4 JET Studies

4.1 SUMMARY

As described in Chapter 3, 2009 saw the end of operation with carbon plasma facing components (PFCs) on JET for the foreseeable future. A major shutdown began in October 2009 to install all metal PFCs – beryllium for the main chamber and tungsten for the divertor, in the form of thin coatings on carbon-fibre composites and solid tungsten for one row of tiles. The campaigns thus focused on core plasma research common to any wall materials, and preparing scenarios for the more delicate metal wall. Notable achievements include operation at high power (using earlier neutral beam enhancements) and high current (4.5MA); new information on access to H-mode in helium compared to deuterium; substantial progress in scenario integration for advanced and 'hybrid' scenarios for long pulse steady state on ITER as well as new results in many cross-scenario or specialist topics.

The major project to design, manufacture install and then test an ITER-like ICRH antenna came to an end having provided a wealth of information and an antenna modelling capability in the EU that is in a different league to the state at the start of the project – together this puts the ITER design (Chapter 8) on a secure footing.

Two CCFE-led enhancement projects were completed, the 60MVA fast enhanced radial field amplifier (ERFA) which was used for the 4.5MA plasmas and the new profile reflectometer for the first high space and time resolution measurements of the plasma density profile on JET.

This chapter is divided into:

- Baseline scenarios for ITER – ELMy H-mode;
- Advanced scenarios – towards steady state;
- Divertor issues and power management for the ITER-Like Wall;
- Research on common and specific topics;
- Progress on enhancement projects;
- Future plans.

4.2 BASELINE SCENARIOS FOR ITER – ELMY H-MODE

The empirical understanding of the ELMy H-mode, the scenario presently planned for the high gain (Q=10) operation on ITER, is now very advanced. There are two areas of uncertainty which are a particular issues for ITER: (a) the conditions for access to H-mode (in particular the power); and (b) the height of the edge pedestal which is expected to largely determine the core density and temperature since the gradients in the core plasma in this scenario are unlikely to vary very much.

4.2.1 H-MODE ACCESS

In the absence so far of a reliable model for the criteria for formation of the edge transport barrier, the conditions for access to H-mode have been developed empirically, focussing on the power required (the power threshold). The resulting scaling has a wide scatter when projected to ITER (which will use helium in its early phases), and in particular has a dependence on the isotope of hydrogen. There has been less data on the threshold in helium and this was a key experiment in the helium plasmas used shortly before the start of the shutdown. Recent experiments on ASDEX Upgrade, DIII-D and MAST have given different results (the power threshold being either about the same or higher in helium compared to deuterium). Figure 4.1 shows the data from a dedicated scan of He concentration, showing little change. This contrasts with earlier JET data (2001) showing the threshold was higher in He. The empirical resolution is that the 2001 data were at lower density, and the density behaviour seems to be different in He and D. The new data is reassuring in that the access conditions do not appear to depend on the heating scheme (i.e. whether NBI or ICRH).

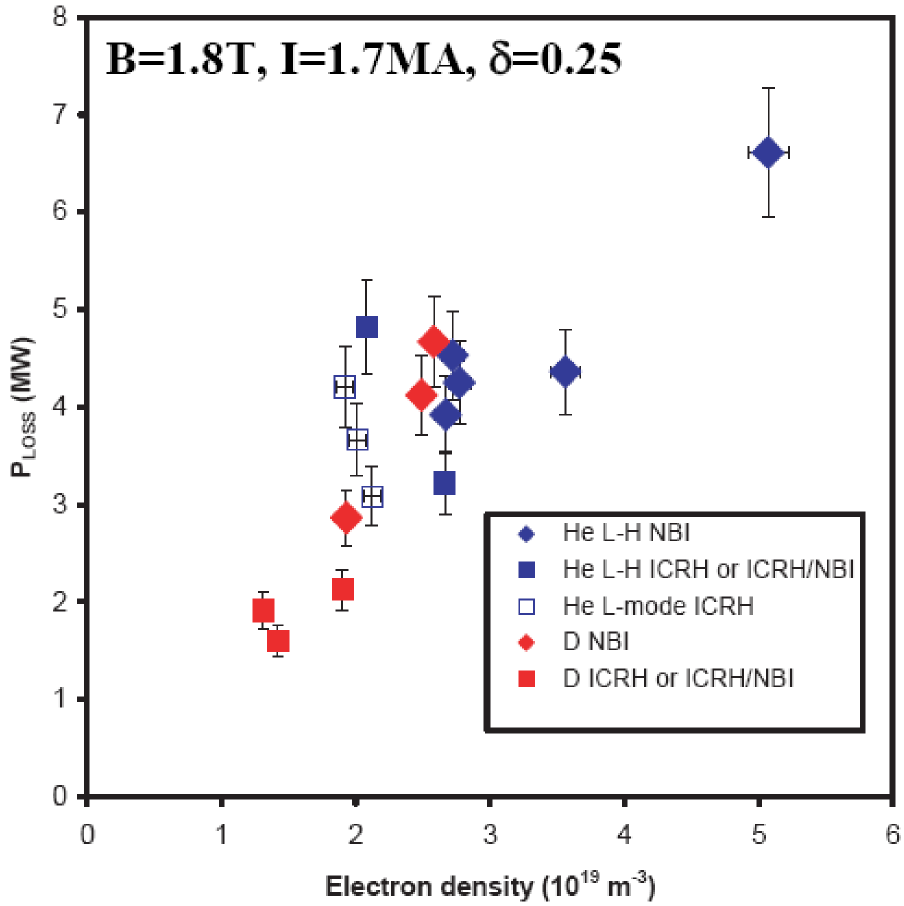

Figure 4.1: *Loss power versus electron density for ^{4}He discharges from the JET 2009 studies (blue) and their D references (red). Open symbols denote L-modes*

4.2.2 EDGE PEDESTAL

In a similar way as for H-mode access, progress on predictions of the pedestal for ITER is based mainly on experiments. The new diagnostics on JET over the last two years have allowed very detailed measurements (see data in last annual report for example). As mentioned above, the core plasma parameters in ITER are strongly influenced by the pedestal height. There are theoretical reasons which limit the pedestal gradient (plasma stability), but only limited understanding of what sets the width (and thus the height if the gradient is fixed), however attempts have been made, as illustrated in Figure 4.2 which compares data from JET and DIII-D with a model (EPED1) where the pedestal height is deduced from consistency of two theory-based models for the average pedestal gradient.

