measure of the plasma position) to make it more immune to spurious magnetic fields; to improve the control algorithms and control hardware and to procure a more powerful power supply (more than a factor of two) to allow control of much larger ELMs. UKAEA leads the project to procure the 60 MVA (5 kA, 12 kV) fast (100 microseconds to apply the full voltage) IGBT-based power supply Figure 4.27) to replace the present ~25 MVA one (which also has obsolescent technology). It also leads the subproject on the controller hardware upgrade, in collaboration with ENEA (Italy) and IST (Portugal) – one of the new digitiser/processor cards is shown in Figure 4.28. The new controller will allow, inter alia, easy changes to the 'observer' and the actual control algorithms, including, if needed, switching controllers during a discharge to maintain an optimum for the state of the plasma at the time.

Figure 4.27: *One of four modules of the new 60 MVA vertical position control supply (ERFA) being lowered into position in March 2009*

Figure 4.28: *One of the new ATCA cards designed by IST (Portuguese Association) combining fast digitisers and FPGA-based processing to provide the 'observers' for the vertical position controller. Only one of the input digitiser cards is shown*

4.7.4 DATA ANALYSIS, MODELLING AND INTERPRETATION

In addition to the hardware developments indicated above, the data analysis and forward modelling tools need to be developed continuously – sometimes better understanding, calibration and use of data can be as powerful as a hardware upgrade. Three areas are indicated here.

Atomic physics data and calculations are vital to many diagnostics. Much of the information comes from the Atomic Data and Analysis Structure (ADAS) system developed at Strathclyde University in co-operation with JET in the 1990s which integrates atomic physics data and modelling in a way that is directly usable by experimentalists at JET and many other tokamaks (and non-fusion groups). Recently a proposal to expand ADAS won a large grant from the European Commission to form ADAS-EU (http://www.adas-fusion.eu/index.php). Related modelling is performed at Queens University Belfast, looking in particular at energy levels and rate coefficients in noble gases such as xenon and krypton which are of interest to fusion as trace species for some diagnostics.

When JET has a metal wall in 2011, the interpretation of some diagnostics will be changed due to wall reflections – this was studied in some detail and shown to be significant but manageable and not as disabling as had been initially feared, and in particular it will usually be adequate to consider only the first reflections.

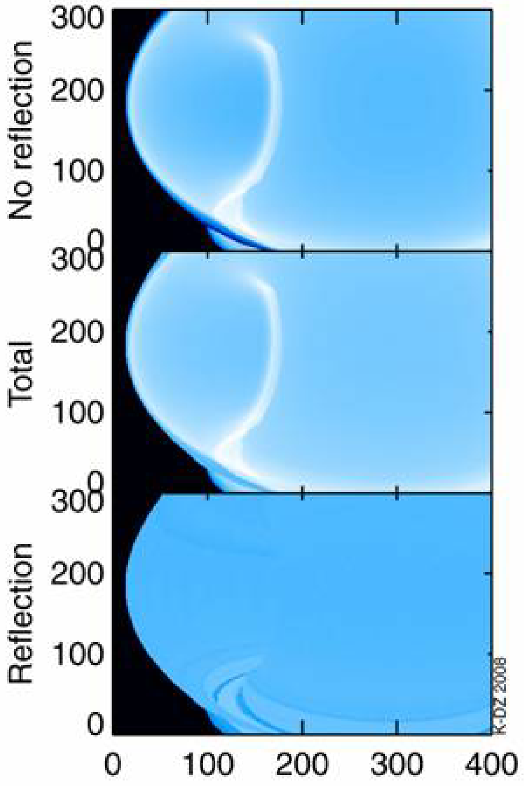

Figure 4.29: Simulated pictures of edge emission at 527 nm for JET-like geometry. Top: direct contribution. Middle: direct contribution plus reflected components. Bottom: reflected component only. All three pictures are shown on the same normalized colour scale. The outlines of the machine's centre column and the divertor tiles can just be seen in the bottom picture

Another area that has seen significant activity during the year is the use of forward modelling or Bayesian techniques applied to diagnostics to make best use of individual measurements to arrive at estimates of the quantities needed for developing and testing plasma models (generally these need many different measurements to be used together). This has been developed for particular JET diagnostics (the interferometer/polarimeter and the LIDAR Thomson scattering) in collaboration with Imperial College (PhD project) as well as other Associations.

4.8 FUTURE PLANS

The remainder of the 2009 campaign is aimed at commissioning and exploiting the improved vertical position control system (ERFA), together with experiments in hydrogen and helium to provide information for the lower activation campaigns on ITER before DT operation as well as preparations for the ITER-like Wall campaigns. The ‘headlines’ are:

- full scientific exploitation of systems commissioned during 2008, and issues that could impact on the design of ITER components;
- qualification of integrated operating scenarios for ITER; and
- physics issues essential to the efficient exploitation of ITER, including low neutron yield experiments, particularly in hydrogen and/or helium.

JET will then go into a long shutdown to install the ITER-like Wall, the neutral beam upgrade and the remaining diagnostics in the EP2 suite of enhancements. In addition there will be some dedicated diagnostics calibrations, notably a calibration of the neutron diagnostics using a neutron source inside the vessel (this is also planned on ITER) and a much improved calibration of the spectroscopy diagnostics using a light source inside the vessel (in the past such end-to-end calibration has not been possible).

During the shutdown there will be extensive analysis and reporting of the 2008 and 2009 campaigns, with several major biennial international conferences, and also preparation for the 2011 campaigns with the new wall and the new types of experiment that will be needed.

For the 2011 campaigns there will be a new task force structure. Two new task forces, E1, E2 will replace the present campaign task forces S1, S2, E, H, M, D, T. Later there may be further TFs including DT. TF FT will continue. E1 will focus on developing and expanding the operating domain with the ILW, and E2 will focus on the scientific exploitation.

The European Fusion Facilities Review panel reported late in 2008, and stressed the importance of JET as the largest tokamak, closest to ITER. It made a strong recommendation for the continued exploitation of JET, including possible operation after 2014. It also made mention of the benefit of some further enhancements, for example an ECRH system (investigated in 2000, but not implemented), and this is being studied in 2009, albeit primarily from the Operator side as far as UKAEA is concerned.