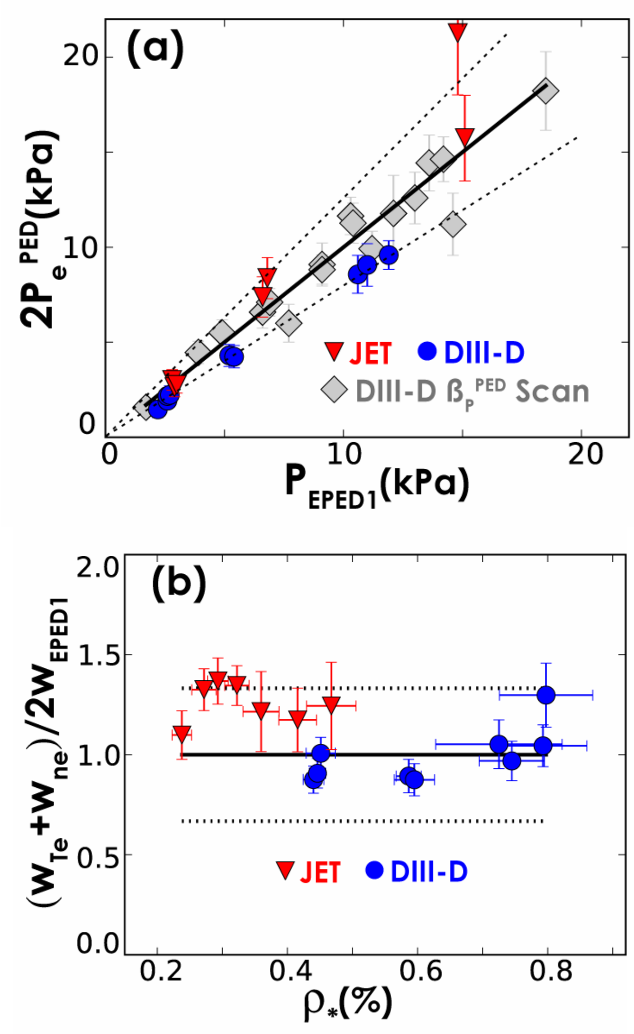

Figure 4.2: *Comparison of pedestal data from the ρ∗ scaling experiment with the EPED1 model. The dotted lines represent the range of variability in the model predictions (see text). (a) Measured versus predicted pressure at the pedestal top (for measured it is assumed electrons and ions have the same pressure); grey symbols are other DIII-D data covering a wide range of shapes and β's, (b) Ratio of measured and predicted pedestal width versus ρ∗ (a key dimensionless parameter in stability considerations)*

4.3 ADVANCED SCENARIOS – TOWARDS STEADY STATE

For fusion to be an attractive energy source, either steady state or very long pulse operation (combined with energy storage) is required (see Chapter 7 on DEMO etc). Both options are pursued on JET, and both are planned on ITER. This is a demanding programming, integrating different elements such as high β, effective non-inductive current drive at the necessary location in the plasma, and sufficiently high density to allow a low temperature (i.e. low erosion) divertor. Two different approaches are used: full steady state (i.e. no steady inductive current drive) and the so-called 'hybrid' mode which combines inductive and non-inductive current drive to allow very long, but ultimately limited pulses. These would lead to different DEMO and reactor designs.

4.3.1 HYBRID SCENARIOS

The issues to be addressed are: stability (avoidance/minimisation of neoclassical tearing modes, NTMs); transport properties and how they will scale; current drive efficiency; integration with a cool, low erosion divertor plasma. CCFE's efforts have been focussed on the first two topics this year. In order to be more attractive than ELMy H-mode, the confinement has to be significantly better, but at present it is preferred to avoid the formation of internal transport barriers, as the steep gradients they produce make the plasma less easy to control. They are however the core of the steady state scenarios (and explain why they are regarded as more advanced and challenging).

Strong neoclassical tearing modes which cause enhanced transport must be avoided: the approach is to tailor the plasma current profile to keep q(0) above or close to unity to avoid triggers for NTMs from sawteeth, or if possible q_{min} above 1.5 or 2 to eliminate the m/n=3/2 and m/n=2/1 NTMs completely. An option on ITER, but not presently on JET, is to use feedback control to stabilise NTMs if they appear. Figure 4.3 shows what has been achieved, and the additional current drive that would be needed to sustain high performance against the natural diffusion of the current across the radius. The best scenarios being with q_{min} close to 1 or 2.

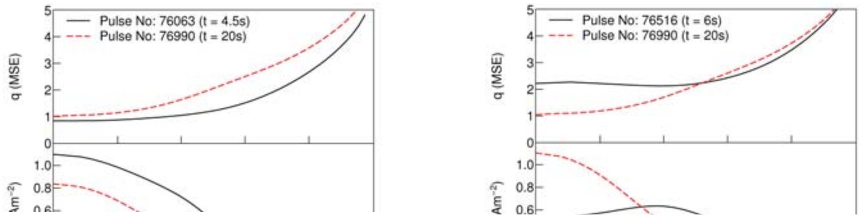

Figure 4.3: *Hybrid scenarios showing in the bottom plot the externally driven current profile needed to maintain the desired q profile against relaxation due to resistive diffusion. Left for q_{min} about unity and right for q_{min} just above 2. The q_{min}~1 case avoids all n=1-3 NTMs for about a resistive diffusion time*

Since plasmas in the hybrid domain commonly exhibit global confinement enhancement compared with present scalings based on the ELMy H-mode scenario, it is not obvious that the dependencies of these scalings can be used to extrapolate hybrid plasma performance to the domain of future devices. To further understand the confinement and transport properties of plasmas in present tokamaks and extrapolate them to larger devices an approach has been to test the underlying scaling in terms of dimensionless plasma parameters. For example, the plasma shape, q, β, ν^* and Mach number can be matched between present plasma devices of different size, and then ρ^*, the ratio of the ion orbit radius to the plasma size, can be varied over a large range across the operational domains of different devices to investigate the dependence of plasma performance on plasma size. This technique has previously been used to characterise the baseline ELMy H-mode scenario, and recent experiments have extended the application of this method to the hybrid scenario. Joint experiments between JET and DIII-D have now established a hybrid plasma database where ρ^* has been varied by a factor of 3 (Figure 4.5). In these conditions the normalised global energy confinement, $B\tau_E$, varies as $\rho^{*-2.14}$ and the kinetic profile data obtained will allow analysis of the local transport behaviour and comparison with models based on turbulent processes.

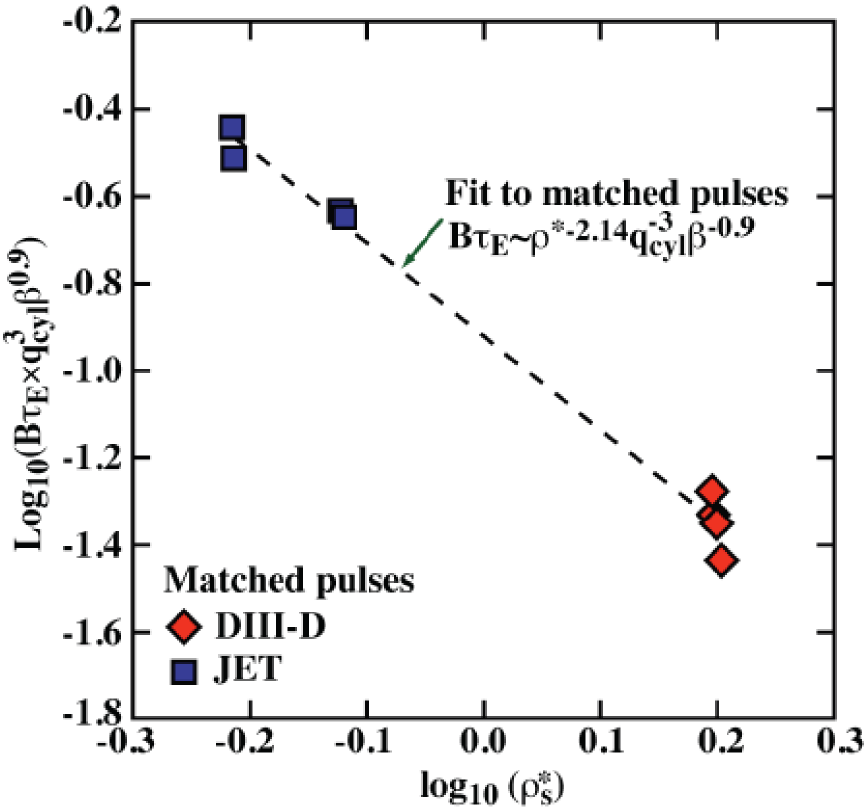

Figure 4.4: *Results from joint hybrid experiments on JET and DIII-D showing the scaling of energy confinement with ρ^*. The plasmas shown (four from JET and four from DIII-D) have been matched as closely as possible in terms of plasma shape, q, β, ν^* and Mach number. ρ^*_S is the scaled value of ρ^* (=$B^{-2/3}a^{-5/6}$) and small residual variations in q and β have been corrected assuming $B\tau_E \propto q^{-3}\beta^{0.9}$*

4.3.2 STEADY STATE SCENARIOS

These are distinguished by the current being driven fully non-inductively by a combination of bootstrap and other schemes such as neutral beam and RF current drive. In some cases these need the presence of an Internal Transport Barrier (ITB) where the turbulent transport is partly or completely suppressed, generally thought to be by strong flow-shear. They are demanding due to the risk of thermal instability of the ITB (the strong gradients drive their own flows which can thus stabilise the turbulence further), and the high pressures also provide further instability drive. However they also drive the non-inductive bootstrap current which reduces the need for (expensive and efficiency-limiting) external current drive systems such as RF and neutral beams. Here we describe progress in scenarios without ITBS; steps towards integrated steady-state scenarios needed for ITER, and some experiments on the physics of formation of ITBs, comparing JET and JT-60U.

A Steady state scenario development and integration

In order to complement the ITB approach, research has been directed towards steady state scenarios without ITBs. Very good performance has now been obtained – see Figure 4.5.

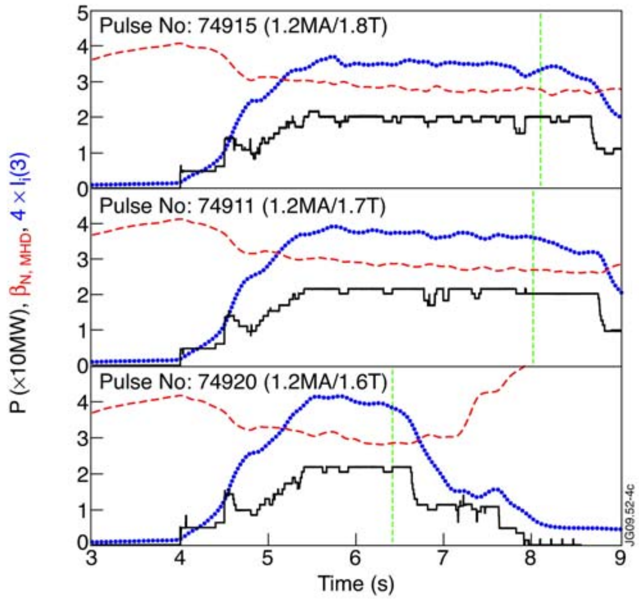

***Figure 4.5:** Three pulses in JET showing the high β_N now achieved in 'steady state' scenarios without an ITB (~4, compared to the reference value of 2.8 for ITER Q=5 steady state), together with the loss of stability as the current profile evolves in time (the vertical green line shows the onset of a 2,1 NTM)*

Substantial effort has been devoted in the past year towards operating at higher current and field, and moving closer towards a plasma with, simultaneously, more of the parameters ITER needs. Figure 4.6 illustrates the progress. The targets are essentially met in terms of β, confinement, thermal (vs fast ion) content, bootstrap and proximity of electron and ion temperature (JET has mainly ion heating normally, ITER will have primarily electron heating). The main gaps concern the collisionality ν^* and normalised Larmor radius ρ^*, which relate to the machine size and cannot be achieved on JET simultaneously, and the density (Greenwald fraction f_{GDL}) which addresses both fusion performance and the compatibility with a low erosion divertor.

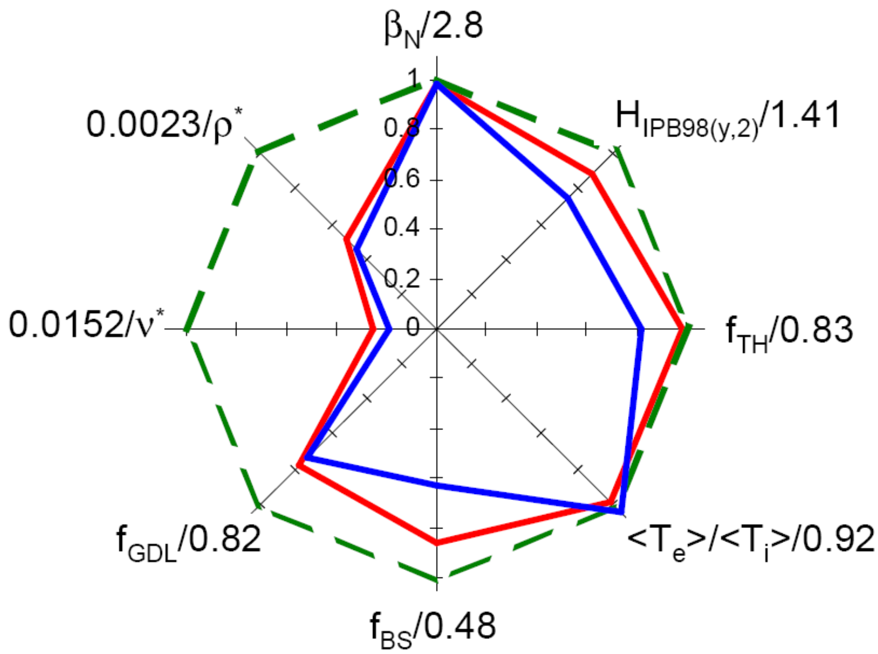

Figure 4.6: *Diagram showing how well the target normalised parameters have been achieved, compared to the goal of matching ITER's planned numbers. The target is the dashed octagon, the achievements in single pulses are the red (recent) and blue (previous). The red contour is for higher field and current (pulse 78052 2.7T / 1.8MA) than in the past (blue, 70069 2.3T / 1.5MA)*

B Internal Transport Barrier onset – comparison between JET and JT-60U

JT-60U (now closed for replacement by JT-60SA, a major new superconducting tokamak to operate alongside ITER) was the tokamak closest in size to JET, and was an ideal place to test whether the results and predictions from JET are robust, and also develop understanding. A series of experiments were conducted to explore similarities and differences in the formation of ITBs on these two devices – there are many contributing/complicating factors such as plasma shape, and level of toroidal field ripple. When a proper match is achieved of the most relevant profiles, very similar ITB characteristics were observed in JT-60U and JET, suggesting that they were governed by the same physics.

The *q*-profile plays a dominant role in the initial formation of the ITB, the mechanism seeming to be independent of the toroidal plasma rotation or rotational shear, as was also found in earlier JET experiments (see last year's Annual Report). After triggering, ITBs of similar strength were observed on both devices, but now the rotation shear plays a key role, stronger barriers being associated with higher rotation shear (as reported for JET last year). Here the ρ^* T parameter was used to compare ITB strength, although noting that earlier analysis has shown that this parameter may not necessarily scale to other devices.

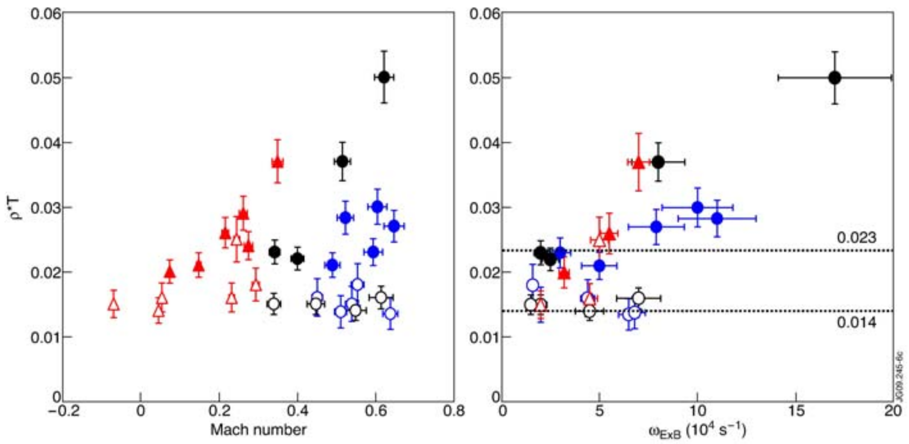

Figure 4.7: *a) The value of ρ*T_i plotted versus the central Mach number for various JT-60U (red triangles) and JET (black/blue dots) cases. The open symbols show the values at the time the ITB is triggered, while the closed symbols are at the time of maximum ITB strength. b) The same data but plotted versus the rotational shear, $\omega_{E\times B}$ in the vicinity of the ITB*

4.4 DIVERTOR ISSUES, AND POWER MANAGEMENT FOR THE ITER-LIKE WALL

There have been two main themes: developing plasma scenarios that will be compatible with the new wall, and improving understanding of the steady and transient power flux. For the latter a goal has been to help ITER use results from H, He and then D plasmas to guide operation in first D and then DT.

4.4.1 PREPARING PLASMAS FOR THE ITER-LIKE WALL ON JET

The new all-metal ITER-like Wall (ILW) now being installed in JET provides a highly realistic yet demanding simulation of ITER, and preparing for this has been a main aim of the 2009 campaigns. There are very good reasons why carbon (often in the form of CFCs – carbon-fibre-composites) are used – good thermal conductivity, no melting, tolerance to very high temperatures, mechanical strength. However they are considered unacceptable on a power plant – high retention of tritium due to formation of hydrocarbons and loss of thermo-mechanical properties under high neutron loads. The ILW comprises beryllium surfaces (either bulk Be or coatings on inconel structures) in the main chamber, and tungsten surfaces (either bulk W or coatings on CFC tiles) for the divertor. See Chapter 3 for details.

The plasma physics challenge is to achieve high performance plasmas without either excessive melting or other degradation of the metal surfaces, and without strong influx of the wall materials into the main plasma (in particular tungsten radiates very strongly and could degrade performance substantially). The prime tool is use of high densities and strong radiative cooling of the divertor plasma to reduce both the power flux and the individual ion energy incident on the tungsten. Figure 4.8 shows how the power between ELMs has been reduced by use of either nitrogen or neon puffed into the divertor, and how this augments the effect of strong deuterium puffing. The conclusion is that the power load appears to have been reduced enough to allow high power operation with the tungsten divertor. There are both temperature and total energy limits to respect, due to surface melting, and thermal stresses in the structure. In addition low plasma temperature is needed to reduce erosion and thus exposure of the underlying CFC of the W-coated divertor tiles.

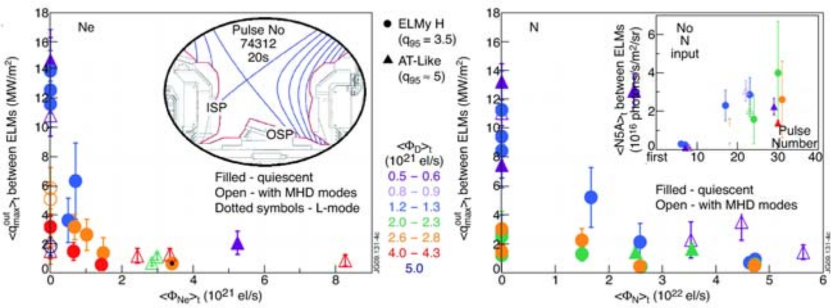

__Figure 4.8:__ Reduction in power density to the divertor achieved between ELMs due to puffing deuterium, neon and nitrogen. Peak power density on the outboard divertor target measured by fast thermography, Ne (left) or N (right). Inset (left): EFIT reconstruction of the divertor configuration. Inset (right): illustration of the shot to shot N legacy effect, by divertor intensity between ELMs of N^{4+} line at 20.93nm in shots with no N input, interspersed throughout the scans

4.4.2 Comparing H, He and D power exhaust in support of the early ITER programme

ITER will probably have to determine its operating programme in deuterium based on the performance in hydrogen and helium phases (and later the DT programme will be based on operation in deuterium). It is therefore important to understand what changes with the gas species, and thus some JET experiments were done in the H, He phases in 2009, and compared with deuterium. Figure 4.9 shows that the inter-ELM profiles appear broader in H, He than in D.

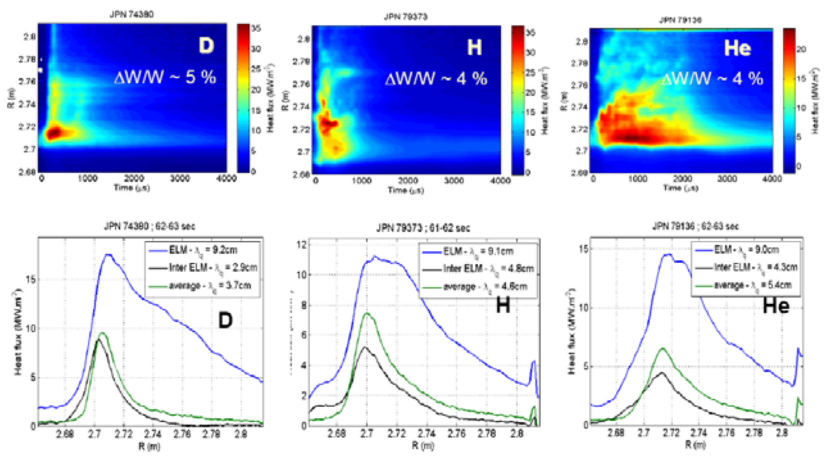

__Figure 4.9:__ Radial heat load profiles on the outer divertor target in comparable D, H and He plasmas. The bottom frames show inter-ELM, ELM and average profiles and their corresponding integral widths. The top frames show the temporal evolution of the heat load profiles during a typical, medium size ELM (fraction of plasma energy lost ΔW/W ~ 4-5%)

4.5 RESEARCH ON COMMON AND SPECIFIC TOPICS

In addition to the core scenario development activities above, a wide range of specialist studies are needed, on specific phenomena/topics, or on techniques. Some of these are indicated here.

4.5.1 DISRUPTIONS: STATISTICS AND ROOT CAUSES

Extensive work was done on plasma disruptions throughout the life of JET, in particular trying to identify the root causes and sequences. This revealed many features – disruptions occur across the operating space not only at the traditional operational boundaries of density, current, pressure (ß), but also in regimes which should be easier to operate without disruptions. There are many reasons for disruptions, only some of which are plasma-physics in origin, others include the consequences of the actions that are taken after some events such as loss of beam power. Figure 4.10 indicates this. The methodology developed could lead to improved ways of operating JET, and could prove very useful to the ITER team as they develop ITER's pulse preparation and real-time control and protection systems.

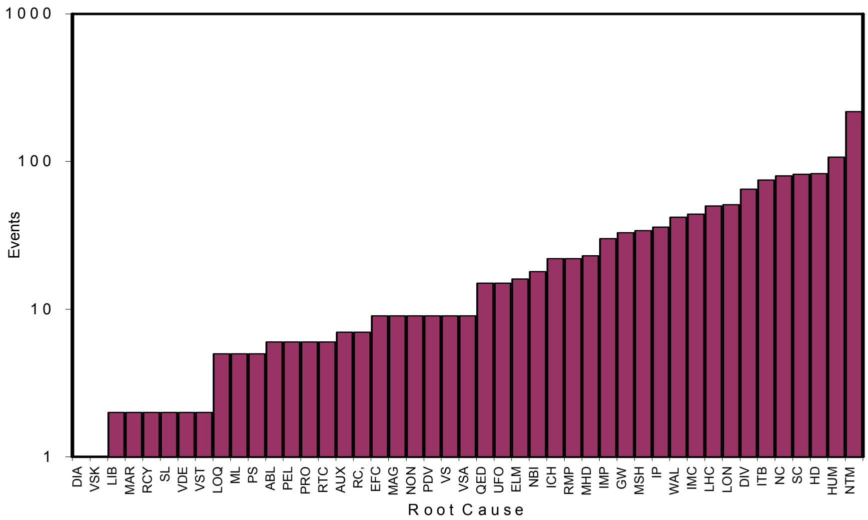

Figure 4.10: *The top ten root causes of Unintentional Disruptions on JET (67%):*

1. Neo-classical Tearing Modes (NTM)	*17%*
2. Human error (HUM)	*8.3%*
3. High density operation (HD)	*6.5%*
4. Shape control problems (SC)	*6.4%*
5. Density control problems (NC)	*6.2%*
6. Internal Transport Barrier (ITB)	*5.9%*
7. No divertor cryo-pumping (DIV)	*5.1%*
8. Low density error field mode (LON)	*4.0%*
9. Lower Hybrid Current Drive (LHC)	*3.9%*
10. Impurity control problems (IMC)	*3.4%*

4.5.2 LOWER HYBRID CURRENT DRIVE – NEW TECHNIQUE TO MEASURE LOCATION

Lower Hybrid Current Drive (LHCD) is used on JET to help access to advanced scenarios by changing the current density in the outer regions of the plasma. There has been, however, concern that the current is not as large as predicted and not located where ray tracing suggests. To address this, experiments with modulation of the LHCD power were performed. The understanding comes via the analysis – the LH waves are damped on fast electrons, and a Fokker-Planck code is used to calculate the knock-on effect on the thermal electrons as measured by electron cyclotron emission (ECE). Figure 4.11 illustrates the results of this very complex analysis. The discrepancy between the two power deposition models shows that there are features of the LH propagation and deposition that are not fully understood, but now revealed more clearly by this technique.

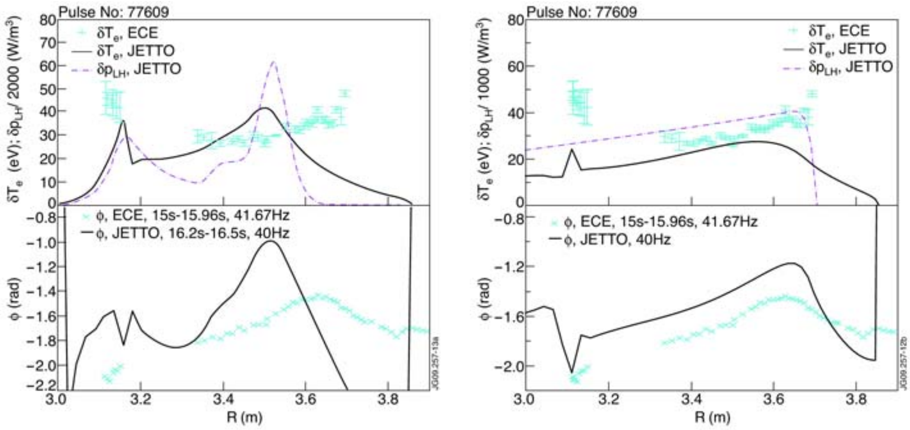

Figure 4.11: *Result of modulation experiments to test LHCD deposition profiles. Left: calculated deposition from ray-tracing and Fokker-Plank, and calculated temperature modulation from JETTO based on a Critical Gradient Model for the transport. The measured data for the modulated Te are shown as blue crosses. Right: the same data, but this time with an empirical power deposition profile to give a better fit to the data*

4.5.3 ICRH OPTIMISATION

Following the crucial demonstration that ICRH launchers can be made ELM-resilient by suitable use of resonant circuits or other techniques to avoid reflected power from turning the power supplies off, the work focussed on improving coupling and arc detection systems. CCFE has been responsible for investigating the so-called Sub-Harmonic Arc Detection system (SHAD, developed under a collaboration with CEA and IPP), and the Advanced Wave Amplitude Comparison System (AWACS). The SHAD has been tested on the ITER-like antenna, and it has been shown that there is a low frequency (below the driver frequency) associated with arcs, even if its behaviour is not entirely reproducible it detects arcs not found by some other systems, so has potential in a multiple protection system. The AWACS system looks at changes in the reflected waves from various points in the systems, and has been shown to be more sensitive than the traditional Voltage Standing Wave Ratio (VSWR system) and has been used effectively on JET to protect the External Conjugate T system.

4.5.4 LHCD AND ICRH INDUCED LOCALISED HEAT-LOADS

In view of the new JET wall, there is concern that under certain conditions both LHCD and ICRH can lead to localised heat-loads that could lead to melting of some plasma-facing components (PFCs). The new infra-red camera diagnostics on JET have allowed a better characterisation of this phenomena and thermal models have been developed to deduce associated heat loads. For the case of LHCD, lost power in the plasma edge can lead to the formation of energetic electron beams that result in hot spots when intercepting limiters and other PFCs. It was possible to estimate how far in front of the antenna these fast electrons can be generated, by mapping the hotspots observed back along fieldlines to the region in front of the antenna. The result is that the region is much wider (several cm) than predicted by simple models (a few mm).

For the ICRH, ions can be accelerated in the rectified potential surrounding the antennas and can lead to overheating of the antenna structure. Particular attention was given in developing a realistic thermal model of the antenna structure taking into account dust deposition in order to correctly describe the temperature increase observed by IR cameras.

4.5.5 PELLET INJECTION – ELM MITIGATION

A new High Frequency Pellet Injector (HFPI) has been implemented on JET, both for fuelling and also for ELM mitigation. The latter builds on earlier work on ASDEX Upgrade showing that pellets can trigger ELMs, and thus provide a potential way to make sure that ELMs are tried before the available energy and thus the potential ELM size gets too large. Although the HFPI is not yet routinely available, it has already been shown that pellets need to penetrate a certain distance (several cm) in order to trigger an ELM, and this has led to a modification to the injector to increase the size of the ‘small’ pellets.

4.5.6 FAST ION PHYSICS AND GAMMA SPECTROSCOPY

Spectroscopy of gamma rays is a tool to gain information on the fast ion distribution in JET, using known reactions between the fast ions (H, ^{3}He, ^{4}He, D) with impurities (C, Be mainly). This can be either via threshold reactions showing that ions exist above some threshold energy of the reaction, or by means of Doppler spectroscopy of the emitted gammas to draw conclusions about the distribution or ‘temperature’ of the fast ions. This is an activity presently carried out in close collaboration with the team providing new spectrometers.

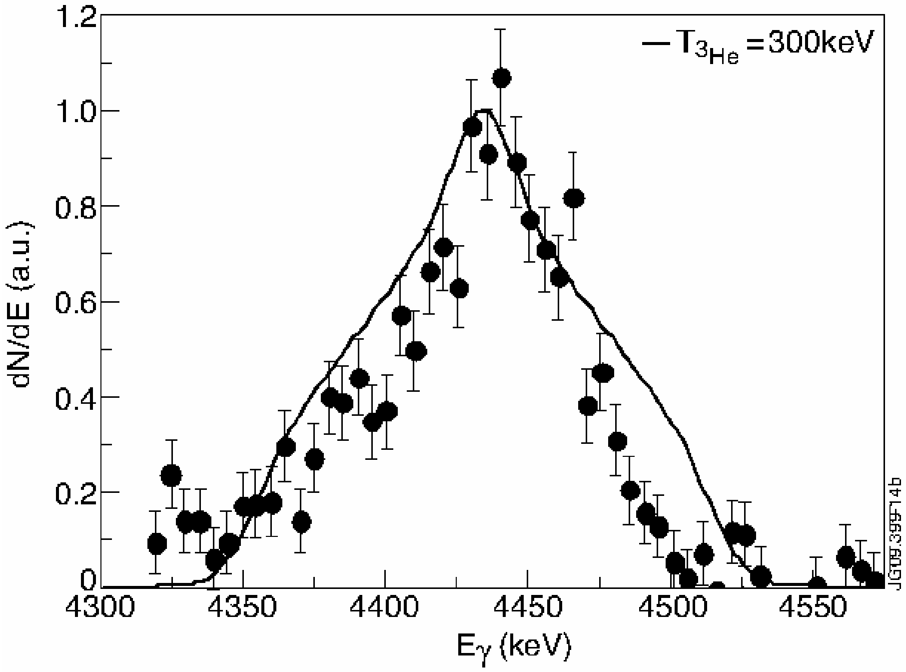

Figure 4.12: *Example of use of high resolution gamma spectroscopy (collaboration with ENEA/CNISM - Italy) to determine the "temperature" of fast ions, from the* $^{9}Be(^{3}He,p\gamma)^{11}B$ *reaction*

4.6 PROGRESS ON ENHANCEMENT PROJECTS

CCFE has been involved in six enhancement projects as an Association (of course all projects that involve equipment to be installed on JET involve CCFE as operator). These are:

- Neutral Beam Enhancement project (increased power and pulse length) – CCFE lead;
- Enhanced Radial Field Amplifier, ERFA (for vertical position control in presence of large ELMs) – CCFE lead;
- Plasma Control Upgrade (vertical position and shape control improvements, linked to ERFA) – CCFE leads a subproject;
- Spectroscopy for the ITER-like Wall (suite of spectrometry changes to measure the new species and locations needed for the ILW exploitation phase) – CCFE leads a consortium;
- Profile reflectometer (a new multi-band fast swept reflectometer to provide high space and time resolution density profiles on JET) – CCFE leads a consortium;
- Gamma diagnostics upgrades (high resolution spectrometer and a set of neutron attenuators to reject neutron noise from the gamma signals) – CCFE provides project physicist;
- Three projects on high resolution spectrometer, neutron attenuators for gamma-ray profile cameras and collimators for gamma-spectrometer to reduce neutron-induced background.

Brief status reports on the first five are given below.

4.6.1 Neutral Beam Enhancement

The aims of this project are, in brief:

- To increase the deuterium NB power to at least 34MW;
- To increase the beam pulse length to 20s;
- To improve NB reliability and availability.

The performance of the improved ion source, accelerator and neutraliser combination has been demonstrated in Figure 4.13.

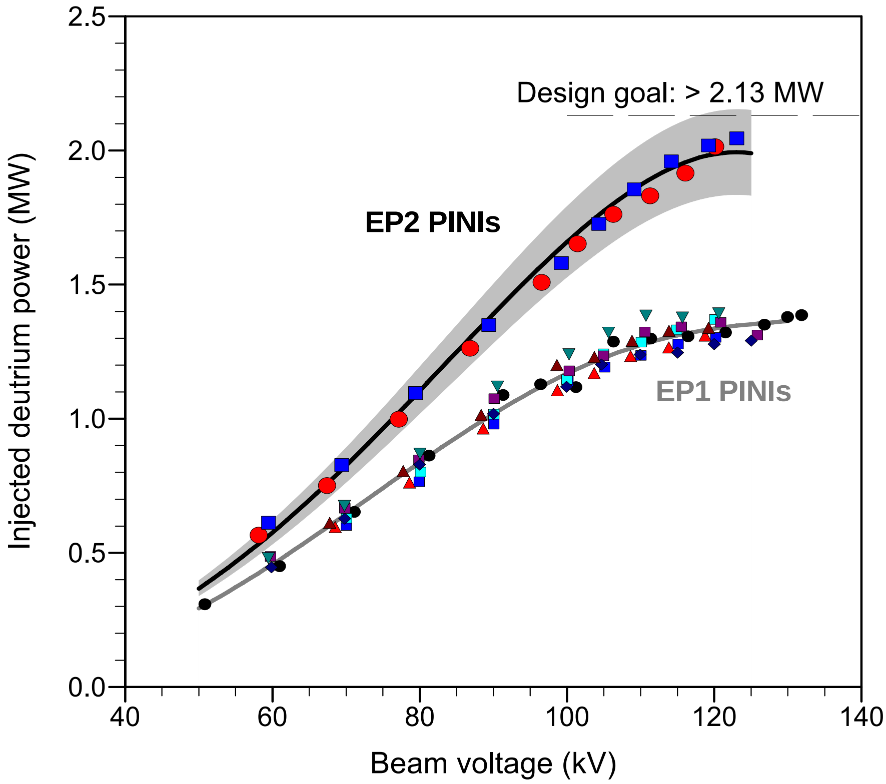

***Figure 4.13**: Demonstration of the performance improvement of the revised PINIs, using larger holes in the extraction and acceleration grids and a revised configuration of the confinement magnets around the plasma source*

All the in-vessel components have been procured (the new high heat flux elements, mainly actively cooled), the large new high voltage power supplies are complete and ready for commissioning. Almost all other components are now on site. The preparations for the installation of the new equipment is well under way (see Chapter 3, JET Operations). All the components should be installed inside JET and on the neutral beam systems during the shutdown, with commissioning in early to mid 2011. It is worth noting that the operational limits of the new wall will mean that the full power will only be deployed gradually, as we learn how to operate JET keeping the power load on the divertor to an acceptable level (e.g. via radiative divertor scenarios, see above).

4.6.2 Enhanced Radial Field Amplifier

It has been known for some time that large ELMs can lead to plasma disruptions due to overload of the vertical position control system on JET. To avoid this, a new system with higher current (i.e. a stronger response) was needed, the so-called ERFA. This 5kA, 12kV IGBT-based supply replaces the existing, now obsolescent 2.5kA, 10kV GTO-based amplifier. The project has been completed, and the new supply used for several months in 2009, including for reliable control of 4.5MA plasmas with large ELMs. This new supply will be critical for operation with the more 'fragile' ILW in 2011. To make optimal use of the supply, the configuration of the coils on the torus was changed. Optimising those and commissioning the whole system was done in June and July 2009. A post-installation modification was made, using engineering contingency in the transformer design, to increase the charging current to the capacitor that supplies the H-bridges, to compensate for larger-than-expected dissipation in the passive structures on the torus.

4.6.3 Plasma control upgrade

This project was led by ENEA (Italy), with CCFE having responsibility for the procurement and installation of the new hardware for the controller, converting magnetic measurements to drive for the ERFA power supply. This collaborative project covering several aspects of the control system as well as hardware design, development and manufacture, involved three Associations (ENEA, CCFE and the Portuguese Association, IST). Despite the complexity and developments needed (the ATCA digitisers were developed specially), the integrated commissioning of the control system and the new power supply was successful, and as mentioned above the combined system successfully used to control high current plasmas with large ELMs, meeting the objective (Figure 4.14).

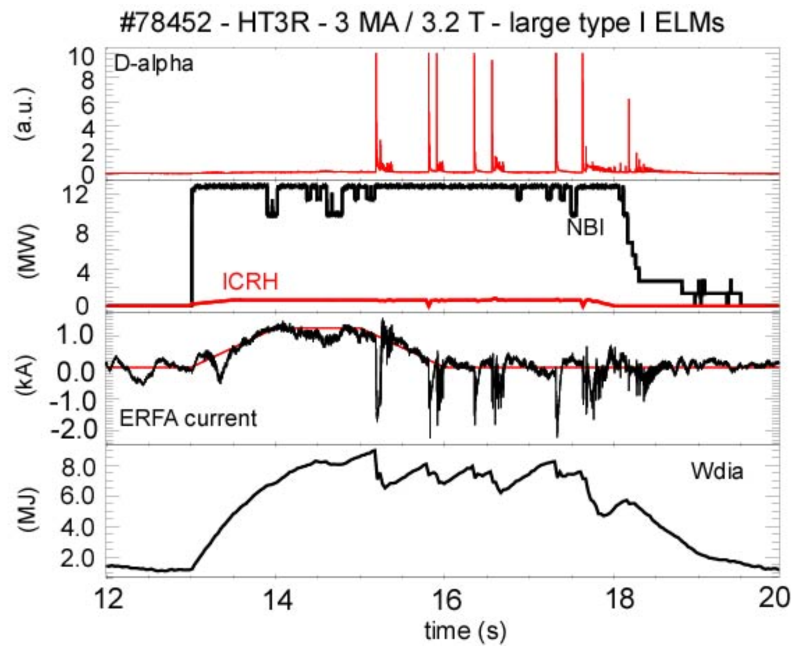

***Figure 4.14**: Plasma controlled with the new controller (PCU project) and ERFA, showing the successful integration of the controller and the power supply*

4.6.4 PROFILE REFLECTOMETER

A collaborative project was set up with CCFE (lead), CEA (France) and IST to construct the microwave electronics, sources, optics and data acquisition. This 6-band system covering Q, V, W and D wavebands operates using both X-mode and O-mode reflectometers, all swept fast, to provide density profiles with high space and time resolution – target sub cm space resolution, sub 15microsecond time resolution from very low densities (below $1x10^{18}m^{-3}$). Examples are shown in Figure 4.15.

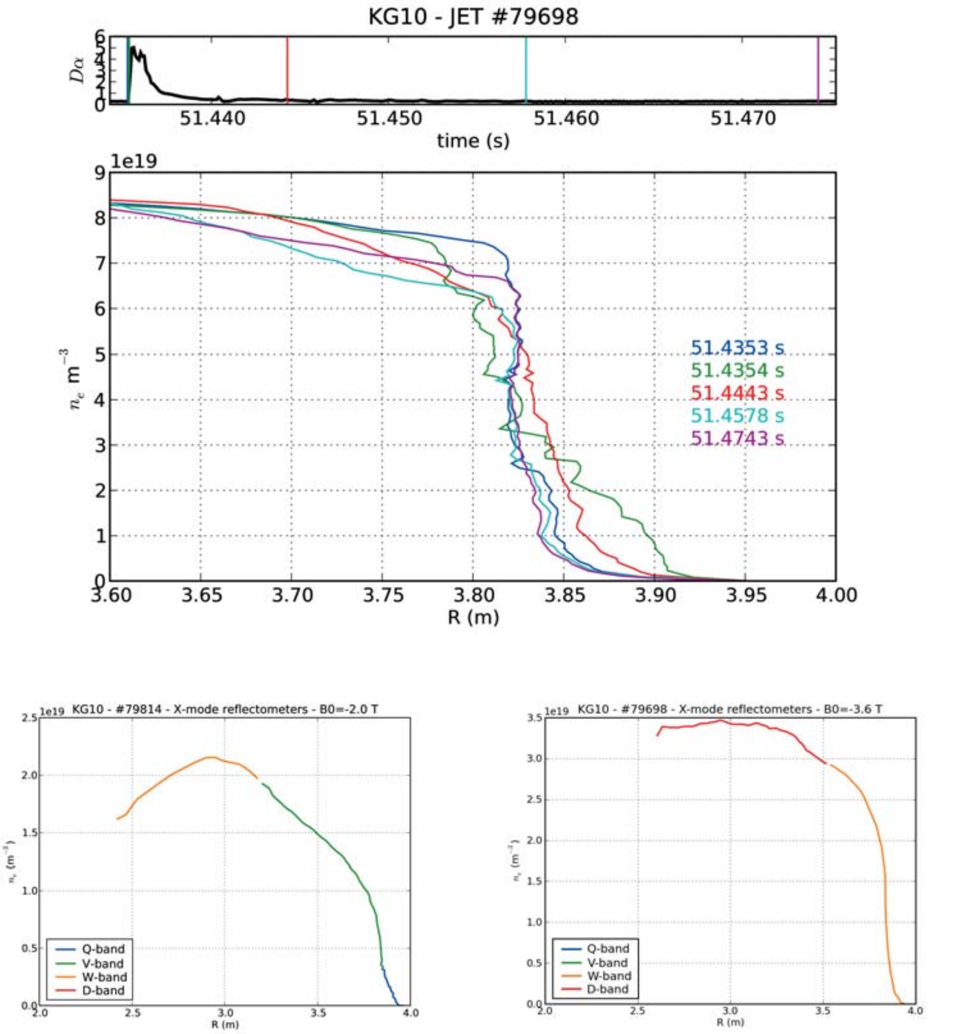

***Figure 4.15**: Density profiles from the new profile reflectometer, during an ELM and at 2T and 3.6T*

4.6.5 SPECTROSCOPY FOR THE ITER-LIKE WALL

This multi-faceted project is still in progress, with the aim to complete by the end of the ILW shutdown. As an example, the upgrade to the view of the new tungsten divertor is shown in Figure 4.16. An improved view has been designed (seeing more of the divertor), along with improved spectrometers, and CCD cameras to record the spectra (more efficient, faster). This will be vital for measurements of the erosion of tungsten.

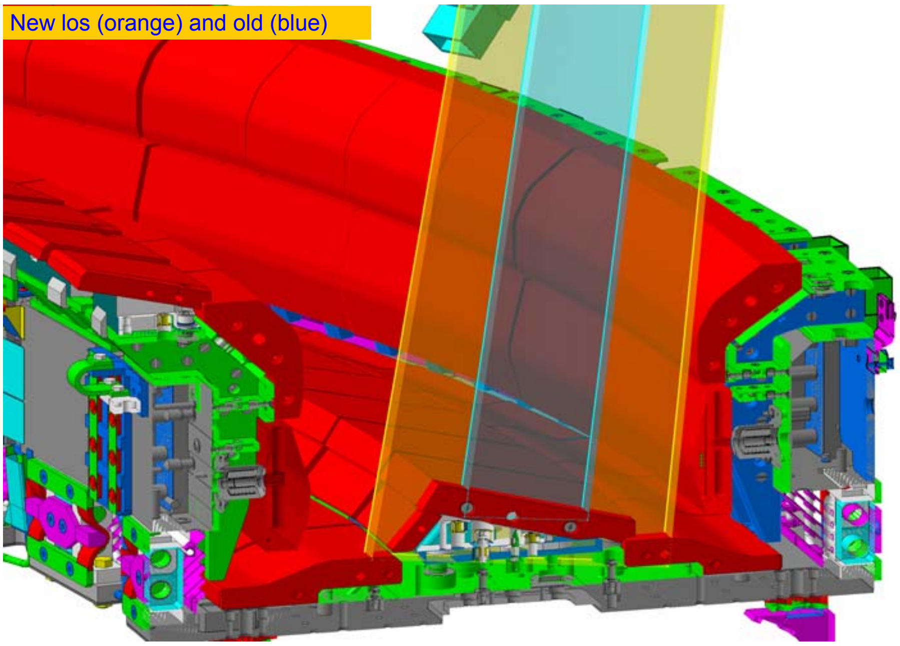

Figure 4.16*: Improved view of the divertor for the wide-band mirror-coupled multi-spectrometer system (KT3). The new (orange) view allows the whole of the bulk-tungsten tile to be monitored, from 200 to 1300nm*

4.7 FUTURE PLANS

During 2010 and to a lesser extent thereafter, scientific effort will be devoted to analysis and interpretation of the results accumulated during the long campaigns in 2008 and 2009, with the IAEA Fusion Energy Conference in October 2010 a prime target. When the shutdown is complete, JET will be a very different machine. There will be new results from the very first plasma as we see how the Be and W wall behaves, and a very cautious start to avoid melting of the precisely machined plasma facing components. The experiments will be more focussed initially on charactering the wall, erosion, heating and developing scenarios compatible with the metal wall. Only later will we move towards higher performance. The techniques developed in 2009 for enhanced radiative power losses with injected impurities (Ne, N) will be key.

There will be some activity preparing for a DT campaign in perhaps 2015.

Finally there will be a lot of intense effort to complete the remaining enhancement projects, notably the neutral beam and the spectroscopy projects.